Incident Chronology at Peach Bottom Atomic Power Plant: 1974- 2021

Peach Bottom-1 was a 40 megawatt (“MWt”) High Temperature Graphite Moderated Reactor that operated from 1966-1974.

Peach Bottom-1 is still in the rate base although there are 0 fuel assemblies stored on site. Exelon/Spin Co plan to delay final decommissioning until PB 2 and PB 3 are shut down. PB 1 has combined - Qualified and Non-Qualifed Trusts - $166 million in its decommissioning fund.
The most recent decommissioning funding assurance report can be found on the NRC website
under Accession number ML21055A776 for Peach Bottom 1, 2, and 3, Limerick, and Salem.

The NRC identified a violation at Peach Bottom-1 on November 10, 2020.
A violation of "low significance” was identified via a “remote inspection.”
"Specifically, from approximately July 1978, until October 16, 2020, the radiological surveys and water accumulation inspections had not routinely included the areas of containment that may be accessed by unlocking barrier B-14 on the Refueling Floor in order to survey and inspect theintermediate and ground-level floors.”
Peach Bottom 2 & 3 , are 1,065 megawatt Boiling Water Reactor designed by General Electric and engineered by Bechtel. Both reactors began operation in July, 1974, but had their licensees extended by the Nuclear Regulatory Commission (NRC) and are expected to operate though 2034.
The Nuclear Regulatory Commission (NRC) and the Institute for Nuclear Power Operations (INPO) have clearly demonstrated that Philadelphia Electric's (PECO), renamed Exelon in 2000, performance has historically been lackadaisical and sub-par. In order to put Peach Bottom's operating history into perspective, it is necessary to review PECO's plant legacy.
According to Eric Epstein, Chairman, TMI-Alert: "Managerial problems further aggravate and compound the inherent flaws with Peach Bottom's reactor and containment structure." The reactors at Peach Bottom are General Electric (GE) Boiling Water Reactors (BWR). Epstein noted, "The GE-BWR is an obsolete design no longer built or constructed. Many in the industry feel it is inferior to Pressurized Water Reactors. Obviously the age of the reactors, and the subsequent embrittlement that ensues, further erode the margin of safety."
Peach Bottom's Mark 1 containment structure has been demonstrated by Sandia Laboratories to be vulnerable during a core melt accident. Epstein explained: "The containment is likely to fail during a core melt accident [like Three Mile Island] allowing radiation to escape directly into the environment." Nuclear industry officials say the problem with the Mark 1 is that it is too small and wasn't designed to withstand the high pressure it is supposed to resist.

1974 - Peach Bottom came on line at a cost of $375 per kilowatt.

March, 1983 - A spill of 25,000 gallons of radioactive water was reported at the plant.

June 1983 - PECO was fined $40,000 by the NRC for a valve violation.

July 1983 - Philadelphia Electric identified cracks in their cooling pipes.

1983 -1987 - PE was issued a number of violation notices that cost the utility $485,000 in civil penalties. All the violations involved failure of personnel to follow procedures.
Examples of violations include: workers entering high radiation areas without required radiation protection; improperly controlling access keys to the plant's high radiation areas; discrepancies in workers' radiation work permits; improper packing of low level radioactive wastes; leaving air lines open while the reactor was producing power between August 12 and September 10, 1982. With these lines open the containment could not be sealed against radiation escape in the event of an accident; allowing excessive leakage from the containment building; improperly setting instrument valves which made the plant incapable of providing back-up signals to automatically shut the reactor down in the event of an accident (Lancaster Independent Press, April, 1988).
Ronald Haynes, the NRC's regional administrator, stated, "These violations demonstrate the need for improvements in the control of operational activity."

June 19, 1984 - The NRC cited PECO for five alleged violations of technical specifications at Units 2 and 3. The NRC also proposed a $30,000 fine.
Three of the alleged violations "involved exceeding the maximum allowable reactor heatup rate, allowing pressure in the reactor to go beyond the limit specified for a given temperature and failing to recognize that a control rod was inserted into the reactor at a rate slower than required."
Continued on the next page...
The other two violations "involved changes to facility procedures in 1977-1979 that were not properly reviewed and three instances in 1980 and 1983 of failures to follow procedures." These violations were identified by an inspector between January 5 and 20, 1984 (United States Nuclear Regulatory Commission, Office of Public Affairs Region I, June 19, 1984).

December 1984 - An Institute of Nuclear Power Operations (INPO)

evaluation found "clear evidence of declining performance". In addition, the report claimed that these problems were "longstanding."
- 1985 - An NRC inspector observed a Peach Bottom operator dozing at the controls. No safety violation was charged.

June 1985 - The plant was shut down due to mechanical problems.

July 26, 1985 - PECO was accused of pressuring the United Way to deny eligibility to Del-AWARE Unlimited, Inc., "a group that is lobbying against the water-diversion project that would supply the utility's Limerick power plant...I wouldn't go as far as to use the word *threatened*, but the message was clear. PE would stop funding if Del-AWARE were made eligible under the donor-option program." (The Philadelphia Inquirer, Front Page, Friday, July 26, 1985.)

October 1985 - A emergency evacuation drill turned into a serious incident when Unit-2 reactor's water level dropped.

October 1985 - PECO is fined by the Occupational Safety and Health Administration (OSHA) for safety violations leading to the death of an **employee.**

December 1985 - An INPO study (as reported by The Nuclear Monitor) concluded that PECO's performance continued to decline. A subsequent letter written in January by Zack Pate, President of INPO, to PECO Chairman John Everett, said "standards of performance at the station are unacceptably low."
Problems were identified in operations and maintenance, radiological protection, material condition and housekeeping. INPO also identified several non-licensed operators reading unauthorized materials. A total of 431 shortfalls were identified; 141 involved personnel performance. Pate noted,", and "we ... have considerable concern that the station's substandard radiological control practices may lead to the spread of contamination off-site, or some other serious radiological event.
Continued on the next page...
Pate concluded, "From my assessment, this pattern will not change, and personnel performance at Peach Bottom will not improve, until you personally acknowledge the need and communicate the need, for real change to your organization."
- February 1, 1986 to May 31, 1987 - The SALP for this period indicated PECO's performance was "unacceptable" because of the operators' inattentiveness and management's "inability to identify and correct operator conduct in other areas."
Among the incidents cited by the NRC: security guards were overworked, and one guard was found asleep on the job; 36,000 gallons of "mildly radioactive water" leaked into the

Susquehanna River; PECO mislaid data on radioactive waste classification causing misclassification of a waste shipment; at the turbine building on March 4, 1987, Unit 3 a major fire occurred at the maintenance cage.

March 1986 - A checking system was bypassed and automatic backups were bypassed by a supervisor during an inappropriate withdrawal of a control rod from the reactor core.

April 1986 - An explosion and fire occurred at the plant's substation for emergency power.

June 1986 - The NRC's annual report concluded that Peach Bottom was "operated by well qualified individuals with a positive attitude toward their positions for nuclear safety."

June 1986 - Unit-2 was shut down when a cooling system pipe sprang a leak.

June 11, 1986 - A $200,000 fine for failing to pay attention to detail was issued. The incident involved the withdrawal of control rods. A highlevel, NRC administrator noted that these violations indicated a continued "pattern of inattention to detail" and "a general complacent attitude." The original fine was set at a $100,000, but doubled because of PE's history. In addition, the NRC reported 17 violations.

July 16, 1986 - While testifying before Congressman Markey's Committee, the NRC revealed that Peach Bottom was one of the 10 most hazardous plants in the country. The underlying reason appeared to be that PECO's attention was focused on the construction and startup of Limerick, rather than the safe operation of Peach Bottom.

August 1986 - The NRC reported that there were 26 cracks in Peach Bottom's two operating reactors (Units 2 and 3).

December, 1986 - The NRC reported that a health physicist was illegally fired for whistleblowing.

February 18, 1987 - An NRC study said Peach Bottom's reactors were more likely to release radiation in the event of a core-melt accident.

March 4, 1987 - At the turbine building at Unit 3 a major fire occurred at the maintenance cage.
The NRC identified several precursor problems with fire protection on the following dates: April 10, May 30 and November 1, 1985. Another

related problem was documented on January 19, 1990.

March 15, 1987 - The NRC levied a $50,000 against PECO for illegally dismissing a worker who was exposed to radioactive gas.

March 31, 1987 - Peach Bottom was indefinitely shutdown. Operators were found sleeping on the job, playing video games, engaging in rubber band and paper ball fights, and reading unauthorized material.

May 1987 - The NRC reported that areas of high radioactivity were not properly marked.

May 1987 - An NRC inspection report revealed 33 operator errors in the past two years as well as cases of operator inattention and poor reaction.

July 15, 1987 - Senior Health Physics Technician, George Fields, filed a lawsuit against PECO for exposing him to dangerous levels of radioactive gas.

September 1987 - An INPO evaluation ranked the plant in the lowest category.

September 30, 1987 - A contractor employee attempted to enter a protected site while intoxicated. Later cocaine was found in the parking lot and in the guard's bathroom.

October 1987 - An INPO visit (as reported by The Nuclear Monitor) found that since shutdown, "little clearly demonstrable action has been taken regarding corporate management's accountability for conditions at the station."
"Control of drawings, procedures, and other documents used by operations personnel was identified as a problem at Peach Bottom ... in 1980. During the recent plant evaluation, 22 of 23 drawings reviewed in the radwaste control room were out of date by as many as 15 revisions. Outdated or unapproved drawings and procedures were also noted at various locations in the turbine building and the auxiliary room."
"[T] here were more than 6,000 open maintenance requests, 300 outstanding money tickets (minor maintenance requests), and 1,200 additional items requiring maintenance on various lists ... 586 preventive maintenance activities ... have been outstanding since June 1986."

October 5, 1987 - A loss of Power at Unit-3 resulted in a containment isolation and a loss of shutdown cooling.

October 8, 1987 - The NRC deferred a review of PECO's reorganization plan because of their failure to address corporate weaknesses.

October 9, 1987 - Philadelphia Electric announced a corporate reorganization plan.

October 29, 1987 - The forced shutdown is costing Philadelphia Electric an additional $5 million a month for replacement electricity. ("Patriot News".)

November, 1987 - A report published by Public Citizen revealed that $400 million was spent on repairs at Peach Bottom between 1981 and 1985. This amount was the highest expended at any of the nation's nuclear power plants.

November, 1987 - The FBI discovered a drug distribution ring at Peach Bottom.(For more details see: January 8, 1988; February, 1988; May 2, 1988; November, 1989; and, May 10, 1999.)

January 8, 1988 - A maintenance sub-foreman pleaded guilty to involvement in a conspiracy to distribute methamphetamine. He is one of six who were indicted last year in a conspiracy to distribute methamphetamine. (For more details see: November, 1987; May 2, 1988; and November, 1989.)

January 11, 1988 - INPO President Zack Pate strongly criticized Philadelphia Electric's management and their revised reorganization plan.
Pate noted that, "The fundamental approach to nuclear operational management at Philadelphia Electric Company has not changed and is unlikely to change noticeably in the foreseeable future." He added, "success ultimately depends on the individual managers in key line positions. Since for the most part, the same managers who have been ineffective in this area for years are in the key line positions in the new organization, substantial improvement is unlikely." Pate concluded, "Major changes in the corporate culture at PECO are required. The recently announced reorganization plan will not achieve this" (The Nuclear Monitor, February 22, 1988, pp.1-2).

January 26, 1988 - Governor Robert P. Casey formally petitioned the NRC for public hearings on PECO's management.

January 27, 1988 - PECO reportedly lost $58 million due to the NRC's shutdown of Peach Bottom. Earnings per share were shaved from $2.60 a share in 1986 to $2.33.

February 3, 1988 - John H. Austin resigned as president of PE after a unusually critical report by the Institute of Nuclear Power Operations (INPO) was published. The report asserted that Peach Bottom "was an embarrassment to the industry and to the nation." Zack T. Pate, president of INPO, added, "The grossly unprofessional behavior by a wide range of shift personnel ... reflects a major breakdown in the management of a nuclear facility."

February, 1988 - The PUC ordered PE to reduce rates by a $37 million a year until Peach Bottom is allowed to restart.

February, 1988 - Four PECO employees were indicted for allegedly distributing drugs at Peach Bottom. PECO maintained that the workers were not working in areas affecting safety. (For more details see: November, 1987; January 8, 1988; May 2, 1988; November, 1989; and, May 10, 1999)

February 9, 1988 - In a editorial, The Patriot News concluded: "PECO's management failed in that basic responsibility to the company's stockholders, to the federal regulations they are required to abide by and the public that was put at risk by this slipshod performance."

March 17, 1988 - PE officials acknowledged that the plant will not be ready for restart until the "...fall frame time." This prediction would mean that the plant would be shut down for "at last 18 months, costing the company $125 million, based on its current rate of expenditures for replacement power and a penalty imposed by the state Public Utility Commission" (The Patriot News, March 17, 1988, p.B-9).

March 29, 1988 - The Public Citizen's Critical Mass Energy Project rated Peach Bottom as one of the poorest rated plants in the country based on the following criteria: "average lifetime operating efficiency; 1987 operating efficiency; average operating and maintenance costs during 1985 and 1986; average capital additions costs from 1982 to 1986; most recent SALP ratings; number of scrams during 1985 and 1986; average annual fines from 1985 to 1987; worker exposures from 1984 through 1986; LERs in 1985 and 1986; potential accident consequences derived through the CRAC-2 computer code" (The Nuclear Monitor, May 2, 1988, p.6).
An NRC's evaluation of the plant's management performance rated Peach Bottom as the *eighth worst in the country*.

April 7, 1988 - The Janny Montgomery Scott basic report on Philadelphia Electric noted that PE still faces many hurdles, including: "...further intense scrutiny from the regulatory commissions, and the uncertainty of future rate relief. Accordingly, the stock remains suitable primarily for investors willing to assume above-average risk." And, "Certainly, the extensive nature of the management reorganization will require time to evolve, but many deep-rooted problems such as those initially developed at Peach Bottom are corrected now."

April 13, 1988 - J. Lee Everett "retired" as Chairman and Chief Executive Officer of Philadelphia Electric as a direct result of the harsh criticism from a January 12, 1988 report released by the Institute of Nuclear Power Operations (Refer to February 3, 1988).

May, 1988 - Bessie Howard filed a complaint with the United States Department of Labor alleging that she was fired "in retaliation for her identification of safety problems relating to security at Peach Bottom." Beginning on January 24, 1988, Mrs. Howard reported that another security guard was sleeping on the job. She continued to report the matter until she was fired On March 16, 1988, by Burns Security, the security contractor for Peach Bottom. She was classified "status nine" and prohibited from working at other nuclear power plants or government facilities.
- A report issued by the NRC indicated "that security personnel were forced to work excessively long hours, sometimes up to 12 hour shifts; were not given meal breaks, and were required to remain at posts for extended periods of time without being rotated to other posts, a violation of NRC regulations" (York Daily Record, May 1988).

May 2, 1988 - Four Peach Bottom employees were charged with conspiracy to distribute methamphetamine at the plant and elsewhere. Thirteen people, most of whom work at Peach Bottom, have been charged with drug-trafficking as a result of an FBI investigation. (For more details see: November, 1987; January 8, 1988; February, 1988; November, 1989; and May 10, 1999.)

Spring 1988 - A cot for sleeping on the job was removed from an area located near the control room, and the NRC acknowledged knowing of its presence prior to its removal.

June 6, 1988 - The NRC warned that the "effort to make sure the Peach Bottom nuclear power plant is run safely is by no means a sure thing " (Centre Daily News, June 1, 1988, A-6).

June 16, 1988 - The General Counsel to the Governor of Pennsylvania submitted comments on the Revised Plan for Restart of Peach Bottom Atomic Power Station and the Actions of Philadelphia Electric Company Leading Up to and Succeeding the March 31, 1987 Shutdown Order of the Nuclear Regulatory Commission.

Counsel noted, "The plan on the whole remains too general to permit proper evaluation. Some of the most crucial areas, for example, the responsibility for individual operators and those managers who are retained for previous misconduct and the justifications for their retention, remain undisclosed. Certain basic problems, such as drug abuse and previous sanctions against whistleblowers, are either not addressed at all or are insufficiently addressed. Independent assessment organizations need even greater independence and must satisfactorily demonstrate reanalysis of problem reports (such as Significant Operating Events and vendor reports) that may have triggered inadequate responses over the last few years. Finally, and most importantly, the reforms generally proposed must be reduced to specific, clear, verifiable commitments and proper avenues outlined for verification."

July 27, 1988 - Public Service Enterprise Group Incorporated and its subsidiary Public Service Electric and Gas Company filed and action in the United States District Court to recover damages resulting for the NRC's shutdown of Peach Bottom. On the same in the same court, Atlantic City Electric Company and Delmarva Power and Light Company filed similar suits against Philadelphia Electric. The suits allege that PECO breached its contract under the Owners Agreement. Several tort claims were also filed, however no dollar amounts were specified. (Based on information from Philadelphia Electric Company's "Report to Shareholders Third Quarter 1988.") (See April 4, 1992 for settlement agreement.)

August, 1988 - Peach Bottom's security contractor was replaced due to incompetence.

August 11, 1988 - The NRC proposed fining PECO $1.25 million for "management problems that resulted in a forced shutdown of the company's Peach Bottom nuclear plant." In addition, the NRC proposed fining 33 reactor operators for sleeping on the job, playing video games, engaging in spit ball battles, and other unprofessional activities. Fines of $500 to $1,000 were recommended. PECO spokesperson Williams Jones disclosed that the company "has lost more than $90 million since the NRC ordered Peach Bottom shutdown..." (Patriot News, August 12, 1988).

August 17, 1988 - Joseph Rhodes, Jr., a member of the Pennsylvania Public Utility Commission, suggested that a deal between PECO and the NRC might have been made in order to get Peach Bottom back on line. In letters to NRC Chairman Lando Zech and PECO CEO Joseph Paquette, Jr., Rhodes stated, "One could draw the conclusion that by announcing these fines, the NRC has cleared the way for PECO to receive expedited approval of its Peach Bottom restart plan"(Patriot News, August 17, 1988).

September 2, 1988 - An electrician, working in the low- level radioactive area, " ... fell from scaffolding into a puddle of radioactive water...suffering slight contamination..." (The Patriot News, September 2, 1988).

September 15, 1988 - NRC Chairman Lando Zech told senior management officials of PECO, "I'm not going to accept what you say today and be anywhere near ready to authorize this plant." Zech noted, "Your operators certainly made mistakes, no question about that. Your corporate management problems are just as serious." Zech added, "The fact that we have a situation like this existing at any plant in the country is very serious. We're responsible to the American people. We can't have plants with this much inattentiveness to anything."
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William Russell, regional administrator, told plant officials that unacceptable levels of contamination exist in three pump rooms that are part of Peach Bottom's water cleanup system. He said the radiation in those locations is "some of the worst I've seen" (The Evening News, September 15, 1988, B 3.)

September 23, 1988 - The Board of Directors voted to take no action to prevent the progress of shareholder lawsuits against former chairman and CEO, James L. Everett, III, and former President and CEO, John H. Austin, Jr., "for claims alleging mismanagement which resulted in the shutdown..." of Peach Bottom (Philadelphia Electric Company, Report to the Shareholders, Fourth Quarter, 1988.)

September 26, 1988 - Governor Casey, through the Pennsylvania Department of Environmental Resources (Pa DER), ordered PECO and INPO to release files on recent investigations of the plant. Governor Casey noted, "We made it clear there were certain kinds of information we needed to evaluate our concerns, but after months of being unable to persuade PECO to provide us with that information on its own, we had to go ahead and issue these orders." (Philadelphia Inquirer, September 27, 1988.)

September 27, 1988 - A jury awarded $130,000 to four pipe fitters who claimed they have health problems as a result of being exposed to asbestos at several construction sites including Peach Bottom, Three Mile Island and Glatfelter paper mill.

September 28, 1988 - Senator William Lincoln of Fayette announced that hearings should be required before a Peach Bottom restart.

October 14, 1988 - PE appealed the Pa DER order to give the Casey administration access to internal documents relating to restarting Peach Bottom.

October 19, 1988 - INPO "provided observations on its corporate evaluation conducted in October and on its plant evaluation conducted in September" (Philadelphia Electric Company, Report to the Shareholders, Fourth Quarter, 1988.)
INPO noted "that the operators needed additional simulator training to properly respond to some plant events, that management and shift supervision must take more effective action to correct significant operational and administrative problems, that administrative provisions must be upgraded to better help control room operators readily and accurately determine plant status, and that improvements are needed in communicating and assessing performance standards."

October 21, 1988 - PECO announced a revision in their restart schedule. The projected date for restart was pushed back to the second quarter in 1989.

October 27, 1988 - A recent safety evaluation conducted by the NRC was favorable for restart, according to PECO spokesman Neil McDermott. "What it [the report] is saying is that our plan addresses the problems which led to the shutdown, and that actions laid out in the plan are appropriate to correct those root causes." He added, "Now, of course, the NRC will continue to monitor the effectiveness of the implementation" (The Patriot News, October 22, 1988, B 9.)

November 17, 1988 - The NRC fined PECO $50,000 because security guards were found sleeping on the job, inattentive duty and improperly posted. The NRC also noted that "a key that could have unlocked doors to a security area was issued to a unauthorized employee, couldn't be found and officials didn't do anything about it once they discovered it was missing." William T. Russell, NRC regional administrator, noted, "The improvements made to date were not effective in precluding the occurrence of the violations" (The Patriot News, November 17, 1988, B 2.)

January 1989 - The state of Maryland published a report of radioactive contamination of the Chesapeake Bay due to to emissions from Peach Bottom. (Note: The city of Baltimore gets 250,000 gallons of drinking water per day from the Susquehanna River.)

January 12, 1989 - Admiral James D. Watkins, a member of Philadelphia Electric's Board of Directors, was nominated for the post of Secretary of the Department of Energy.

February 1, 1989 - The NRC staff recommended that nuclear power plants that utilize the Mark 1 containment shell, modify the structure to reduce the risk of failure during a serious accident. PECO said it would make the $2 to $5 million changes only if the Nuclear Regulatory Commission makes the modifications a requirement. This is the second time in two years that the NRC staff has advised the Commission to make changes to the Mark 1 containment structure.

February 8, 1989 - The NRC announced that despite improvements at Peach Bottom, a restart vote will not take place until April, 1989.

February 18, 1989 - The NRC's Integrated Assessment Team's Inspection announced that PECO was close to restarting Peach Bottom.

February 28, 1989 - The Commonwealth of Pennsylvania and Philadelphia Electric concluded an agreement that would give the Commonwealth access to confidential material and allow the state to monitor PECO's operation of Peach Bottom. The agreement was not an endorsement for restarting Peach Bottom.

February 28, 1989 - The Lancaster New Era declared in an editorial on restart that, "While the company claims it sincerely has reformed, we have the overriding impression that reopening the plant, not safety, is the bottom line for the plant operator, Philadelphia Electric Co."

April 21, 1989 - By a 3-0 vote, the NRC approved the restart of Peach Bottom. PECO spokesman Bill Jones calculated that the shutdown cost Philadelphia Electric $300 million. (Patriot News, April 21, 1989, B-3.) "Whistleblower" W. Allan Young, who was fired from Peach Bottom after raising concerns about workers being exposed to high levels of radiation, said in an open letter to the NRC, that the same people who fired him and prevented his rehiring at the plant, are still there. Young told WITF-TV, "They have idiots running that plant."

April 27, 1989 - "An unplanned shutdown was made to repair three malfunctioning intermediate range monitors (IRM) during reactor startup" (SALP 50-277/88-99; 278/88-99.)

April 28, 1989 - Peach Bottom began its ascent towards full power.

May 11, 1989 - "An unplanned shutdown was made to replace a malfunctioning safety relief valve (SRV) which was slow to reclose" (SALP 50-277/88-99; 278/88-99.)

May 14, 1989 - The reactor was taken to subcriticality due to problems with the the electro-hydraulic control (EHC) system (SALP 50-277/88-99; 278/88-99.)

May 19, 1989 - Peach Bottom was shut down due to mechanical problems. Unit 2 "automatically scrammed from 20% power. The cause of the scram was a failed 'three element/single element control switch in the feedwater system" (SALP 50-277/88- 99; 278/88-99.)

May 22, 1989 - "A malfunction in the offgas recombiner system caused the licensee to shutdown the turbine generator and reduce power to 5%" (SALP 50-277/88-99; 278/88-99.)

May 31, 1989 - Peach Bottom was ranked the third worst nuclear power plant in the nation according to a report released by the consumer group Public Citizen. The report, "Nuclear Lemons: An Assessment of America's Worst Commercial Reactors," was based on information obtained from the government and nuclear industry.

June, 1989 - Although the NRC revised its its list of troubled reactors, Philadelphia Electric's Peach Bottom reactors remained on the list.

June 21, 1989 - The NRC released a report on Mark 1 containment buildings entitled "Severe Accident Risks: An Assessment for Five U.S. Nuclear Plants." The NRC's six-member panel were evenly divided as to whether the Mark 1 containment would be breached during a serious accident. Accordingly, "The NRC decided not to order immediate changes in the Mark 1 containment". (The Patriot News, July 21, 1989, B3.) Yet half of the panel stated "with near certainty" the Peach Bottom's containment structure would fail during a core melt accident.

July 21, 1989 - At Peach Bottom 2: "An automatic reactor scram on main steam isolation valve (MSIV) closure occurred when troubleshooting activities in an electro-hydraulic control cabinet caused a false indication of high reactor pressure"(NRC SALP 50-277/89-99; 278/89-99,p.3.)

August, 1989 - PECO "operated Unit 2 at power for about 32 hours with the emergency service water system inoperable." PECO was cited and paid a civil penalty on August 15, 1990.(See February, 1990 for related incident.) (NRC IR 50-277/92-09 and 50-278/92-09.)

August 5, 1989 - PECO reached an agreement with the Public Utility Commission "not to charge customers for $24.3 million in costs incurred by the company when the Peach Bottom nuclear power plant was shut down under a federal order" (Patriot-News, August 4, 1989, B-6.) However, PECO is seeking to "recover" $107 million from its customers through a rate increase.

September, 1989- The NRC released a SALP report indicating weaknesses "...in the performance of and support for some engineering projects, corporate technical assessment activities and management support for health physics training programs and technical facilities" (Annual Report 1989, p.13.)

September 15, 1989 - The Pennsylvania Superior Court reversed a lower court's decision dismissing charges by George Field against the Philadelphia Electric Company. Field, a health- physics technician, alleged that PECO directly released radiation on him to avoid shutting the plant down. The three judge panel concluded:
We can visualize no conduct more outrageous in character, so extreme in degree, that went beyond all possible bounds of decency and to be regarded as atrocious and utterly intolerable in a civilized community, than to vent highly radioactive steam upon an employee. Furthermore, this was an intentional act. They elected to do this to him and then attempted to conceal the resulting situation
The three judge panel remanded the case back to York County Common Pleas Court. Field is seeking $5.2 million in damages.
(The Philadelphia Inquirer, September 15, 1989, 3-B.)

September 19, 1989 - In a report entitled Nuclear Legacy: An Overview of the Places, Problems and Politics of Radioactive Waste in the United States, (Public Citizen September 1989), Peach Bottom was identified as hosting one the largest irradiated fuel pool inventories in the nation. (Peach Bottom-2 was ranked seventh and Peach Bottom-3 was ranked eighth.) The combined volume of irradiated fuel being stored at Peach Bottom is 299.8 cubic meters. The material stored in these pools is classified as high-level reactor waste.

October 5, 1989 - The NRC lifted its shutdown order on Peach Bottom. (The order was enacted on March 31, 1987.) This action allows Unit-3 to restart immediately. (Unit-2 has been operating since April, 1989, while the shutdown order was in effect.) The order also reduces the "strict" monitoring presence of the NRC at Peach Bottom. "The total cost of the shutdown was about $250,000 million, including $168 million for replacement power and a $46 million fine imposed by the state and Public Utility Commission" (Patriot News, October 6, 1989, B-6.)

October 5, 1989 - An automatic scram occurred at Unit 2 due to equipment failure. The plant was at 100% power when "... an outboard MSIV closed during surveillance testing, causing a pressure spike and a high high flux reactor scram" (NRC SALP 50-277/89-99;278/89-89, p.4.)

October 5-10, 1989 - Peach Bottom shut down due to mechanical problems.

November, 1989 - A former PECO employee was convicted by a federal jury of possessing methamphetamine at Peach Bottom in 1985 and 1986. (For more details see: November, 1987; January 8, 1988; February, 1988; and, May 2, 1988.)

November 26, 1989 - An unplanned shutdown at Unit 2 resulted from equipment failure and design weakness.The plant was operating at full power when "an unplanned shutdown was made to repair an unisolable steam leak outside containment emanating from the RCIC injection check valve hinge pin picking" (NRC SALP 50-277/89-99; 278/89-99, p.4.)
Precursor RCIC problems were identifed by the NRC on the follwoing dates: December 10, 1982, March 8 and June 28, 1984, and August 14, 1985.

December 11, 1989 - PECO restarted Unit-3 which was shutdown by the NRC on March 31, 1987. The company has estimated the total cost of the shutdown now exceeds $214 million, including monies spent for replacement power and a rate penalty levied by the Pennsylvania Public Utility Commission (Patriot News, December 13, 1989.)

December 20, 1989 - Unit-2 experienced an "unusual event" and was shutdown. The plant was automatically shutdown from 100% power "after a technician tested a power monitor, according to officials of Philadelphia Electric Co." (Patriot News, December 21, 1989.)

December 27, 1989 - Peach Bottom 2 restarted after shutdown.

January 8, 1990 - The Patriot News reported, "Philadelphia Electric Co. conducted psychological screenings of control-room operators at its Peach Bottom nuclear power plant to determine how many could be retrained after the plant was closed down by the Nuclear Regulatory Commission in 1987" (Patriot News, January 8, 1990, C3.)
The behavior-modification and rehabilitation program, "People: The Foundation of Excellence," was conducted by the psychologists' firm of Rohrer, Hibler & Replogle. Twenty-four out of the 36 control-room operators at the time of the shutdown entered the program. In addition, "10 of the remaining 12 were demoted and reassigned. Of the other two, one retired and one resigned. None of the five shift supervisors were considered for retraining, and were among the group demoted and reassigned" "Patriot,C3)
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In a memo from Julius J. Persensky, a section chief in the NRC's Human Factors Assessment Branch, Mr. Persensky noted the program was of limited value and operators still believe "that their previous behavior was safe." Persensky's memo also noted that Rohrer, Hibler &

Replogle found the operators to be: a depressed, powerless, angry, humiliated and victimized group who didn't think they were doing wrong; practical as opposed to theoretical; open, candid and forthright; sheltered, narrow, parochial and naive; and, loyal to the organization, their profession and the company. According to Rohrer, Hibler & Replogle, up to ten people in may have to retake the program. (Patriot, C3.)

January 27, 1990 - Unit 2 was shutdown again due to equipment failure and design weakness. The plant was shutdown to "repair an unisolable leak outside containment on a "B" reactor feedwater pump discharge flow instrument line" (see November 26, 1989 for a related incident) (NRC SALP 50-277/89-99;278/89-99, p.4.)

January 28, 1990 - Unit 3 was forced into, "A fast power reduction and manual reactor scram were initiated when an electro-hydraulic control system fluid leak developed. The leak was caused by a failed sealing "O" ring (NRC SALP 50-277/89-99; 278/89-99, p.4.) The plant was operating at 100% power.

February, 1990 - The emergency service water system "became inoperable due to improper restoration from maintenance activities." (See August 1989 for related incident.) (NRC IR 50-277/92-09 and 50-278/92-09.)

March 6, 1990 - Unit 3 was shut down due to a "mechanical problem with the system's generator, officials said. Unit 2 had been shut down last week for maintenance" (York Daily Record, March 7, 1990.) However, an inspection report compiled by the NRC stated that "equipment failure complicated by inadequate surveillance procedures" resulted in an automatic scram. The event was caused when "the main turbine tripped at a reactor power of 35% due to A loss of main generating stator cooling" (NRC SALP, 50-279/89-99;278/88-99, p.5.)

March 31, 1990 - In PECO's Report to Shareholders First Quarter 1990, the "Company reported a loss of $84 million, equivalent to 40 cents per share, compared with earnings of $118.9 million or 57 cents per share for the same period a year ago when 2.6 percent fewer shares were outstanding."

April 11, 1990 - Peach Bottom's Unit 2 and Unit 3 reactors were rated *third and fourth worst* in the nation in terms of worker exposures, according to a report released by Public Citizen's energy policy group. The report was based on data obtained from the NRC.

April 21, 1990 - Peach Bottom 2 was "taken off line due to vibrations in the unit's generator exciter" (York Daily Record, May 1, 1990.) Personnel error, procedure weakness and equipment failure contributed to the shutdown.

April 23, 1990 - In a letter to Philadelphia Electric Shareholders, Joseph Paquette, Chairman and CEO, announced, "... the Company's Board of Directors voted to reduce the Company's quarterly dividend from $.55 per share to $.30 per share per share effective with the payment for the second quarter of 1990 to be made June 29, 1990." This action was linked to a rate request regarding the costs of operating and owning Limerick.
- In the Report to Shareholders for the *Third Quarter 1990*, Philadelphia Electric reported reaching a settlement "in the shareholders' derivative suit brought by certain shareholders against the Company's former Chairman and former President in connection with the events leading to the shutdown....Under the terms of a settlement agreement, two of the Company's director and Officer liability insurance carriers paid approximately $34.5 million. The settlement became final on October 30, 1990. The plaintiffs' recovery, less $6.5 million for their attorneys' fees and expenses were paid to the Company on November 1."
However, In PECO's annual statement, the company admitted, "The penalties associated with the [Peach Bottom's] shutdown for 1989 amounted to 23 cents per share, compared to 25 cents per share for 1988" (Annual Report 1989, p.14).
In addition, "The Company did not request recovery of any Peach Bottom replacement power costs incurred solely as a result of the NRC's shutdown order. In 1989, replacement power costs attributable to the shutdown order were approximately $57 million , representing a reduction in common stock earnings of 17 cents per share" (Annual Report, p.21.)

May 11, 1990 - "...instrument and controls technicians replacing a voltmeter on the '3B' battery charger caused a DC electrical system voltage transient" (NRC IR 50-277/92-09 and 50-278/92-09.)

June 15, 1990 - The Public Utility Commission (PUC) ruled that Philadelphia Electric had to refund to its customers $15 million. "The PUC ruled that PECO kept sloppy records, did not use enough competitive bidding and did not bid projects frequently enough" (Patriot News, June 15, 1990.)

June 26, 1990 - The Pennsylvania Public Utilities Commission (PUC) released its twelfth annual report on utility consumer complaints to the PUC's Bureau of Consumer Services. The report noted that PECO was one of the companies whose overall performance "was worse than that of other companies" and "would benefit both from a critical review of their own operations and from attempting to emulate the operations of the companies which performed best."

July 18, 1990 - The NRC fined PECO $75,000 for violations of technical specifications involving the "plant's emergency service water system, a support system designed to cool safety equipment, other than the reactors, at Peach Bottom's Units 2 and 3" (The Patriot, July 18, 1990, B 5.)

July 28, 1990 - Philadelphia Electric declared an unusual event from "5:38 am to 6 am because of a momentary increase in radiation levels in an internal gas-filtering system" (Patriot News, July 28, 1990, A 3.) Radioactive gas was released into the environment for ten minutes.

August 15, 1990 - PECO paid a civil fine to the NRC for an August, 1989 incident involving the emergency service water system. (Also see February, 1990.)

August 16, 1990 - In NRC inspections from July 1,1989 to May 31, 1990, Peach Bottom 2 "experienced six unplanned shutdowns because of personnel errors or equipment failures, while the Unit 3 reactor had two shutdowns " (Philadelphia Inquirer August 16, 1990, 17 D).

September 11, 1990 - PECO "discovered that indications derived from Unit 3 reactor water level transmitters...were abnormally high when compared to actual reactor water level" (NRC IR 50-277/92-13 and 50-278/92-13.) (See March 26 and 27, 1992 and July 26, 1992 for related incidents.)

December 1, 1990 - In Philadelphia Electric's Report to Shareholders Third Quarter 1990,PECO announced: "For the three months ended September 30, 1990, the Company reported a loss of $8 million, or 4 cents per share ....Earnings for the twelve months ended September 30, 1990 were 53 cents per share, $1.68 under the earnings of the previous twelve month period."

February 1, 1991 - In PECO's Annual Report 1990, the company noted that earnings per share plummeted by a $1.78. Operating and maintenance costs rose by $406 million or 38%.

February 11, 1991 - "A contractor working inside the dormant Unit 2...took an 8-foot fall and was flown to York Hospital with slight contamination to his forehead." Neil McDermott, a company spokesman for PECO, said: "They resolved it by, (the contamination), well, soap and water" (Patriot, February 11, 1991.)

February 12, 1991 - A, "Unit 2 primary containment isolation system (PCIS) and standby gas treatment system (SGTS) initiated (9:10 am) due to an electrical ground. "The event was not detected by the plant operators until about 10:00 am, because related annunciators had been removed from service for outage work" (NRC inspection reports 50-277/91-08; 50-278/91-08, p.2.)

February 20, 1991 - At about 1:10 pm, a full Unit 2 reactor scram occurred due to inadequate blocking. "The unit was in refueling at the time with all control rods inserted" (See related incident on February 21, 1991)(NRC inspections 50-277/91-08;50-278/91-08, p.2.)

February 21, 1991 - Inadequate blocking caused a loss of shutdown cooling. The "isolation occurred when an auxiliary operator (AO) inadvertently grounded a lead in the control room panel while applying a blocking permit" (See related incident on February 20, 1991) (NRC inspections 50-277/91-08;50-278/91-08, p.3.)

February 21, 1991 - At 10:00 pm at Unit 2, fuel bundles were misplaced during a core reload. "An investigation revealed that the bundle had been erroneously loaded ...at 1:47 of the same day" (See related incidents on February 21-22, 1991)(NRC inspections 50-277/91-08; 50-278/91-08, p.4.)

February 22, 1991 - A fuel bundle at Unit 2, at a separate location from the previous day's error, was "incorrectly loaded" at 1:15 pm. The errors was not found until 6:00 am on February 24, 1991. Contributing to this error Poor CCTAS legibility" and "less than adequate communications."
On the same day a third and fourth error occurred!
"The third error was identified at about 3:00 pm....Fuel movement was suspended and the core and spent fuel pool (SFP) were inspected, leading to the discovery of fourth error" (See February 21 1991 for a related incident) (NRC inspections 50-277/-91-08; 50-278/91-08.)

February 23, 1991 - The refueling moderator temperature was exceeded. "The lower moderator's temperature results in the addition of positive reactivity, and a decrease in shutdown margin....Fuel reload was halted..." (NRC inspection reports 50-277/91-08;50-278/91-08, p.6.)

February 25, 1991 - Unit was at 100% power when "a high pressure coolant injection (HPCI) was declared inoperable when the mechanical overspeed trip (MOTD) did not operate as designed during performance of a routine surveillance test" (NRC inspection reports 50-

279/1-08/50-278/91-08, p.3.) (For related events see: May 18 and 21, 1991; July 15-19, 1991; August 25, 1991; and, October 16, 1991.)

March 21, 1991 - PECO "found four normally locked open unit 2 valves unlocked. Two of these valves were also closed" (NRC inspection reports 50-277/91-13;50-278/91-13, p.11.)

April 1-5, 1991 - The NRC issued a Notice of Violation. "The violation is of concern because of the possible incompatibility of the insulation with materials it is in contact with and the fact that it may compromise fire loadings and propagation potentials" (NRC inspections 50-277/91-14 and 50-278/91-14.)

April 7, 1991 - The Chief Rector Operator discovered that the Technical Specifications surveillance requirement to log Unit 2's reactor vessel heat up rate had not been performed . (NRC inspections 50-277/91-13;50-278/91-13, pp. 2-3.)

April 10-11, 1991 - The Unit 3 high pressure coolant injection system failed several times.

April 15, 1991 - During maintenance testing it was discovered that "valves were reinstalled in the wrong direction following the current valve refurbishment" (NRC inspection reports 50-277/91-13/50-278/91-13, p. 5.)

April 22, 1991 - "...a fault developed in one of the conductors connecting the secondary side of the # 2 Emergency Auxiliary (2EA) transfer to the safety and non-safety related 4 KV busses" (NRC inspection reports 50-277/91-13;50-278/91-13, p.7.)

April 23, 1991 - At Unit 2 "reactor power was decreased, the mode switch was placed in startup and power was held at 5% to replace cable on an emergency transformer when its insulation was found to be shorted" (NRC inspection reports 50-277/91-16 and 50-278/91-16, Details.)

April 25, 1991 - Peach Bottom 2 was rated the third worst nuclear reactor in the county. Peach Bottom 2 and 3 were tired for seventh worst rate of worker exposure to radiation. (Public Citizen, Nuclear Lemons: An Assessment of America's Worst Commercial Nuclear Power Plants.)

May 2, 1991 - "Due to further degradation of emergency transformer cable insulation the unit (2) was shut down on may 2 to replace the cables" (NRC inspection reports 50-277/91-16 and 50-278/91-16, Details.)(See July 4, 1992 for a related incident.)

May 9, 1991 - The Unit 3 reactor experienced "an unexpected isolation of the reactor water cleanup (RWCU) system occurred when technicians placed a jumper in an incorrect location" (NRC inspections 50-277/91-16 and 50-278/91-16, p.2.)

May 13-20, 1991 - An NRC inspection noted that: "During the 1991 Unit 2 refueling outage, leaks in the Unit 3 Offgas System allowed noble gas to be released to many areas of the plant"(NRC inspection reports 50-277/91-17 and 50-278/91-17, p.3.)

May 15, 1991 - During the performance of a surveillance test at Unit 2, "system engineers incorrectly removed fuse DD-29 from panel 20C15 instead of the specified fuse DD-28. Pulling the fuse removed power from the primary containment isolation system (PCIS) group III inboard isolation logic, causing the associated components to isolate" (NRC inspection reports 50-277/91-16 and 50-278/91-16, p.3.)

May 18, 1991 - The Unit 2 high pressure coolant injection (HPCI) system was made inoperable during fire protection system surveillance testing. (NRC inspections 50-277/91-16 and 50-278/91-16.) (For related event see: February 25, 1991; May 21, 1991; June 19, 1991; July 15-19; August 27, 1991; and, October 16, 1991.)

May 20, 1991 - At Unit 3, "the residual heat removal (RHR) pump automatically started when technicians incorrectly removed a switch from the 'test position'" (NRC inspection reports 50-277/91-16 and 50-278/91-16, p.4.)

May 21, 1991 - During a routine surveillance procedure at Unit 2, "an unexpected isolation of the HPCI system steam line" occurred (NRC inspection reports 50-277/91-16 and 50-278/91-16, p.4.) (For related events see: February 25, 1991; May 18, 1991; June 19, 1991; July 15-19; August 25, 1991; and, October 16, 1991.)

May 21, 1991 - Both units were affected by the inoperability of the emergency diesel generator due to unqualified relays. (NRC inspection reports 50-277/91-16 and 50-278/91-16, pp.5-6.)

May 23, 1991 - Units 2 and 3 were shutdown "due to a belief that the 4 station Emergency Diesel generators (EDG's) could potentially be rendered inoperable during design basic events" (Licensee Event Report 50-277 and 50-278.)

May 29, 1991 - Both standby liquid control (SLC) pumps at Unit 3 were rendered inoperable due to high tank temperatures. (NRC inspection reports 50-277/91-16 and 50-278/91-16.)

June 7, 1991 - Unit 2 was shutdown (tripped) due to inadequate recirculation pump seal cooling.((NRC inspections 50-277/91-16 and 50-278/91-16.)

June 15, 1991 - An NRC inspector "found a security guard asleep on the Unit 2 refuel floor...The guard had been assigned to watch a cask which had not been opened and searched" (Inspection reports 50-277/91-20 and 50-278/91-20.)

June 19, 1991 - A Notice of Violation was issued for an incident which involved the high pressure coolant injection system on May 21, 1991.(See February 25, 1991; May 18 and 21, 1991; and, July 15-19, 1991 for related incidents.)

June 24, 1991 - Unit 2 pressure transmitters were identified as not being seismically supported."The support for the PT's was mounted on non seismic floor grating and only one of four anchor bolts was installed" (Inspection reports 50-277/91-20 and 50-278/91-20.)

June 24-28, 1991 - A Notice of Violation was issued for the following: "Two instances were identified in which corrective actions taken by your staff had not adequately resolved deficiencies related to quality classification of safety-related equipment (Q-List), and control of measuring and test equipment" (NRC inspection 50-277/91-20 and 50-278/91-20.)

June 24-28, 1991 - An NRC radiological safety inspection observed, "Audit findings indicated that, at times, management had provided poor oversight of program activities. For example, individuals who failed to perform radiologically sound work were not always held accountable for their work. Examples of poor quality were observed for individuals both internal and external to the HP organization" (NRC inspections 50-277/91-22 and 50-278/91-22)

June 27, 1991 - An unplanned manual scram occurred at Unit 2 due to low condenser vacuum.(NRC inspection reports 50-277/91-20 and 50-278/91-20.)

July 7, 1991 - Unit 3 was scrammed following a trip of the main generator output breakers. (NRC inspections 50-277/91-20 and 50-278/91-20.)

July 8-12, 1991 - The NRC staff "...identified several instances of failure to take effective corrective action in response to previously identified problems in the surveillance testing area. We are concerned with this matter because of the time which has elapsed since these problems were first identified. Management has not developed detailed plans or goals to improve performance in this area" (NRC inspections 50-277/91-23 and 50-278/91-23.)

July 10, 1991 - At Unit 3, "licensee technicians inadvertently caused a trip of the "B" reactor protection system (RPS) motor generator (MG) set." The secondary containment was also isolated during troubleshooting. (NRC inspections 50-277/91-21 and 50-278/91-21.)

July 16-17, 1991 - The licensee determined that there was low emergency water flow to Unit 2's Emergency Diesel Generators and residual heat removal pumps. "As a result, the Unit 2 RCIC and 'B' loop of low pressure coolant injection (LPCI) were declared inoperable on July 16 and 17" (NRC inspections 50-277/91-21 and 50-278/91-21.)

July 15-19, 1991 - During an inspection the NRC observed: "...one of your activities related to the operability of the high pressure coolant injection (HPCI) system appears to be in violation of NRC requirements..." (NRC inspections 50-277/91-24 and 50-278/91-24.) (For related events see: February 25, 1991; May 18 and May 21, 1991; June 19,1991; August 25, 1991, and, October 16, 1991.)

July 18, 1991 - The Unit 2 high pressure coolant injection system isolated during surveillance testing. (NRC inspections 50-277/91-21 and 50-278/91-21.)

July 24, 1991 - An initiation of a Unit 3 plant shutdown occurred due to an inoperable DG Auto-start logic. (NRC inspections 50-277/91-21 and 50-278/91-21.)

July 27, 1991 - There was a partial containment isolation at Unit 3 following the failure of a 500 KV disconnect switch.

July 24, 1991 - A letter from the Assistant Associate Director of FEMA noted: "Twenty-two Areas Requiring Corrective Action were identified during the [emergency preparedness practice on February 7, 1990] exercise. FEMA's Region III staff will monitor the status of the corrective actions" (Letter to the NRC from Dennis H. Kwitatkoski.)

July 30- August 1,8 and 22, 1991 - The NRC conducted safety inspections of emergency preparedness exercises and found: "While no violations were noted during the inspection, one exercise weakness was identified. This weakness concerned a significant breakdown in the communication, distribution, and tracking of scenario data" (NRC inspections 50-277/91-25 and 50-278/91-25.)

July 31, 1991 - A Notice of Violation was issued for an "event at the Peach Bottom facility during which you [PECO] overheated the Unit 3 standby liquid control (SLC) solution storage tank" (See May 29, 1991 for more details) (NRC inspections 50-277/91-16 and 50-278/91-16.)

August 5, 1991 - Unit 2 scrammed at 98% power. "The main turbine tripped due to high level in the 'D' moisture separator drain tank (MSDT)" (NRC inspections 50-277/91-27 and 50-278/91-27.)

August 12, 1991 - The NRC revealed that they did not have current copies of Peach Bottom's Emergency Operating Procedures.

August 25, 1991 - Unit 3 was shutdown due to inoperable room coolers. PECO "found that both the high pressure coolant injection (HPCI) and the reactor core isolation cooling (RCIC) system pump component coolers were inoperable" (NRC inspections 50-277/91-27 and 50-278/91-27.) (For related incidents see: February 25, 1991; May 18 and 21, 1991; July 15-19, 1991; and, October 16, 1991.)

August 27, 1991 - Both units were "shutdown following discovery that two of the four emergency diesel generators (EDG) were inoperable" (NRC inspections 50-277/91-27 and 50-278/91-27.)

September 8, 1991 - Philadelphia Electric "discovered that the "A" CAD sample line from the torus was plugged" (NRC inspection 50-277/91-27 and 50-278/91-27.)

September 12, 1991 - An unusual event was declared when jet pump components dropped into the spent fuel pool" (NRC inspections 50-277/91-27 and 50-278/91-27.)

September 17, 18 and 24, 1991 - The control room emergency ventilation system isolated and transferred to the emergency ventilation mode" (Another occurrence was reported on October 25, 1991.) (NRC inspections 50-277/91-27 and 50-278/91-27.)

September 19-20 and 23-24, 1991 - A Notice of Violation was issued by the NRC. The staff reported: "Of concern to us associated with the work on RWCU Pump 3B was the failure of your staff to perform an assessment of the radiological hazards associated with pump components and subsequent failure to establish appropriate radiological controls for the work. Surveys for beta radiation hazard of the pump impeller and internal components were not made prior to allowing work to commence on them. After the work was completed contact beta radiation dose rates were determined to be as high as 1,100 Rads per hour. While performing the work without accurate knowledge of the beta radiation dose rate did not lead to an overexposure, it may have resulted in unnecessary exposure" (NRC inspections 50-277/91-28 and 50-278/91-28.)

September 24, 1991 - PECO determined that there was "induced fuel failure" at Unit 3. "The licensee visually inspected the six bundles and identified that one of the bundles had experienced failure caused by a malfunctioning defect, while the other five bundles had experienced debris induced failure. The debris appeared to be small metal chips" (NRC inspections 50-277/91-33 and 50-278/91-33.)

September 27 through November 4, 1991 - During this inspection period the NRC found "certain" of PECO's activities to be in "violation." A Notice of Violation was issued. "Inadequate initial and independent verification of a valve position resulted in an emergency core cooling pump being inoperable for about seven days.The consistency and quality of worker and independent verification of safety-related operations, maintenance and test activities is a recurring weakness" (NRC inspections 50-277/91-30 and 50-278/91-30.)

October, 1991 - Employees using the wrong shutdown manual caused an overheating of the plant's boron injection water. Larry Doerflein of the NRC commented: "By and large, there has been little overall progress. We're still seeing the same problems we saw a year ago" ("Atoms & Waste," October, 1991.)

October 2, 1991 - The NRC issued a violation "associated with inadequate radiation surveys during work on highly radioactive components" (NRC IR50-277/92-80 50-278/92-80.)

October 16, 1991 - Unit 2 was shut down at 73% power due to the inoperability of the high pressure coolant injection. A steam isolation valve packing leak had been detected.(NRC inspections 50-277/91-30 and 50-278/91-30.) (For related incidents see: February 24, 1991; May 18 and 21, 1991; July 15-19, 1991; and, August 25, 1991.)

October 21-25, 1991 - "One non-cited violation was noted involving radioactive material receipt practices (NRC inspections 50-277/91-32 and 50-278/91-32.)

October 22, 1991 - A fire in the Unit 3 condenser bay occurred from 10:23 p.m. to 10:37 p.m. (NRC inspections 50-277/91-30 and 50-278/91-30.)

October 25, 1991 - "The main control ventilation system automatically isolated and transferred the emergency ventilation mode..." (This type of actuation also occurred on September 17, 18 and 24, 1991.) (NRC inspections 50-277/91-30 and 50-278/91-30.)

October 26, 1991 - An unusual event was declared when a "potentially contaminated individual" was transported offsite.(NRC inspections 50-277/91-30 and 50-278/91-30.) (See December 8, 1991 for related incident.)

October 27, 1991 - Nuclear Maintenance Division "found the fuel bundle at spent fuel pool location Z-31 to be oriented improperly" (50-277/91-30 and 50-278/91-30.)

October 28, 1991 - "Smoke was detected coming from the Unit 2 "B" Low Pressure Coolant Injection (LPCI) swing bus. Further examination revealed that the power monitoring relay for the bus had burned up" (NRC inspections 50-277/91-30 and 50-278/91-30.)

October 28, 1991 - The "B" auxiliary boiler was contaminated with radioactive iodine-131. The boiler was isolated and radioactive liquid was drained to the radwaste system. (See December 23, 1991 and February 24, 1992 for related incidents.)

November 4, 1991 - "The Unit 2 'B' reactor protection system (RPS) motor generator (MG) set unexpectedly tripped" (NRC inspections 50-277/91-30 and 50-278/91-30.)

November 8, 1991 - PECO "determined that the automatic depressurization system (ADS) had been inoperable from shortly after the plant startup in December 1989 to shutdown for the refueling outage on September 14, 1991. The licensee concluded that the environmental qualification (EQ) of the solenoid operated valves (SOV), electrical cables and splices, to the five ADS safety related valves (SRV) had expired shortly after startup. The thermal insulation over all 11 SRVs, including the 5 SRVs dedicated to ADS, had been installed backwards during the last refueling outage" (NRC inspections 50-277/91-33 and 50-278/91-33.)

December 1, 1991 -In PECO's "Report to Shareholders, Third Quarter, 1991,"it was revealed that a management audit was conducted from July, 1989 to May, 1990. The audit was completed by Ernst & Young and released in August, 1991. Philadelphia Electric admitted that the audit "details a significant number of opportunities for the Company to improve in almost every aspect of operations, and we have submitted a detailed implementation plan to the PUC addressing each of the recommendations for improvement."

December 5, 1991 - Unit 2 was forced to shutdown due to excessive leakage past the residual heat removal system injection check valve. (NRC inspections 50-277/91-33 and 50-278/91-33.)

December 5, 1991 - A reactor core isolation occurred at Unit 2. (NRC inspections 50-277/91-33 and 50-278/91-33.)

December 8, 1991 - An unusual event was declared when a potentially contaminated individual was transported off site. (NRC inspections 50-277/91-33 and 50-278/91-33.) (See October 26, 1991 for related incident.)

December 16, 1991 - At Unit 3, "an unexpected primary containment isolation occurred..." during instrument line-up (NRC inspections 50-277/91-43 and 50-278/91-34.)(See March 10, 1992 for related incident.)

December 18, 1991 - A shutdown cooling isolation occurred at Unit 3 "when a PCIS logic fuse blew" (NRC inspections 50-277/91-43 and 50-278/91-34.) (See January 4, 1992 for related incident.)

December 23, 1991 - Low-level iodine-131 contamination was reported at the "B" and "C" auxiliary boilers. (See October 28, 1991 and February 24, 1992 for related incidents.)

December 24, 1991 - In a letter to Mr. D.M.Smith, Senior Vice President-Nuclear, the NRC identified two problems at Peach Bottom. "The first problem concerns the degradation, and potential extended inoperability, of the Unit 3 automatic depressurization system due to the incorrect installation of the valve thermal insulation. In addition, your immediate corrective actions following discovery of this problem were not completely effective. A similar problem on one Unit 2 valve was not identified and corrected until raised by the inspector. Based on our review of the issues, two apparent violations of NRC requirements were identified and are being considered for escalated enforcement action..." (Charles W. Hehl, Director, Division of Reactor Projects.)

January 4, 1992 - Due to valve fuse failure, PECO "determined that containment integrity could not be assured for the reactor core isolation cooling suppression pool suction line" (NRC inspections 50-277/91-34 and 50-278/91-34.) (See December 18, 1991 for related incident.)

January 17, 1992 - High oxygen concentration levels were recorded in the Unit 3 control room.

February 24, 1992 - The NRC reviewed PECO's efforts to desludge the flood drain waste storage tank and found several problems: "...The radiation protection technician who wrote the permit was unaware that personnel would be walking in radioactive sludge measuring up to 350 millirem per hour (mr/hr) on contact...The radiation protection supervisor who signed the RWP was not aware that workers would be working in sludge...the planning process did not evaluate the collective radiation exposure that would result from desludging all tanks over the life of the PM process... The work activity was not reviewed by the ALARA group, which precluded in-depth evaluation of all exposure reduction methods, including the use of state-of-the-art cleaning techniques or design changes to tanks to provide for ease of future cleaning that would reduce aggregate exposure...The filter clogged and resulted in additional personnel exposure...the licensee contacted no other stations to identify state-of-the-art methods to perform tank desludging" (NRC IR 50-277/92-80 and 50-278/92-80.)

February 24, 1992 - Low-levels of iodine-131 contamination in the "A" auxiliary boiler were reported. (See October 28 and December 23, 1991 for related events.)

February 24 through March 13, 1992 - The NRC's Integrated Performance Assessment Team (IPAT) issued its findings and "concluded that several weaknesses merit near-term corrective actions to reduce the potential for future safety problems...the team observed weaknesses in licensee evaluation of degraded or inoperable control room instrumentation and permanently installed plant instrumentation. Weaknesses were also identified in the lack of interim corrective actions for self-assessment findings and in the control of documents related to modifications and temporary plant and procedure changes" (NRC Region I IPAT IR 50-277/92-80 and 50-278/92-80.)

February 25, 1992 - Philadelphia Electric agreed to pay $285,000 in fines for the improper insulation of safety system relief valves at Unit 3. Company spokesman Neil McDermott admitted there is "absolutely no question and we readily admit that the insulation was improperly installed" (Patriot News, February 25, 1992.)

March 6, 1992 - The NRC observed: "Several weaknesses were noted in the training program during the conduct of the examinations. Differences between Peach Bottom and Limerick had a negative impact on some LSRO lesson plans in that the lesson plans did not track actual plant practice. LSRO responsibilities were not well defined at Limerick and differ from those at Peach Bottom. Training was not always given as described in the task to training matrix or the qualification manual. In general, the candidates' knowledge of the site and plant at which they were not normally stationed was weak." (Lee H. Bettenhausen, Chief, Operations Branch, Division of Reactor Safety.)

March 10, 1992 - PECO "concluded" that Units 2 & 3 had deficiencies in their primary containment isolation systems.(NRC inspections 50-277/92-07 and 50-278/92-07.) (See December 16, 1991 for related incident.)

March 10, 1992 - The NRC's Integrated Performance Assessment Team (IPAT) observed, "an operator exit the fourth floor administration building radiological control point...without properly surveying personal articles being removed from the radiological control area" (NRC Region I IPAT 50-277/92-80 and 50-278/92-80.)

March 13, 1992 - Philadelphia Electric "discovered" Unit 2 residual heat removal equipment valves were not installed."With the check valves on the discharge of the sump pumps for the 'B' and 'D' RHR rooms not installed, this design basis can not be met. Specifically, during a loss of coolant accident, concurrent with a loss of off-site power, the reactor building sump pumps would not be available due to the loss of off-site power" (NRC inspections 50-277/92-07 and 50-278/92-07.)

March 16, 1992 - Due to a turbine exhaust drain line valve failure, the Unit 2 high pressure coolant injection system was rendered inoperable.(NRC inspections 50-277/92-07 and 50-278/92-07.) (See March 23, 1992 for related incident.)

March 23, 1992 - PECO "declared the HPCI system inoperable when the turbine overspeed trip device did not reset during testing" (NRC inspections 50-277/92-07 and 50-278/92-07.) (See March 16, 1992 for related incident.)

March 26, 1992 - PECO "declared all Unit 2 reactor water level instrumentation associated with the 2B reactor water level reference leg condensing chamber inoperable" (NRC IR 50-277/92-13 and 50-278/92-13.) (See September 11, 1990, March 27, 1992 and July 26, 1992 for related incidents.)

March 27, 1992 - Unit 2 was shutdown due to inoperable reactor level instrumentation. (See September 11, 1990, March 26, 1992 and July 26, 1992 for related incidents.)

April 2, 1992 - A settlement was announced on the two lawsuits brought against PECO by Peach Bottom's co-owners: Public Service Electric and Gas Company, Delmarva Power and Light Company and Atlantic City Electric Company. The suits were related to the NRC shutdown of Peach Bottom on March 31, 1987."As part of the settlement, Philadelphia Electric will pay $130,985,000 on October 1, 1992 to resolve all pending litigation." (Joseph Paquette, April 8, 1982.) (See July 27, 1988 for background material.)

April 7, 1992 - PECO began a planned shutdown for Unit 2 from about 100% power. "The shutdown was required because a one inch vent line failed at a welded connection on the condensate supply herder to the offgas recombiner condenser...A reactor scram and primary containment isolation system (PCIS) group II and III occurred" (NRC IR 50-277/92-09 and 50-278/92-09.)

April 17, 1992 - The NRC issued a Notice of Violation for the following infractions: "Contrary to the above requirements, the ODCM [Offsite Dose Calculation Manual] specified composite water sampler at the intake had been inoperable during the period August 30, 1991 to March 19, 1992, and the specified composite water sampler at the discharge had been inoperable since August 8, 1991 and remains inoperable at the time this inspection [was] conducted March 23-27, 1992. The licensee's efforts to complete corrective action prior to the next sampling period were ineffective" (NRC inspections 50-277/92-08 and 50-278/92-08.)

April 29, 1992 - A Health Physics technician was contaminated in the de-watering facility when "contamination controls were compromised. According to the licensee's investigation, a defective latch and hinge on the fill-head access door allowed contamination to escape from the liner to the room during processing. Contamination levels on near-by radwaste equipment were as high as 200 mrad/hour. The general area surfaces in the truck bay were contaminated up to 30,000 dpm/100cm (2)" (NRC IR 50-277/92-12 and 50-278/92-12.)

May 4, 1992 - Philadelphia Electric "initiated a planned shutdown [at Unit 3] in order to repair a large steam leak through a manway on the 'F' moisture separator tank" (NRC inspections 50-277/92-11 and 50-278/92-11.)

May 12, 1992 - Unit 3 recirculation pump trip occurred at 80% power.(See June 27, July 23, July 26 and July 27, 1992 for related incidents.)

May 15, 1992 - PECO initiated a shutdown of Unit 2 "due to inoperability of the high pressure coolant injection and the reactor core isolation cooling systems" (NRC inspections 50-277/92-11 and 50-278/92-11.) (See June 25, 1992 for related incident.)

May 20, 1992 - Unit 2 experienced a reactor scram and turbine trip due to a malfunctioning combined intermediate valve.

May 22, 1992 - The motor for the Unit 3 residual heat removal pump failed and was declared inoperable.

June 1, 1992 - "Common stock earnings for the first quarter of 1992 were $0.33 per share, $0.25 lower than the $0.58 per share earnings for the corresponding period last year. The reduction in earnings was primarily the result of the previously reported settlement of litigation by the co-owners of Peach Bottom Atomic Power Station which reduced first quarter earnings by approximately $0.27 per share" (J.F. Paquette, Jr., Chairman of the Board and Chief Executive Officer, Report to Shareholders First Quarter, 1992).

June 25, 1992 - The Unit 3 high pressure coolant injection system was declared inoperable "due to excessive water buildup in the turbine casing" (NRC IR 50-277/92-13 and 50-278/92-13.) (See May 15, 1992 for a related incident.)

June 27, 1992 - The 'A' recirculation pump tripped at Unit 2.(See May 12, July 23, July 26 and 27, 1992 for related incidents.)

July 4, 1992 - An Alert was declared at Peach Bottom due an explosion at the #1 transformer station. Units 2 and 3 were operating at at, or around, 95 % power. As a result of the explosion, Unit 3 scrammed and there were several emergency safeguard actuations.(See May 2, 1991 for a related incident.)

July 14, 1992 - "Unit 3 was manually scrammed from 63% power due to a decreasing main condenser vacuum" (NRC IR50-277/92-13 and 50-278/92-13.)

July 17, 1992 - Unit 2 experienced a turbine trip and reactor scram at 95% power during a severe lightning storm.

July 23, 1992 - The Unit 3 recirculation pump tripped at 95% power.(See May 12, June 27, July 26 and July 27, 1992 for related incidents.)

July 25, 1992 - "Unit 2 was shutdown due to a safety relief valve bellows rupture alarm" (NRC IR 50-277/92-13 and 50-278/92-13.)

July 26, 1992 - The 'A' recirculation pump tripped at Unit 2. (See May 12, June 27, July 23 and July 27, 1992 for related incidents.)

July 26, 1992 - A safety device used at Peach Bottom and 35 other American nuclear reactors may be defective according to the NRC. "Engineers are concerned that in a serious accident involving the rapid loss of coolant and pressure from the reactor, the device would give a false reading, indicating the reactor core was still covered with water when it actually was not and therefore in danger of melting down" (Sunday Patriot News, July 26, 1992 A3.) (See September 11, 1990 and March 26 and 27, 1992 for related incidents.)
Peach Bottom has had a history of problems in this area.
" In August 1990, the licensee identified that the Unit 2 level instrumentation served by the 2B condensing chamber and reference leg was indicating values about 11 inches higher than similar instruments served by the 2A condensing chamber...They [PECO] concluded that the actuation set points for several safety systems would be exceeded during transients or accidents, declared the instruments inoperable and completed a plant shutdown. Following the 1990 event, the licensee revised the channel check procedures to provide better monitoring and evaluation of the instruments...A second level offset event, again
Continued on the next page...
involving the Unit 2B condensing chamber, occurred in March 1992. The improved surveillance procedures helped the licensee identify the offset before it had exceeded 3 inches. In response, the licensee established a 4 1/2 inch offset operability limit, and closely monitored the instrumentation..." (NRC IR 50-277/92-16 and 50-278/92-16.) (For related incidents see September 11, 1990 and March 26-27, 1992.)

July 27, 1992 - The 'A' recirculation pump tripped at Unit 2. (See May 12, June 27, July 23 and July 26, 1992 for related incidents.)

July 27, 1992 - Peach Bottom and 86 other suspected nuclear reactors "depend on a defective and dangerous fire-barrier system to protect electrical cables used for a safe shutdown during a fire/accident." (Nuclear Information and Resource Service (NIRS), July 27, 1992.) The company who produces the Thermo-Lag 330 system is Thermal Science, Inc. (TSI), St. Louis, Missouri. Among the problems with Thermo-Lag are: combustibility, toxicity, seismic qualification, vulnerability to water, incomplete installation and ampacity calculation errors.
In an IR issued on September 10, 1992, PECO requested a temporary waiver of technical specification compliance for certain fire barriers. The NRC observed: "...the licensee could not post the required fire watch for

residual heat removal system cables running through the Unit 3 offgas pipe tunnel because it is a high radiation area".
(NRC IR 5 277/92\16 and 50-278/92-16.)

August 6, 1992 - The NRC issued a violation "for operation of the reactor cleanup system in a mode not established in approved operating procedures, is of concern because it represents a weakness in your control of operating activities" (NRC IR 50-277/92-13 and 50-278/92-13.)

August 10, 1992 - PECO entered a seven day maintenance outage on the E-4 emergency diesel generator.

August 17, 1992 - A generator lock-out and reactor scram occurred at Unit 2 due to improper blocking. PECO "determined that the generator lock-out occurred because the permit being applied in the South Substation was incorrect" (NRC IR 50-277/92-16 and 50-278/92-16.)

August 20, 1992 - The Unit 3 Emergency Core Coolant System power supply failed. The root cause was a failed topaz inverter.

September 14, 1992 - A licensed operator tested positive for marijuana use.

October 6, 1992 - During an NRC inspection relating to plant security, one unresolved Fitness-for-Duty(FFD) item was identified. The NRC also cautioned that "... additional attention is warranted on the effectiveness of routine security patrols since we identified certain deficiencies during this inspection that should have been identified by your officers on patrol" (NRC IR 50-277/92-20 and 50-278/92-20.)

October 15, 1992 - Unit 3 scrammed and the high pressure coolant injection (HPCI) system initiated: "... Unit 3 experienced a primary containment isolation system (PCIS) group I isolation on main steam line (MSL) low pressure. This resulted in closure of the MSIVs and a reactor scram. During the post-scram pressure and level transient, vessel water decreased to the ECCS Lo Level initiation setpoint. The high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems initiated and injected into the reactor vessel. The alternate rod insertion and reactor recirculation point trip logic also actuated. Three main steam safety relief valves (SRV) opened automatically for a short period to control pressure, and later re-closed. The licensee declared an unusual event (UE) at about 9:25 p.m. due to the initiation and ejection of an ECCS system in response to a valid signal...At about 11;16 p.m., while proceeding with the plant cooldown, reactor vessel level increased above the normal operating band and caused a HPCI and RCIC high reactor

vessel water level turbine trip. Due to the temporary loss of HPCI as a means of pressure control, reactor pressure increased to the high pressure scram setpoint. the operators manually operated an SRV to reduce pressure, and restarted HPCI and RCIC. the licensee also reported this second scram signal to the NRC via the ENS. All systems responded as expected following the PCIS group I isolation and reactor scram, and the subsequent high reactor pressure scram" (NRC IR 50-277/92-27 and 50-278/92-27.)
PECO management decided to shut the plant for five days.
After reviewing the events the NRC issued a Notice of Violation and criticized, "The control room staff did not effectively monitor developing reactor coolant stratification following the Unit 3 automatic scram, and certain Technical Specification reactor pressure/temperature limits were exceeded. Adequate controls were not in place to ensure that the transient was appropriately evaluated before plant restart. Also, operators did not record required pressure data used to evaluate compliance with pressure/temperature limits following a Unit 2 shut-down." (E. Wenzinger, Chief, Projects Branch 2, Division of Reactor Projects, November 16, 1992.)

October 16, 1992 - The NRC found one potential problem with senior reactor operators (SRO) examinations:"Since SRO Upgrades are currently licensed individuals at your facility, we are concerned that your training program may not be emphasizing a high level of performance among reactor operators in referring to and using procedures" (NRC IR 50-277/92-18 and 50-278/92-18.)

October 15, 1992 - Unit-3 scrammed and recirculation pumps shutdown, “there was a significant cool down in the bottom head as a result of the loss of forced circulation” (IR 50-277/94-04 and 50-278/94-04.)

October 16, 1992 - The NRC identified programmatic weaknesses related to the System Manager program. (NRC IR 50-277/92-26 and 50-278/92-26.)

November 16, 1992 - The NRC noted: “An industrial safety concern, which involved the potential for loss of power in the drywell...had not yet been resolved and warrants your attention” (NRC IR 50-277/92-30 and 50-278/92-30.) (See December 12, 1995 for a related incident.)

December 2 and 11, 1992 - Failures of the containment, atmospheric, dilution (CAD) system gas analyzer occurred at Unit-2. On both occasions PECO personnel did not “understand” or “recognize” the problem with the CAD. (NRC IR 50-277/92-29 and 50-278/92-29.)

December 4, 1992 - Several weaknesses were reported during the the Initial SALP of Licensee Performance “including numerous component failures, lapses in the operating procedure and deficiencies in engineering and technical support" (York Daily Record, January 9, 1993.) "Among the areas identified for improvement were plant performance monitoring and engineering and technical support" (PECO, Report to the Shar eholde r s, March 1, 1993.)

December 7, 1992 - During Unit-2 start-up, the '2B' Recirculation Pump failed. (NRC IR 50-277/92-32 and 50-278/92-32.) (See March 2, 1993 for a related incident.)

December 17, 1992 - Turbine control oscillations occurred while Unit-2 was operating at 89.5% power. The plant was "stabilized" at 76.5% power. (NRC IR 50-277/92-32 and 50-278/92-32.)

December 19, 1992 - An Unusual Event was declared "due to a loss of emergency communications capabilities. Both units were operating at 20% power" (NRC IR 50-277/92-32 and 50-278/92-32.)

January 1, 1993 - The Unit-2 high pressure coolant injection system was declared inoperable. (NRC IR 50-277/92-32 and 50-278/92-32.) (See January 25 and 31, March 1 and August 9, 1993, for related incidents.)- January 21, 1993 - A Notice of Violation (NOV) was issued relating to the NRC's Motor-Operated Valve (MOV) Inspection on October 19-23 and November 3, 1992. PECO "1) did not document nonconforming positions, 2) did not properly disposition existing nonconforming conditions, and 3) did not take timely corrective actions to evaluate and resolve nonconforming conditions in MOVs..." (NRC IR 50-277/92-82; 50-278/92-82.) (See August 8-16, 1998, for a related incident.)

January 25, 1993 - During surveillance testing, the Unit-3 high pressure coolant injection system was declared inoperable. (NRC IR 50-277/93-01 and 50-278/93-01.) (See January 1 and 31, March 1 and August 9, 1993, for related incidents.)

January 31, 1993 - The Unit-2 high pressure coolant injection system was declared inoperable. (NRC IR 50-277/93-01 and 50-278/93-01.) (See January 1 and 25, March 1, and August 9, 1993, for related incidents.)

March 2, 1993 - Unit-2 scrammed while operating at 70% reactor power. (NRC IR 50-277/93-03 and 50-278/93-03.)

March 2, 1993 - The Unit-2 '2A' reactor recirculation pump and '2A' condensate pump tripped while the Unit was operating at 100% power" (NRC IR 50-277/93-03 and 50-278/93-03.) (See December 7, 1992 for a related incident.)

March 3, 1993 - The Unit-2 high pressure coolant injection system was declared inoperable. (NRC IR 50-277/93-03 and 50-278/93-01.) (See January 1, 25 and 31 and August 9, 1993 for related incidents.)

March 7, 1993 - [R]eactor scram, due to a low reactor vessel level. Reactor feed pump trip while lowering reactor power to with in bypass valve capacity, to allow work on turbine valves" (IR 50-277/94-04 and 50-278/94-04.)

March 10, 1993 - During a radiological safety inspection (February 8-9, 1993 and March 1-2, 1993), relating to a "breakdown of personnel access controls associated with the Transversing In-core Probe (TIP), the NRC found: "...control of personnel during such operations is considered very important as the TIPs represent one of the higher radiation sources that personnel have a potential for encountering" (NRC IR 50-277/93-02; 50-278/93-02.) (For related incidents see June 22 and 25, September 24, October 4, and November 11, 1993; June 19 and November 29, 1994 and August 24, 1995.)
March 23, 1993 - High oxygen concentration was found in Unit- 2 containment during power operation. (NRC IR 50-277/93-03 and 50-278/93-03.) (See January 17, 1992 for a related incident.)- April 24, 1993 - Unit-2 was manually scrammed "following declaration of all reactor vessel level instrumentation served by the '2B' condensing chamber inoperable" (NRC IR 50-277/93-06 and 50-278/93-06.) (See related incident on March 27 and July 26, 1992 and September 22, 1993.)

April 30, 1993 - A Notice of Violation was issued following an an NRC inspection of the electrical distribution system. Other design and operational weaknesses were identified relating to the emergency diesel generator. (NRC IR 50-277/93-80 and 50-278/93-80.) (See July 17, 1995 for a related development.)

May 26, 1993 - Three individuals were found to be "inattentive" or "sleeping." (C. Anderson, NRC Region I.)

June 22, 1993 - "Controls over a special high radiation area entry were not fully effective in that a higher than expected dose rate was identified upon the entry" (IR 50-277/94-04 and 50-278/94-04.) (See March 10, June 25, September 24 and October 4 and November 11, 1993 and January 19 and November 29, 1994.)

June 24, 1993 - PECO discovered a "mispositioned" control rod at Unit-2. The reactor was operating at 60% power. (NRC IR 50-277/93-15 and 50-278/93-15.) (For related events see February 22, 1994, April 21, 1995 and February 15, 1997.)

June 25, 1993 "[U]unlock[ed] high radiation area door" (IR 50-277/94-04 and 50-278/94-04.) (See March 10, June 22, July 22, September 24, October 4 and November 11,1993 and January 19 and November 29, 1994.)

July 4, 1993 - Unit 3 was shutdown. "An unplanned Unit 3 mid-cycle outage began on July 6, 1993, to replace to known leaking fuel bundles." A fuel leak was detected in May 1992. (NRC IR 50-277/93-15 and 50-278/93-15.)

July 30, 1993 - Unit-3 was manually scrammed "after a loss of condenser vacuum" (NRC IR 50-277/93-15 and 50-278/93-15.)

August 9, 1993 - The Unit-3 high pressure injection system was rendered inoperable (NRC IR 50-277/93-17 and 50-278/93-17.) (For related incidents see, January 1, 25 and 31 and March 1, 1993.)

August 11, 1993 - Unit-2 was manually scrammed. (NRC IR 50-277/93-17 and 50-278/93-17.)

August 14, 1993 - Unit-3 was shut down after three of four residual heat pumps were deemed inoperable. The plant was operating at 100% power. (NRC IR 50-277/93-17 and 50-278/93-17.)- September 14, 1993 - The reactor feed pump tripped due to "flow
oscillations" at Unit-3.

September 16, 1993 - An inspection of Peach Bottom's Emergency preparedness program on June 28-30, 1993 found: "Significant areas for potential improvement included wind direction information use by emergency response groups, event announcements in the Emergency Operations Facility by the ERM [Emergency Response Manager], and ERM recognition of the best indication of main stack radiation" (NRC IR 50-277/93-10; 50-278/93-10.)

September 22, 1993 - The NRC "noted that weaknesses in isolation of the reactor vessel water level instrumentation during installation of the [water level backfill] modification resulted in the generation of a false low signal. This low label signal caused the ECCS initiation signals and entry into a technical specification required shutdown condition at Unit 3" (For related incidents see, March 27 and July 26, 1992 and April 24, 1993.) Also the NRC completed their investigation into the recirculation pump trip on July 27, 1992. (NRC IR 50-277/93- 17 and 50-278/93- 17.)

September 24, 1993 - "Workers in Unit-3 were unaware of higher than expected radiation levels" (IR 50-277/94-04 and 50-278/94-04.) (See March 10, June 22 and 25, October 4 and November 11, 1993 and January 19 and November 29, 1994.)

September 24, 1993 - "During core off load a fuel bundle became stuck partially inserted in its storage rack in the Unit 3 fuel pool..." (NRC IR 50-277/93-24 and 50-278/93-24.) (See February 21-22, 1993 for related events.)

October 4, 1993 - An NRC inspection (August 2-6, 1993) found: "The lack of comprehensive corrective actions for some radiological discrepancies developed under the ROR [Radiological Occurrence Reporting] process was considered a significant radiological controls program weakness. A previous audit of the radiological controls program by the NQA [Nuclear Quality Assurance] identified a significant breakdown concerning radiological controls oversight. In particular, a weakness was noted in the area of radiation worker attention to detail and adherence to instructions provided by radiological controls staff" (NRC IR 50-277/93-19; 50-278/93-19.) (See March 10, June 22 and 25, October 4, September 24 and November 11, 1993 and January 19 and November 29, 1994.)

October 6, 1993 - "[C]ontrol switch for control room emergency ventilation left in the off position following restoration" (IR 50-277/94-04 and 50-278/94-04.)- November 11, 1993 "Unlocked high radiation door" (IR 50-277/94-04
and 50-278/94-04.) (See March 10, June 22 and 25, September 24 and October 4, 1993 and January 19 and November 29, 1994.)

November 15, 1993 - "5th point heater valve out of position following Unit-3 start-up, leading to a steam leak to the turbine building" (IR 50-277/94-04 and 50-278/94-04.)

November 22, 1993 - A Notice of Violation was issued for "a poor safety review of a temporary change to a reactor core isolation cooling testing procedure led to the inadvertent release of radioactive contamination within the Unit 3 reactor building. While this resulted in a minor clothing contamination, our review indicated poor management review and control of activities related to the specific testing" (NRC IR 50-277/93-24 and 50-278/93-24.)

December 18, 1993 - "Missed continuous fire watch" (50-277/94-04 and 50-278/94-04.) (See similar incidents on August 4, 1994 and January 11, 1998 and related data on Thermo-Lag, September 29, 1994 and October 1, 1996.)

January 1 , 1994 - Philadelphia Electric Company changed its name to PECO Energy Company.

January 19, 1994 - "During the inspection [October, 4-8 and November 8- 10, 1993] the NRC reviewed the circumstances associated with three examples of failure by three different individuals to adhere to procedural requirements concerning entries to high radiation areas in two cases, and a respiratory protection required area in the third case." A Severity Level III violation was announced by the NRC.
"Particularly disturbing to the NRC is the fact that the plant equipment operator, on October 27, and the engineer on October 29, willfully violated the

radiological controls in that they understood that they were no to enter the areas, yet did so anyway to complete certain tasks without first meeting the necessary radiation protection requirements. The entry by the engineer on October 29 was more significant since he had been warned by health physics personnel not to enter the area pending receipt of air activity results, yet did so anyway" (Thomas Martin, NRC, Regional Administrator, January 19, 1994.) (See March 10, June 22 and 24, September 24 and October 4, 1993 and November 29, 1994 for related incidents.)

January 24, 1993 - The High-Pressure Coolant Injection system was declared inoperable in Unit-3.

February 3, 1994 - Unit-3 was manually scrammed due to a Generator Field Ground alarm. The reactor was operating at 100% power.- February 22, 1994 - During power restoration at Unit-2, a control rod (38-15) was mispositioned for approximately two minutes. (For related events see June 24, 1993, April 21, 1995 and February 15, 1997.)

February 23, 1994 - A jet pump grappling hook was dropped into the Unit-3 spent fuel pool.

March 3, 1994 - Two four hour event notification reports were filed with the NRC due to the inoperability of the control room emergency system and problems associated with the Unit-2 high pressure coolant injection system. Both reports were later retracted.

March 9, 1994 - Increased contamination was detected in the Unit-3 high pressure coolant injection, pump room. As a result, seven shoe contamination reports were filed.

March 31, 1994 - A high-pressure coolant injection leak was identified. - Spring 1994 - "The Public Utility Commission (PUC) recently approved a settlement with PECO Energy Company (PECO.) PECO will give $217,000 to a grant program for low income consumers and pay a $24,000 fine for violating PUC regulations. For 1991, the PUC found 241 violations of the Commission's regulations. Many had to do with PECO's handling of billing disputes and service shut-offs" ("Utility Consumer Line," Bureau of Public Liaison, PA PUC, Spr ing/Summe r 1994.)

April 18, 1994 - Further weld thinning was identified in the Emergency Service Water supply .

April 27, 1994 - Unit-s experienced a reactor vessel water transient. "Pitting" was identified in this area in November 1993.

May 14, 1994 - Power was reduced at Unit-2 to "approximately 77% to perform a rod pattern adjustment and to repair a non-safety main steam moisture separator drain tank (MSDT) drain valve. During the power restoration on May 16, the 2A reactor recirculation pump (RRP) speed increased unexpectedly, (See September 22, 1995) causing reactor power to increase above the average power range monitor flow biased high power scram setpoint, resulting in a reactor scram" (IR 50-277/94-06 and 50-278/94-06.) (See October 24 and November 10, 1994.)

May 26, 1994 - A Severity Level IV violation was issued after the NRC "identified requirements for collecting a representative sample of the water river flowing into the site were not being met" (Edward C. Wenzinger, Chief, Projects Branch 2, Division of Reactor Projects, NRC.)- June 16, 1994 - The NRC reported the following problems during Peach Bottom's most recent Radiological Emergency Preparedness Exercise: "...14 Areas Requiring Corrective Action (ARCA), two Planning Issues (PI), and eight Areas Recommended for Improvement (ARFI) were identified in the Commonwealth of Pennsylvania and the State of Maryland combined." (James Joyner, Chief, Facilities Radiological Safety and Safeguards Branch, NRC.)

June 22, 1994 - "PECO made four 10 CFR 50.72 four hour notification reports to the NRC during the period. Subsequently, PECO retracted three of the event reports" (IR 50-277/94-06 and 50-278/94-06.)

June 23, 1994 - "The [NRC] inspectors continued to review the installation of the new control room radiation monitoring system...Specifically, system operating procedures were not in place when the system was placed in service and considered operable, the system was operated in an unanalyzed mode of operation because of unclear documentation, and one channel of the system was inadvertently removed from service due to the use of an improper drawing [A Notice of Violation was issued.]" Edward C. Wenzinger, Chief, Projects Branch 2, Division of Reactor Projects, NRC.)

June 30, 1994 - "Two small surface cracks were found last September in welds on the core shroud of Peach Bottom Unit 3 near Delta., Pa., said Bill Jones, a spokesman for PECO Energy Co., the plant's operator...The shrouds are 2-inch thick stainless steel cylinders that direct the flow of radioactive water around the fuel core. A nuclear reaction boils water into the steam used to generate electricity" (The Patriot News, July 1, 1994 A5.) (See June 30, 1994 and August 18, 1995.)

"Peach Bottom Unit No. 3 was initially examined during its refueling outage in the fall of 1993. Although crack indications were identified at two locations, the Company presented its findings to the NRC and recommended continued operation of Unit No. 3 for a two-year cycle. Unit No. 3 was reexamined during its refueling outage in the fall of 1995 and the extent of the cracking identified was determined to be within industry-established guidelines. The Company has concluded, and the NRC has concurred, that there is a substantial margin for each core shroud weld to allow for continued operation of Unit No. 3. Peach Bottom Unit No. 2 was initially examined during its October 1994 refueling outage and the examination revealed a minimal number of flaws. Unit No. 2 was re-examined during its refueling outage in September 1996. Although the examination revealed additional minor flaw indications, the Company concluded, and the NRC concurred, that neither repair nor modification to the core shroud was necessary. The Company is also participating in a GE BWR Owners Group to develop long term corrective actions." (PECO Energy Company, Form-10/K-A, 1999, p. 1999) A three-inch crack was identified in the reactor vessel shroud at Brunswick-1 in the summer of 1993. Cracks have also been found in the coreshrouds of Dresden-3 and Quad Cities-1. All of these reactors are GE Mark 1 designs.

July 18, 1994 - A Severity Level IV Violation was issued for failure to implement maintenance procedures on the Unit-2 high pressure coolant injection system. PECO issued an LER.

July 22, 1994 - "PECO identified that the existing instrument reference calibration placards were incorrectly installed with respect to the bottom of the torus of each unit" (IR 50-277/94-013 & 50-278/94-013.) PECO issued an LER.

July 27, 1994 - An NRC inspection "noted that there had been no indepth training provided to some of the [rad waste] shipping engineers since 1988...As such, the training provided to shipping engineers remains a program weakness. Licensee management informed the inspector they consider their current shipping engineer training program to be adequate" (IR 50-277/94-18 and 50-278/94-18.)

August 3, 1994 - "...PECO Energy personnel unknowingly placed the emergency cooling water system in a configuration that prevented safetyrelated equipment from receiving design cooling water flow rates...The overall safety consequences of this event were small...however, this condition represented a significant degradation in plant safety..." An enforcement conference was held on October 18, 1994. (Richard W. Cooper, II, Director, Division of Reactor Projects, NRC, September 29, 1994.) (See November 21, 1994 for civil penalty and violation.)

August 4, 1994 - PECO personnel missed a fire watch. (See December 18, 1993 and January 11, 1998 for related incidents, and August 10 and September 29, 1994 for more data.)

August 10, 1994 - A "minor" fire was extinguished on the Unit-2 reactor building roof. During this episode, the Unit-2 secondary containment was breached.

August 11, 1994 - The high-pressure, coolant-injection system was inoperable during maintenance activities. (See September 24, 1994 for related incident.)

August 17, 1994 - "...procedures were not implemented for the operation of the reactor building [Unit-3] ventilation and standby gas treatment system" (PECO Energy, Gerald R. Rainey, Vice President, Peach Bottom Atomic Power Station, October 19, 1994.) A Severity Level IV Violation was issued. - August 18, 1994 - An NOV was issued relating to vision problems of a LRO.

August 26, 1994 - A NOV was issued relating to Motor Operated Valve Testing

September 7, 1994 - A high-pressure, service water pump failed at Unit-3.

September 8, 1994 - "Standard and Poor's Corporation (S&P) has revised its rating outlook on the company from 'negative" to stable'" (J.F. Paquette, Jr., Chairman of the Board and Chief Executive Officer.)

September 20, 1994 - During the refueling outage, air bubbles were found leaking into the reactor cavity.

September 21, 1994 - PECO notified the NRC of a loss of shutdown cooling at Unit-2 due to a preventive maintenance operation.

September 23, 1994 - A broken fuel rod was discovered.

September 24, 1994 - A high- pressure, coolant-injection steam supply leak was discovered at Unit 3. (See August 11, 1994 for related incident.)

September 29, 1994 - "Thermal Science Inc. and its president, Rubin Feldman, were indicted September 29 by a federal grand jury on seven criminal charges, including willful violations of the Atomic Energy Act, a decade-long conspiracy to defraud the US government, false statements, and more. The charges are the culmination of a nearly two-year grand jury investigation of the company, which manufactures Thermo-Lag, the ineffective fire barrier used in more than 70 nuclear reactors [including Peach Bottom.]" (The Nuclear Monitor, October 17, 1994.) (See December 18, 1993 and October 1, 1996.)

October 10, 1994 - The NRC reported "four individuals entered the Unit 2 offgas pipe tunnel high radiation area (HRA), which was visibly posted as a HRA, and the individuals were not provided with the required radiation monitoring device, nor was positive control provided by an individual qualified in radiation

protection procedures, nor did the individuals adhere to posted instructions regarding entry requirements, a requirement of the Radiation Work Permit under which the entry was made" (IR 50-277/95-05 and 50-278/95-05 and Notice of Violation.) (See October 31, 1994, November 29, 1994 and March 14, 1995 for related incidents and Notice of Violation.)- October 16 -17, 1994 The Unit-2 reactor pressure vessel (RPV) exceeded
212 degrees F. "After reviewing operators' involvement in this event, Region I management initiated continuous coverage of the Unit-2 start-up, to ensure that operators performed a controlled and safe return of the unit to power operation" (Richard W. Cooper, II, Director, Division if Reactor Projects, November 21, 1994.) Severity Level IV Violations were issued.

October 21, 1994 - FEMA assessed a Deficiency against the State of Maryland Emergency Operations Center for communications failure during the full-participation exercise on August 22, 1994.
- October 24, 1994 - A Licensee Event Report (LER) was filed for "Main Safety Relief and Safety Valve Setpoint Drift." (See May 14 and November 10, 1 9 9 4 .)

October 27, 1994 - The DER reported that the "PECO inspection of the core shroud of Peach Bottom-2 did not find any significant flaws...Therefore, there is no repair needed for the time being." The NRC stated: "During the Unit 2 outage PECO conducted an ultrasonic inspection of the reactor vessel core shroud accessible weld areas. These examinations identified cracking of a similar nature found at Unit 3, but of much less magnitude. Based on an engineering analysis of the examination results, PECO determined that the Unit 2 shroud was structurally sound and that no actions were required to ensure its stability over the next operating cycle" (IR 50-277/94-21 & 50-278/94-21.) (See June 30, 1994 and August 18, 1995 for related incidents.)

October 31, 1994 - The NRC reported "a Senior Reactor Operator (SRO) entered the Unit 2 high pressure coolant injection (HPCI) turbine room, which was visibly posted as a HRA, and the individual was not provided with the required alarming dosimeter, nor positive control provided by an individual qualified in radiation protection procedures, nor did the individuals adhere to posted instructions regarding entry requirements, a requirement of the Radiation Work Permit under which the entry was made" (IR 50-277/95-05 and 50-278/95-05 and Notice of Violation.) (See October 10, 1994, November 29, 1994 and March 14, 1995 for related incidents and a Notice of Violation.)
November 10, 1994 - A LER was filed for "Non-Conservative Flow Biased Setpoints." (See May 14 and October 24, 1994.)

November 18, 1994 - "A load drop to about 55% power occurred on November 18, 1994, to support cleaning of the main condenser waterboxes." Unit-2 returned to full power the following day. (IR 50-277/94-27 & 50-278/94-27.) (See May 31,July 16, September 10 and October 25, 1996; and, September

12, 1997 for related incidents.)- November 21, 1994 - The NRC proposed a Severity Level III Violation and
an $87,500 fine for the emergency service water configuration problem on August 3, 1994.

November 21, 1994 - Three items of weakness were noted by an NRC Nondestructive Examination Laboratory Inspection: "these were not marking the weld centerline on welds for UT [ultrasonic inspection] as part of the ISI [inservice inspection] program, not finding or recording a geometric reflector in excess of 50% of DAC [distance amplitude correction] while conducting UT per the ASME [American Society of Mechanical Engineers] code on a RWCU [reactor water clean-up] system weld, and having radiographs that show signs of aging in storage for work performed after original construction" (IR 50-277/94-28 & 50-278/94-28.)

November 29, 1994 - "Two separate events occurred, involving a total of five radiation workers, where personnel entered a high radiation area without having the required dose rate monitoring equipment. Individually, these events were of low radiological consequence; however, they reflect a continuing station weakness in personnel adherence to posted boundary requirements (Section 6.0). These events are considered an Unresolved Item (URI- 94-25-01) (IR 50-277/94-25 & 50-278/94-25.)
"While we recognize that you are aggressively taking actions* to prevent recurrence the events are similar in nature to other recent radiological events for which escalated enforcement action was taken" (Clifford J. Anderson, Section Chief, Projects Section 2B, Division of Reactor Projects.) (For related incidents see October 10 and 31, 1994 and March 14, 1995
*For similar events see March 10, June 22 and 25, September 24 and October 4, 1993 and January 19, 1994.
- December 9, 1994 - PECO made a four hour event notification after the utility discovered two doors that separate the main stack from the environment were left open for four hours.

December 12, 1994 - PECO was among a consortium of 33 utilities actively pressuring the Mescalero Apaches to build a high-level radioactive waste dump on their land.

December 19-23, 1994 - An inspector "identified a condition where manual operation of fire protection system controls located outside of the vital security areas could affect the operation of vital safety systems" (William H. Ruland, Chief, Electrical Section, Division of Reactor Safety, NRC, February 3, 1995.)- December 20, 1994 - An NRC inspector determined there was poor control over the use of a non safety-related battery charger at Unit-2.

December 22, 1994 - A steam/water discharge to the reactor building during reactor water cleanup system testing resulted in minor shoe contamination to three individuals and contamination in portions of the Unit-2 reactor building.

January 7, 1995 - "Reactor power was reduced to below 75% [Unit 2]...to allow for the repair of a steam leak that developed from the stem packing of an outboard MSIV" (IR 50-277/95-10 and 50-278/95-01.)

February 14, 1995 - A Violation was issued (Severity Level IV) for PECO's "failure to properly evaluate the installation, during outages in 1993, of 'temporary' shielding above each bank of hydraulic control units (HCU) at Units 2 and 3 (four locations total), which shielding is till in place...your staff's response, past and present, to questions about the shielding arrangements demonstrated a poor questioning attitude" (Clifford J. Anderson, Section Chief, Projects Section 2B, Division of Reactor Projects, NRC.)

March 1, 1995 - A High Pressure Service Leak was identified by PECO at Unit-2.

March 6, 1995 - "...operational errors involving a mis-positioned valve, an inadequate valve position verification, and poor communications resulted in the loss of keep fill pressure on the 2B core spray (CS) sub-system [Unit 2.]" (IR 50-277/95-04 and 50-278/95-04.)

March 14, 1995 - "However, based on the results of this inspection, certain of your activities were in violation of NRC requirements, as specified in the enclosed Notice of Violation (Notice). The violation is of concern and being cited because of the number of improper high radiation area entries which are described in the enclosed inspection report...in the most recent events, radiological control personnel failed to carry out their assigned duties in accordance with radiological control management's expectations; no similar causal factors were identified in the 1993 events.") (James H. Joyner, Facilities Radiological Safety and Safeguards Branch, Division of Radiation Safety and Safeguards, NRC.)

March 17, 1995 - "An automatic recirculation pump runback reduced power [Unit-2] to about 70% on March 17, because of a mis-conducted reactor feed pump test." (IR 50-277/95-04 and 50-278/95-04.) The incident was caused by an operator error. (See related incidents on March 4, 1996 and May 16 and June 7, 1998.)- March 19, 1995 - High Pressure Coolant Injection (HPCI) suction valve
was mispositioned at Unit-2 due to operator error. A Notice of Violation was issued. (Severity Level IV.) "Also, two subsequent shift turnover panel walkdowns failed to identify the abnormal system line-up and allowed the HPCI system to remain in the abnormal lineup for 18 hours." (Clifford J. Anderson, Section Chief, Projects Section 2B, Division of Reactor Projects.)

March 23, 1995 -Unit-3 was manually scrammed "after the air-operated main steam supply isolation valve to the 'B' steam jet air ejector (SJAE) failed closed causing a loss of condenser vacuum." (IR 50-277/95-08 & 50-278/95-08.)

April 10, 1995 - "The inspectors opened the three unresolved items pending review of your staff's assessment and planned corrective actions. The first issue addresses the possibility that, due to an equipment failure, a low pressure coolant injection sub-system (one of four) was not maintained with its piping full to prevent water hammer following an injection. The second issue deals with the secondary containment flood control portion of your emergency operating procedures, which could lead an operator to flood two emergency cool cooling pumps rooms, a condition outside the plant's design basis. Lastly, the third issue deals with inconsistencies between the standby liquid control system inservice testing methodology and ASME Section XI requirements for pump run time before operational data is requested." (Clifford J. Anderson, Section Chief, Projects Section 2B, Division of Reactor Projects.)

April 16, 1995 - All control rods were "conservatively" declared inoperable at Unit-2 for 4.5 hours.

April 21, 1995 - Control rod 46-07 "unexpectedly drifted" out of position at Unit-2. (IR 50-277/95-08 & 50-278/95-08.) (For related events see June 24, 1993, February 22, 1994 and February 15, 1997.)

April 24, 1995 An unplanned power reduction to 35% occurred at Unit-3 when the 3B reactor recirculation pump tripped. (See May 13, 1995 for related development.)

May 13, 1995 - The 3B reactor recirculation pump "unexpectedly" tripped. (See April 24, 1995 for related incident.)

May 24, 1995 "...several events involving plant operators indicate a negative trend in plant operations performance. These instances include problems with procedural adherence, attention to detail, and control of maintenance activities." Executive Plant Performance Results, Richard W. Cooper, NRC, Director, Division of Reactor Projects.)- June 10, 1995 - "Unplanned Engineered Safety Feature Actuation During Diesel Testing" caused a Licensee Event Report. (IR 50-277/95-15 & 50-278/95-15.)

June 13, 1995 - The calibration check of the Feedwater Inlet Temperature instruments utilized equipment that was later "found out of tolerance." (IR NOS. 50-277/98-01 AND 50-278/98-01.)

June 18, 1995 - "Condition prohibited by TS when two EDGs were Inoperable at the same time" caused a Licensee Event Report. (IR 50-277/95-15

& 50-278/95-15.) (See August 17, 1995 for proposed fine. Related incidents begin on December 10, 1996.)

June 29, 1995 - "During the conduct of troubleshooting an electrical ground on the Unit 3 station battery, we noted an apparent lack of attention to detail and questioning attitude on the part of your staff." (Glenn W. Meyer, Chief, BWR & PWR, Division of Reactor Safety, NRC.)

July 6, 1995 - A Licensee Event Report occurred when due to a, "High Pressure Coolant Injection System Valve Motor Failure."

July 10, 1995 - The NRC accepted the following changes at Peach Bottom, "... eliminating the Independent Safety Engineering Group composition commitment while retaining the independent technical review function, relocating Nuclear Review Board requirements, and reducing the frequency of certain nuclear quality assurance audits." (Michael C. Modes, Chief, Materials Section, Division of Nuclear Safety, Nuclear Regulatory Commission.)

July 17, 1995 - "Inspector review of the E-2 and E-4 emergency diesel generator modifications indicated that pre-existing drawing errors [see April 30, 1993] and insufficient post-modification testing caused both operating reactor units to be placed in a situation where only two emergency diesel generators (i.e., E-1 and E-3 operable; E-4 in a maintenance outage, while the E-2 output breaker would not automatically close) remained able to automatically respond to a loss of off site power or a design basis accident condition. The inspectors also identified that inadequate review of the modification led to a loss of power of an emergency power bus during testing, and the introduction of a design flaw such that E-2 and E-4 were not able to automatically perform their safety functions..."The emergency diesel generator modification issues are of concern to us since your normal design and testing process did not uncover a basic error that would have led to the E-2 and E-4 machines being unknowingly inoperable. This condition could have remain unknown until challenged or until the Unit 3 Fall 1995 post outage loss of off site power testing. Based on these results of the inspection, three apparent violations were identified and are being considered for escalated enforcement action..." (Richard W. Cooper II, Director, Division of Reactor Projects, NRC.)
(See August 17, 1995, for enforcement information.)

July 21, 1995 - The NRC's review of PECO's emergency preparedness plans at Limerick and Peach Bottom found: "...quality control was lacking for Emergency Plan [EP] and procedure revisions, as the omission of a portion of an essential paragraph, concerning public emergency information, as well as numerous other minor errors, was found. Inspectors also noted that the corporate EP staff had no documented plan in place to carry out the EP training of corporate emergency responders." (James H. Joyner, Chief, Facilities Radiological Safety and Safeguards Branch, Division of radiation safety and safeguards, NRC.)

July 30, 1995 - Unit-3 scrammed "on high reactor water level due to a control signal failure for the 3A reactor feed pump." (IR 50-277/95-15 & 50-278/95-15.) (See November 6, 1995 for a related incident.)

August 9, 1995 - An Unusual Event was declared for a "potentially contaminated injured man being transported off-site by ambulance..." (IR 50-277/95-15 & 50-278/95-15.)

August 13, 1995 - PECO identified excessive average control rod scram times at Unit-3.

August 14, 1995 - PECO failed to meet technical specification requirements when a Reactor Water Clean-up temperature switch was found to be inoperable.

August 15, 1995 - The NRC determined a partial loos of off-site power was cause by poor maintenance activities.

August 17, 1995 - The NRC proposed a $50,000 fine for the Severity Level III violation associated with EDGs identified on July 17, 1995.

August 18, 1995 - "HPCI [High Pressure Coolant Injection steam lines] system piping in both units is experiencing high vibration levels due to unknown causes." (IR 50-277/95-18 & 50-278/95-18.)- August 18, 1995 - The NRC identified a crack about 3" (length) by 2.5.
"...The crack is believed to be caused by intergranular stress corrosion (IGSC)." (IR 50-27/95-18 & 50-278/95-18.) Rich Janati of the Pennsylvania Department of Environmental Protection stated, "...the new cracks are not exactly on the core shroud. They are on the core spray line." (September 5, 1995.) (See June 30, 1994 and October 27, 1994 for related incidents.)

August 24, 1995 - During the disassembly of a transversing incore probe (TIP), the NRC "identified weaknesses in personnel communications, understanding of radiological conditions associated with the work activity, supervisory oversight, and control of contractor work activities. (See March 10, June 22 and 25, September 24, October 4 and November 4, 1993 and June 19 and November 29, 1994). Four examples of personnel failing to adhere to radiation protection procedures, a violation of NRC requirements [Severity Level IV], were identified." James H. Joyner, Chief, Facilities Radiological Safety and Safeguards Branch, Division of Radiation Safety and Safeguards, NRC, September 22, 1995.) (See March 10, 1993 for a related incident.)

August 25, 1995 - Reactor power was reduced at Unit-3 to 30% due to a problem with a main turbine control valve.

September 22, 1995 - At Unit-3 "an unexpected reactor recirculation pump (RRP) motor generator (MG) set trip occurred due to a maintenance technician inadvertently bumping a loose resistor lug in the RRP in the RRP MG control cabinet." (IR 50-277/95-22 & 50 2787/95-22.) (See May 14, 1995.)

October 18, 1995 - Excessive scram times were identified at Unit-3.

October 20, 1995 - Results of examinations of senior reactor operators "reflect an unexpected poor level of performance in the simulator." (Michael C. Modes, Acting Chief, Operator Licensing and Human Performance Branch, Division of Reactor Safety, NRC.) (See December 27, 1995 for follow-up report.)

October 22, 1995 - Power was reduced to 90% at Unit-2 "in response to a loss of feedwater heating caused by a partial loss of offsite power. During the recovery from this event, PECO discovered that an existing '5B' feedwater heater (FWH) leak had degraded. PECO returned reactor power to 100% until October 26, when PECO reduced power to 68% to isolate the 'B' FWH train and then limited Unit 2 power operations to 95% power. On November 4, PECO declared the 'C' safety relief valve inoperable because of a leaking bellows. On November 7, PECO returned the unit to 100% power after completing a safety evaluation allowing full power operation with one train isolated. Full power operations continued until November 20, when PECO reduced power to 95% to minimize vibration of the 2A reactor feed pump (RFP)." (IR 50-277/95-26 & 50-278/95-2 6 .)- October 27, 1995 - An NRC inspection found two, technical unresolved issues: 1)...Peach Bottom fire protection program and the impact of inadvertent discharge of CSR (cable spreading room) carbon dioxide system on the installed safety equipment; and 2)...the appropriateness of Peach Bottom's response to an inadvertent carbon dioxide discharge alarm." (IR 50-277/95-24 & 50-278/95-2 4 .)

November 6, 1995 - At Unit-3, an "unexpected"t trip occurred at the '3A' circulating water pump. (See September 2, 1997 and, January 14, 1998, for related incidents.)

December 2, 1995 - A main turbine trip caused a full reactor scram at 100% power Unit-3.

December 5, 1995 - On September 22, 1995 A Notice of Violation was issued relating to PECO's "failure to adhere to radiation protection procedures...We have evaluated your response to the violation and found that you have not completely responded as required by the Notice of Violation. While your response identifies immediate actions that were taken, it does not adequately address generic and long-term actions to prevent recurrence. For example, you indicate that a Performance Enhancement Process (PEP) investigation was initiated to determine the causes and reasons for the contamination event, and that the actions taken as a result of that effort are expected to prevent recurrence. However, you have not indicated what the findings of that effort revealed (i.e., what were the causes and reasons), and what consequent corrective actions were implemented to address those factors. Further, you indicated that a Quality Improvement Team (QIT) performed an evaluation of the work process, and their recommendations will improve radiological and work control. However, you did not provide any discussion of what recommendations were implemented and how improved performance will be be achieved." (James T. Wiggins, Director, Division of Reactor Safety, NRC, December 5, 1995.)

December 12, 1995 - A Severity Level IV Notice of Violation was issued due to PECO's failure to monitor drywell leakage at Unit -3. "Specifically, a modification prepared by your engineering staff lead to the installation of drywell drain tank pump control instrumentation that did not function as designed. Further, post-maintenance testing should have identified the problem and did not. Operators also initially failed to identify that the drywell pumps were not functioning, based on changes in in the calculated drywell leakage." A similar incident occurred in October 1994 at Unit-2 according to the NRC. (Walter J. Pasciak, Section Chief, Projects Branch 4, Division of Reactor projects, NRC.) (See November 16, 1992 for a related incident.)- December 27, 1995 - On December 14, 1995, PECO and the NRC held a meeting to determine the causes of "weak performance" on operator exams. (See October 20, 1995.) The Company's conclusions included "... the unrecognized need for senior reactor operator (SRO) candidates to have additional plant familiarization, the weak understanding of system details including protection and control logic, the need to upgrade the cognitive level of written questions, and the infrequent evaluation of the candidates' ability to prioritize mitigating actions during simulator scenarios. In addition, your staff stated that your guidance for examination validation and proadministration review will be revised to promote prompt escalation of any unresolved examination concerns to PECO Energy management." (Glenn W. Meyer, Chief Operator, Licensing and Human Performance Branch, Division of Reactor Safety, NRC, December 27, 1995.)

January 20, 1996 - Power reduced at both units due to the high river level.

January 30, 1996 - The NRC praised the radioactive waste program but "noted weaknesses in training provided shipping personnel on radioactive material hazards and considered this an unresolved item." (Walter J. Pasciak, NRC, Chief Projects Branch 4, Division of Reactor Projects.)

February 1, 1996 - Power was reduced at Unit 3 "for condenser water box cleaning. (IR 50-277/96-01 & 50-278/96-01.)

February 2, 1996 - Plant operators "identified a hydrogen leak on the Unit 3 generator neutral bushing. Operators reduced power to 23% to remove the generator from the grid and effect repairs." (IR 50-277/96-01 & 50-278/96-01.)

February 3, 1996 - At Unit-2, power was reduced to "85% for repair of a hydraulic control unit and rod pattern adjustment." (IR 50-277/96-01 & 50-278/96-01.)

February 5, 1996 - Power was reduced at Unit 2 to 78% "in response to a loss of condenser vacuum event..." (IR 50-277/96-01 & 50-278/96-01.)

March 4, 1996 - Power was stabilized at 65% power at Unit 2 after "a recirculation pump runback due to the 2B reactor feedwater pump (RFP) trip." (IR 50-277/96-01 & 50-278/96-01.) (See related incidents on March 17, 1995 and May 16 and June 17, 1998.)- March 25, 1996 - The NRC issued two violations during a routine
inspection. "They involved not properly performing functional testing of the safety-related degraded grid under voltage relays to ensure their operability, and inadequate controls over a 125 vdc circuit breaker supplying power to portions of the Unit 2 remote shutdown panel." (Walter J. Pasciak, NRC, Chief, Projects Branch 4, Division of Reactor Projects.) (See April 24, 1996.)

April 17, 1996 - The Unit-2 "High Pressure Coolant Injection (HPCI) system was declared inoperable and removed from service following the discovery of a 10 drop per minute leak from the inlet nipple of the HPCI cooling water line relief valve." (IR -277/98-02; 50-278/98-02.)
- April 24, 1996 - Two Severity Level IV violations were issued by the NRC. "...since 1989, PECO had calibration data that indicated that the 98% and 89% degraded bus under voltage relay setpoints were found to be outside of the Technical Specification allowable values and did not take appropriate actions to the correct the issue...Contrary to the above, PECO did not properly identify or implement corrective actions to identify and correct an adverse circuit breaker position that caused portions of the Unit 2 Remote Shutdown panel to not receive alternate control power for over a year. This failure led to several functions of the remote shutdown panel being inoperable from October 1994 through January 1996." (PECO Nuclear, Thomas N. Mitchell, Vice President, Peach Bottom Atomic Power Station.) (See March 25, 1996.)

Spring, 1996 - PECO Energy Company has expressed interest in an Energy Department proposal to use fuel made from decommissioned warheads at Peach Bottom and Limerick. Peco spokesman William Jones stated, "It is just something we've expressed interest, if the DOE picks up the cost and there is a net benefit for our customers." But Greenpeace spokesman Tom Clements observed, "Consumers now will be forced to produce bomb material and encourage international plutonium use by simply flipping their light switch."

All told, eighteen utilities, including a Canadian entity, are interested in using fuel made from weapons-grade plutonium. (From U.S. Newswire, Greenwire and The Houston Chronicle.)

May 9, 1996 - Power was reduced to 65% at Unit 2 due to turbine control valve (No. 2) failure.

May 9, 1996 - An Notice of Violation was issued when "Control Room Emergency Ventilation Filter Train 'A' Test, was identified as being out of sequence." (NRC, August 6, 1996.) - May 31, 1996 - Power was reduced at Unit 3 to 62% "to allow condenser waterbox cleaning, control rod pattern adjustments, and other preventive maintenance activities." (IR 50-277/96-04 and 50-278/96-04.) (See November 18, 1994; July 16, September 10 and October 25, 1996; and, September 12, 1997 for related incidents.)

May 22, 1996 - A Notice of Violation was issued for "...an unexpected loss of the Unit 2 'B' RPS power supply occurred when an equipment operator mispositioned the voltage adjustment rheostat for the ORS Alternate feed transformer." (NRC, August 6, 1996.)

June 3, 1996 - The NRC notified PECO that "we are unable to close your NRC Generic Letter 89-10 motor operated valve program at this time." (Walter J. Pasciak, NRC, Chief, Projects Branch 4, Division of Reactor Projects.)

June 9, 1996 - Power was reduced to 71.5% at Unit 2 "to secure the 2C reactor feed pump (RFP) for scheduled maintenance." (IR 50-277/96-04 and 50-278/96-04.)
June 12, 1996 - "...the hatch between the Unit #3 refuel floor and the refuel floor roof was propped open to allow access to the roof for performance...Personnel performing this test believed that the only procedural requirement to open the hatch was to have a security guard present." (August 6, 1996.)

June 22, 1996 - Power was reduced to 25% at Unit 3 "to repair electrohydraulic control (EHC) oil leaks on the No. 4 TCV [Turbine Control Valve] and No.2 TSV." (IR 50-277-96-04 and 50-278/96-04.) (See June 23, 1996 for related incident.)

June 23, 1996 - "Manual unit shutdown and forced outage [Unit 3], during the June 22 load drop the No. 2 TCV [Turbine Control Valve] mechanically failed. PECO completed the outage and restarted the unit on June 27, the unit reached 100% on June 28. (See June 22 1996 for related event.)

July 16, 1996 - Power was reduced to 72% at Unit-3 for main condenser waterbox cleaning. (See November 18, 1994; July 16, May 31, September 10 and October 25, 1996; and September 12, 1997 for related incidents.)

August 2, 1996 - Power was reduced to 70% at Unit-3 "to transfer the steam jet air ejectors and repair a steam leak from the packing of the steam isolation valve." (IR 50-277/96-06 and 50-278/96-06.) (See August 10, 1996 for a related incident.)- August 6, 1996 - A Notice of Violation was issued after NRC inspectors
"noted three examples where station personnel performed activities without properly implementing the established written procedures. These procedural adherence deficiencies involved various parts of the site organization and indicated a decline in station procedural adherence." Walter J. Pasciak, NRC, Chief, Projects Branch 4, Division of Reactor Projects.

August 6, 1996 - Power was reduced to 85% at Unit-3 "in response to an off-gas recombiner isolation." (IR 50-277/96-06 and 50-278/96-06.)

August 10, 1996 - Power was reduced to 55% at Unit-3 "to transfer the steam jet air ejectors." (See August 2, 1996 for a related incident.)

September 1, 1996 - "...the Company's stock price under performed the Dow Jones Utilities Index and S&P 500 Stock Index due to the forced shutdown of Salem Units No. 1 and No. 2, uncertainty about the pace of competition in Pennsylvania and the decline in 1996 earnings [down $0.24 per share.]" ("Report to Shareholders, " J.F. Paquette, Jr., Chairman of the Board.)

September 5, 1996 - PECO joined a consortium of utilities asking the DOE "to consider them as candidates for the disposal of U.S. and Russian stockpiles of weapons-grade plutonium...Under the proposal, the utility companies would burn fuel pellets hat include small amounts of plutonium oxide in addition to the pellet's traditional ingredient, uranium oxide..." (AP, September 5, 1996.)

September 10, 1996 - Unit-3 "...unit load was reduced to approximately 75% power for condenser water box cleaning." (See October 25, 1996, for related incident.) (IR 50-277/96-08 & 50-278/96-08.)

September 20, 1996 - "...with Unit 3 shutdown, the maintenance personnel mistakenly pulled the primary containment isolation system (PCIS) inboard and outboard mechanical vacuum pump trip logic fuses...while working on a local leak rate test activities". (IR 50-277/97-04 & 50-278/97-04).

October 1, 1996 - The Nuclear Regulatory Commission (NRC) fined Thermal Science, Inc. (TSI) $ 9 0 0 , 0 0 0 for "deliberately providing inaccurate or incomplete information to the NRC concerning TSI's fire endurance and ampacity testing programs." (James Lieberman, Director of Enforcement.) The fine was the largest assessed against a nuclear contractor and the second highest in the agency's history. In 1992, the NRC declared TSI's fire barrier, ThermoLag, "inoperable." (For related incidents, see December 18, 1993, September 29, 1994, May 19, 1998, October 12, 1999, and July 21, 2000.)

October 6, 1996 - Unit-2 scrammed due to equipment problems. (See October 15, 1996 for a related incident. Also, see November 18, 1994 and May 31 and July 16, 1996 for related problems.)- October 9, 1996 - "Based on the results of this inspection, an apparent violation was identified and is being considered for escalated enforcement action...Specifically, the failure to establish adequate performance criteria that would demonstrate appropriate preventive maintenance for several systems and components was identified." (NRC, James T. Wiggins, Director Division of Reactor Safety.)

October 10, 1996 - "The violation deals with your procedures allowing operation of the [standby gas treatment] system that was unanalyzed in accordance with the updated final safety analysis report..." A predecisional enforcement conference was also announced. (NRC, Richard W. Cooper, II, Director, Division of Reactor Projects.)

October 15, 1996 - Unit-2 scrammed for the second time in nine days due to equipment problems.

October 25, 1996 - Unit-3 "...unit load was reduced to about 58% for waterbox cleaning, control rod drive scram testing time, and 3A reactor feed pump maintenance." (See September 10, 1996 for a related incident. Also, see November 18, 1994; May 31 and July 16, 1996; and, September 12, 1997 for related problems.) (IR 50-277/96-08 & 50-278/96-08.)

October 29 - 1996 - Unit-3 "power was reduced to about 60% power to mitigate a lowering condenser vacuum condition which developed due to off-gas recombiner system problems." (IR 50-277/96-08 & 50-278/96-08.)

December 10 and 27, 1996 - Emergency diesel generator power fluctuations were reported. (IR 50-277/97-01 & 50-278/97-01.) (See December 27, 1996 and January 24, February 7 and March 6, 1997 for related developments.)

December 18, 1996 - The NRC recognized two, Severity Level IV violations during an inspection from September 8, through November 9, 1996: "The first issue involved the failure to maintain an adequate contractor qualification program, to ensure the qualification of contractor personnel performing independent safety-related work activities. The second issue involved the failure of engineering and operation personnel to identify and prevent the calibration of average power range monitors outside of the technical specification limits. This resulted in a failure to enter a technical specification required shutdown action statement for inoperable average power range monitors." (Walter J. Pasciak, NRC, Chief, Projects Branch 4, Division of Reactor Projects.)- December 20, 1996 - "Based on the results of this inspection, an apparent violation was identified and is being considered for escalated enforcement...The apparent violation concerned the failure to control safeguards information in accordance with NRC requirements. The circumstances surrounding this apparent violation, the significance of the issue, and the need for lasting and effective corrective action were discussed with members of our staff at the inspection exit meeting on November 27, 1996." (James T. Wiggins, Director, Division of Reactor Safety, NRC, December 20, 1996.)

December 27, 1996 - The NRC cited PECO for a violation involving the failure to verify a modification change on an emergency diesel generator. (IR 50-277/96-06 & 50-278/96-06.) (See December 10, 1996 and January 24, February 7 and March 6, 1997 for related developments.)

January 3, 1997 - A Severity Level III Violation was issued by the NRC for "the failure to establish, for several structures, systems, and components (SSC), adequate performance criteria to monitor the effectiveness of preventive maintenance...Since this violation involved multiple examples of failures to establish, or adequately establish, performance criteria...the violation has been categorized at Severity Level III..." (NOV 50-277/96-07 & 50-278/96-07.)

January 8, 1997 -FEMA identified several deficiencies during the emergency preparedness drill on November 19, 1996 including: coordination of information with the York County Communication Center and the county's emergency management staff and the failure of the Cecil County Emergency Operations Center to notify the public promptly and maintain the proper notification sequence.

January 21, 1997 - NRC inspectors determined that core thermal power was operating at a rate greater than mandated in the technical specifications since June 12, 1995, due to improperly calibrated feedwater temperature instruments. (IR 50-277/97-01 & 50-278/97-01.) "Thus, this issue represented a missed precursor event." (June 4, 1997, IR 50-277/97-02 & 50-278/97-02.)

January 21, 1997 - High Pressure Coolant Injection stop valve timing and gland condenser gasket failure was reported at Unit-3. A similar event occurred in August 1996. (IR 50-277/97-01 & 50-278/97-01.)

January 24, 1997 - PECO declared the EDG [E1] inoperable due to observed power swings of 200 to 300 KW while increasing load, 500 KW at rated load, and a 500 KW during shutdown." (IR 50-277/97-01 & 50-278/97-01.) (See December 10 and 27, 1996 and February 7 and March 6, 1997 for related de v e lopment s .)- February 1, 1997 - "...an unexpected reactor water conductivity increase " followed a "load drop." (IR 50-277/97-01 & 50-278/97-01.)

February 7, 1997 - An "unresolved item" was identified during an inspection "dealing with your staff's inability to identify the cause of load fluctuations on the E-1 emergency diesel generator during testing operations. This item was of concern since, without a root cause, the possible affects on operability may not be clearly identifiable." (Walter J. Pasciak, NRC, Chief, Projects Branch 4, Division of Reactor Projects.) (See December 10 and 27, 1996 and February 7 and March 6, 1997 for related developments.)

February 10, 1997 - Two violations were identified in the turbine building. "These violations involved failure to assure that the turbine building atmosphere was processed through the turbine building gaseous waste treatment system as specified in the ODCM, and failure to provide an adequate safety evaluation to support certain aspects of the modification in accordance with 10 CFR 50.59." (John R. White, NRC, Chief, Radiation Safety Branch, Division of Reactor Safety.) (See May 7, 1997, for NRC rebuke on PECO's lack of followu p.)

February 15, 1997 - "...with Unit-3 at 100% of rated power, while performing [a control rod exercise], the reactor operator (RO) selected control rod 58-39 and moved it in, from position 48 to 46. Subsequently, after becoming distracted by a telephone call, the operator returned to the test and mistakenly moved control rod 58-43, from position 48 to 46, without first returning control rod 58-39 to position 48." (IR 50-277/97-01 & 50-278/97-01.) (For related events see June 24, 1993, February 22, 1994 and April 21, 1995.)

February 27, 1997 - "PECO Energy Inc. had a yield of 7.44 percent...Those are stocks to be avoided" because these companies are high-cost producers that may not be able to afford to keep paying their dividends, said Miller, who manages the Better Than Bonds/Utility.' (Dow Jones News Service.)

March 1997 - "Common stock earnings for the year ended December 31, 1996, were $2.24 per share, $0.40 per share lower than last year." (PECO Energy, "Report to Shareholders", J. F. Paquette, Jr., Chairman of the Board.)

March 6, 1997 - On March 6, operators declared the E-3 EDG inoperable because of observed fluctuations in generator output load..." (IR 50-277/97-01 & 50-278/97-01.) (For related developments see December 10 and 27, 1996 and January 24 and February 7, 1997.)

March 9, 1997 - A manual reactor scram was initiated at Unit 3 "...as operators lowered reactor power to allow a drywell entry to correct the low lube oil level, the A recirculation pump tripped..." The reactor returned to operation three days later. (IR 50-277/97-02 & 50-278/97-02.)- March 24, 1997 - The Dow Jones utilities average "has dropped 8.1 percent since reaching a 52-week high in late January on the expectation that the Fed will soon raise interest rates, investors said. Niagara Mohawk Power

Corp., PECO Energy Corp. and Unicom Corp. led the drop. The Dow Jones Industrial average, meanwhile, is little changed for that period." (Bloomberg Business Service.)

March 25, 1997 - Inadvertent shutdown of Unit-3 drywell chiller occurred. (See August 22, 1998 for a repetitive incident.)

April 1, 1997 - At Unit 2, "Reactor power was reduced from 100% to approximately 48% due to a leak at a main turbine control valve (TCV) drain line." (IR 50-277/97-02 & 50-278/97-02.)
In addition, "... the 2' A' Reactor Feedwater Pump Turbine high water level trip capability was inoperable for greater than two hours while Unit 2 reactor power was [greater than] 25%." (IR 50-277/98-03; 50-278/98-03.) The NRC issued a Level IV violation. (Also, see November 7, 1997, for a similar incident.)

April 1, 1997 - PECO filed its Restructuring Plan with the PUC and asked to recover $6.8 billion in "uneconomical", stranded costs. The initial proceeding will deal with a request for $3.7 billion. (See April 14, May 22 and June 18, 1997, for more information.)

April 10, 1997 - Unit 3 was operating at 100% power when "the B recirculation pump tripped unexpectedly due to a fault to ground the power cabling to the motor generator set." (IR 50-277/97-02 & 50-278/97-02.)

April 14, 1997 - "PECO entered a two hour TS actions (TSA)...for loss of the C reactor feed pump (RFP) high water level trip capability on Unit 3 due to the discovery of a blown fuse. The blown fuse made the trip function, required TS 3.3.2, inoperable." (IR 50-277/97-02 & 50-278/97-02.)

April 14, 1997 - Administrative Law Judge Louis Cocheres issued a decision stating PECO was not entitled to recoup and "stranded assets" primarily associated with its nuclear generating stations at Limerick and Peach Bottom. (Associated Press, April 14, 1997.) ((See April 1, May 22 and June 18, 1997 for more information.)

April 15, 1997 - A high pressure water service system leak developed at Unit 3. "The size of the hole was determined to be about 2 mm in diameter, and the leak rate was less than 1 gallon per minute." (IR 50-277/97-02 & 50-278/97-02.)- May 7, 1997 - A follow-up Inspection dealing with violations identified by
the NRC on February 10, 1997, found that PECO failed to provide data:
During the telephone discussion we conveyed several concerns with the [PECO's] response. Principally, the discussion of reasons for the violations did not clearly identify root or proximate causes. Accordingly, we could not conclude that corrective actions you

specified effectively addressed the cause of the violation.
Additionally, your response indicated that your safety evaluation was based on the premise that the Turbine Building was maintained at a negative pressure so that air would not be expected to be released through the penetrations. However, no information was provided as to why the Turbine Building was not maintained at a negative pressure, as presumed by your safety evaluation. Further, no commitment was made to document and report your estimate of the unmonitored release...
(James T. Wiggins, NRC, Director, Division of Reactor Safety.)

May 9, 1997 - PECO entered into an agreement with Delmarva Power & Light Company and Public Service Electric and Gas Company (PSE&G) regarding the shut down of the Salem nuclear power plant. “Under the terms of the settlement, PSE&G will pay the Company [PECO] $69.8 million and Delmarva $12.1 million. The settlement also provides that if the current outage exceeds 64 reactor unit months, PSE&G will pay the two companies an additional $1.4 million per reactor unit month, up to an aggregate of $17 million, to be divided proportionately. A reactor unit month is a month during the current outage in which a unit is off-line. (J. F. Paquette, Jr., Chairman of the Board, “Report to Shareholders,” June 1997.)

May 22, 1997 - The PUC ignored the recommendation of Administrative Law Judge Louis Cocheres and allowed PECO to recoup $1.1 billion in stranded investments from customers. As part of Negotiated Settlement worked out between PECO and intervening parties and approved by the PUC, PECO was awarded $5.4 billion in “stranded costs”. (For more information see April 1 & 14 and June 18, 1997.)

June 1997 - “Common stock earnings for the quarter ended March 31, 1997, were $0.49 per share, $0.16 per share lower than the earnings of $0.65 per share for the first quarter of last year...Earnings for the twelve months ended March 31, 1997 were $2.08 per share as compared to $2.64 per share for the corresponding period in 1996.” (J. F. Paquette, Jr., Chairman of the Board, “Report to Shareholders,” June 1997.)- June 4, 1997 - Two violations were identified by the NRC including
failure to full “understand” or “review” the significance of a reactor feed pump trip and temporary scaffolding was located too close to safety-related equipment.

June 5, 1997 - PECO announced it was interested in buying a portion of the 25-year-old Main Yankee nuclear power plant. (Main Yankee was closed by its owners on May 27, 1997. Day-to-day operations were taken over by the Entergy.) Earlier, in the year, PECO offered to purchase Cajun Electric Power Cooperative’s 30% stake in the River Bend (940 MWe) nuclear generating station for $50 million. The Agreement with Cajun was approved by a US Bankruptcy Court on May 29, 1997. (Complied from articles in the Patriot News, June 5 & 23, 1997 and a PECO Press Release, June 5, 1997.) (See September 11 and October 3, 1997 and June 17, 1998, for related developments. Cajun updates can be found on May 27, 1998 and May 27, 2000).

June 18, 1997 - A number of environmental and consumer organizations and Senator Vincent Fumo filed separate appeals to the PUC's May 22 decision allowing PECO to bill customers $1.1 billion in "stranded costs." (PR Newswire, June 18, 1997.) (See April 1 & 14 and May 22, 1997, for background data.)

July 1, 1997 - Two high pressure service water system motor operated valves failed to close.

July 10, 1997 - Problems relating to the Main Control Room Emergency Ventilation radiation monitor were identified by the NRC. (See May 15, 1998, for additional issues and a violation resulting from this deficiency. Also, see September 12, 1997, for a related problem.)

July 17, 1997 - During the SALP evaluation, the NRC found "...there were several instances where operating procedures, surveillances, and tests were not consistent with the design and licensing basis...However, some balance of plant equipment problems challenged operators, indicating continued attention to equipment performance is needed. Also, we found problems with the development and management oversight of efforts to implement the maintenance rule program." (Hubert J. Miller, NRC, Regional Administrator, Jul y 17, 1997.)- July 24, 1997 - The NRC found: "...in one instance, an operator
installing a jumper caused the loss of high pressure coolant injection automatic initiation capability for a short period of time. Our review of the issue found procedural guidance provided to the operator was lacking, in that, it did not specify how to install the jumper or precautions on possible problems that could occur. Maintenance personnel performed, well...However, in one instance a single control rod scrammed due to maintenance technicians pulling the wrong fuses during electrical isolation....Your evaluation and control of non-routine effluent/material release paths, such as sampling and analysis of sewage solids and burning of slightly contaminated oil, showed some weaknesses, indicating a need for further attention in this area....Based on the results of this inspection, the NRC has determined that a violation of NRC requirements occurred...This violation is of concern because several grand master keys were not properly controlled." (Paul D. Swetland, Acting Chief, Projects Branch 4, Division of Reactor Projects, July 24, 1997.)

August 14, 1997 - "...during surveillance testing, the diesel driven fire pump starting battery exploded shortly after the start of the pump. Operators immediately shut down the the pump and notified supervision...Plant management initiated a full root cause investigation for this event. Initial reviews by the investigation team determined that on June 25, predictive maintenance personnel had identified uneven battery electrolyte heating. Also, a separate action request had identified higher than normal current on the battery charger. maintenance recognized that the combination of high current and uneven heating was an indication of cell failure; however, no action was taken to accelerate the scheduled replacement of the battery. Further investigation revealed that the battery cables had a low resistance to ground , which could contribute to the premature failure of the battery. The diesel driven pump uses stranded 24 Volt truck batteries." (IR 50-277/97-06 & 50-278/97-0 6 .)

August 28, 1997 - At Unit-2, "operators experienced trips of the two running drywell chillers, resulting in a loss of drywell cooling for a period of several minutes." (IR 50-277/97-06 & 50-278/97-06.)

August 29 and 30, 1997 - At Unit-2, "power was reduced to 90% for work on a condensate demineralizer." (IR 50-277/97-06 & 50-278/97-06.)

September 1997 - "Earnings for the six months ended June 30, 1997 were $1.02 per share as compared to $1.08 per share for the corresponding period in 1996." (Report to Shareholders, C.A. McNeill, Jr., Chairman, and CEO.)

September 2, 1997 - At Unit-2, "a fire occurred in the 3B circulating water pump motor." (IR 50-277/97-06 & 50-278/97-06.) (See November 6, 1995 and January 14, 1998 for related incidents.)- September 11, 1997 - "PECO Energy Company (NYSE: PE), of
Philadelphia, and British Energy, of Edinburgh, Scotland, announced today formation of a joint venture, AmerGen Energy Company, LLC, to pursue opportunities to acquire and operate nuclear generating plants in the United States." (Company Press release.) (See June 5 and October 3, 1997 and May 27, July 17, 1998, June 25, 1999, and June 9, 2000, for related developments.)

September 12, 1997 - A Notice of Violation was issued dealing with PECO's "troubleshooting of the main control radiation monitor, during which and communication weaknesses led to a noncompliance with technical specifications...in a few instances, your staff did not formally review issues with potential for learning opportunities. Examples included the missing E-2 emergency diesel generator exhaust gasket, and inconsistencies between plant procedures and technical specifications associated with emergency diesel generator starting air reservoir pressure." (Clifford J. Anderson, NRC, Chief Projects Branch 4, Division of Reactor Projects.) (See July 10, 1997 and May 15, 1998, for related problems.)

September 12, 1997 -At Unit-2, "power was reduced to approximately 60% power for hydraulic control unit maintenance and condenser waterbox cleaning." (See November 18, 1994; July 16, September 10 and October 25, 1996; and , September 12, 1997 for related incidents.) (IR 50-277/97-06 & 50-278/97-06.)

September 12, 1997 - At Unit-2, "workers identified a minor leak in the HPSW [High Pressure Service Water] monitoring system caused by a slightly opened instrument valve and a missing threaded cap." (IR 50-277/97-07 & 50-278/97-06.)

October 3, 1997 - The Financial Times of London identified PECO Energy Company as making a bid to purchase Three Mile Island from GPU Nuclear. Due to a confidentiality agreement, GPUN would not confirm the name of the company interested in purchasing TMI. (See July 5 and September 11, 1997 and June 17, 1998 for related developments.)

October 8, 1997 - "Enron Corp. is seeking to takeover PECO Energy Co.'s Pennsylvania service area, offering to lower customers' electric rates by 20 percent and assume $5.5 billion in Peco costs." Patriot News, October 8, 1997. (See November 28, 2001, for a related development.)- October 15, 1997 - "We noted during this period two examples where
personnel either failed to follow procedures or failed to take adequate selfchecking measures, resulting in one case in the conduct of a surveillance test on
the wrong unit. Moreover, two days after this inspection period ended, your staff identified an event inn which a safety-related high pressure service water (HPSW) pump was electrically uncoupled without being isolated because contractor personnel thought they were working on a non-safety-related service water pump that was electrically isolated. This event highlighted weaknesses in procedural adherence, particularly in the use of work package documentation at the job site, self-checking, and a questioning attitude that led to multiple breaches in work process barriers.
"The HPSW event is of particular concern since it impacted a safetyrelated piece of equipment. It also represented the third significant
industrial safety event since late February at Peach Bottom, (bold faced
added), the other two being the unexpected start of a cooling tower fan while a worker was preparing to take an oil sample from the fan gear box, and the injection of chlorinated water into a circulating bay while two workers were conducting a pump inspection. (See December 16, 1997 for a related HPSW incident.) Management's attention to effectively correcting the work clearance process and worker performance weaknesses noted in these events is warranted, particularly given the increase in the number of work activities and contract workers during the Unit 3 outage." (NRC, Clifford J. Anderson, Chief Projects Branch 4, Division of Reactor Projects.)

October 15, 1997 - "A discovery of a licensee operating their facility in a manner contrary to the Updated Final Safety Analysis Report (UFSAR) description highlighted the need for a special focused review that compares plant practices, procedures and/or parameters to the UFSAR. description. While performing the inspections discussed this report, the inspector reviewed the application portions of the UFSAR that related to areas inspected. The inspector verified that the UFSAR wording was consistent with the observed plant practices, procedure and/or parameters. (IR 50-277/97-06 & 50-278/97-06.)

October 20, 1997 - The potential for the suppression pool to be bypassed during a loss-of-coolant-accident at Unit-1 & Unit-2 was identified. PECO identified this event (#33121) as an "outside design basis" incident. (See August, 1999, for more information.

October 29, 1997 - At Unit 3, PECO identified a temperature differential of 84 degrees F. "RPV [Reactor Pressure Vessel] coolant temperature was 163 degrees F with the 'B' recirculation loop temperature at 79 degrees F. (IR 50-277/98-06; 50-278/98-06; NOV.) (See March 23, 1998, for related problems and a Notice of Violation.)- November 1, 1997 - A failure to trip at Unit-2 involving the Reactor
Feedwater Pump Turbine, "was originally attributed to intermittent mechanical binding of some trip mechanism sub components." (IR 50-277/98-03; 50-278/98-03.)
(See April 1, 1997, for a related incident.)

November 7, 1997 - "PECO Energy of Philadelphia had the highest number of justified consumer complaints in 1996 among electric utilities, as well as the longest response time to those complaints [Pennsylvania Public Utility Commission]." (Patriot News, November 7, 1997, B7.)

November 9, 1997 - The unit 2 reactor scrammed. (See December 6, 1997, for root causes of scram.)

November 28, 1997 - Unit 3 was shut down to replace the 'E' steam relief valve.

December 1997 - "Earnings for the nine months ended September 30, 1997 were $1.71 per share as compared to $1.73 per share for the corresponding period in 1996." (PECO Energy, Report to Shareholders, Third Quarter 1997, C.A. McNeill, Jr., Chairman, President and CEO.)

December 16, 1997 - Following an NRC inspection, the staff reported, "...the practice of permitting blanket approvals for overtime work on safetyrelated activities for multiple weeks with no hourly limit specified resulted in abuses that were considered a breach in the intent of the overtime authorization process." (02.3) (Executive Summary.)
Although the Agreement between PECO and the Commonwealth expired in 1993, Section 5.4 established "restrictions on the use of overtime for plant personnel who perform safety-related functions." (June 1989.)

December 16, 1997 - During an NRC inspection, the staff observed: "... findings by your staff late in the Unit-3 refueling outage regarding the existence of cracking of three of the ten recirculation riser pump elbow welds posed a noteworthy challenge to your engineering organization and resulted in the development of a plant operating strategy that limited recirculation flow until a mid cycle outage can be performed in 1998.

Continued on the following page..."Multiple examples of a violation of NRC requirements were identified during this period. Specifically, three examples of a failure to follow procedures were identified, two in the Operations area and one in the Maintenance area. We are concerned with these examples of procedure non-adherence given their impact on plant equipment and their potential industrial safety implications (i.e., one which directly caused a Unit 2 reactor scram [November 9, 1997 at 100% power] and another which significantly contributed to maintenance personnel inadvertently rendering a safety-related HPSW [high pressure service water] pump inoperable [September 22, 1997] without it being electrically isolated during the conduct of work.) (See October 15, 1997 for a related HPSW event.)
"This violation is cited in detail in the enclosed Notice of Violation and the circumstances are described in detail in the enclosed inspection report." (NRC, Clifford J. Anderson, Chief, Projects Branch 4, Division of Reactor Projects.)

December 23, 1997 - "...Unit 2 was shut down to replace the secondary pressure amplifier card and the potentiometer assemblies on the pressure control unit fro the 'B' EHC [electro-hydraulic control] regulator." (IR 50-277/97-08 & 50-278/97-08.) (See December 29, 1997 for a related incident.)

December 23, 1997 - "...plant management chose to shut down Unit 2 due to problems with the pressure regulator control circuit. On December 15, the back up EHC [electro-hydraulic control] pressure regulator 'B' took control of reactor pressure without operator action." (IR 50-277/97-08 & 50-278/97-08.)
- December 29, 1997 - "...all nine bypass valves unexpectedly opened at 155 psig EHC [electro-hydraulic control] pressure during the normal depressurization/cool down of Unit 2. Operations and engineering personnel failed to understand the effect of the EHC system of a temporary plant alteration...This lack of system understanding contributed to all bypass valves unexpectedly opening which resulted in a reactor vessel level transient." (IR 50-277/97-08 & 50-278/97-08.)

December 29, 1997 - "...Unit 2 was shut down to replace amplifier card and potentiometer assemblies." (IR 50-278/97-08; 50-277/97-08.) (See December 23, 1997 for a related incident.)- January 1, 1998 - "... the Unit 2 main turbine tripped on main oil pump low pressure during plant start-up after the turbine rolled to a speed of 1400 RPM. Operations personnel were unaware that the turbine had been rolling for

over two hours just prior to the trip. This issue appeared to involve a failure of an instrument and control test document to restore the original [electro-hydraulic control] EHC [electro-hydraulic control] system alignment after testing and the failure of operations personnel to fully follow procedures. Concerns were also identified with the pulling of control rods to increase reactor pressure during this event and failure of operations personnel to recognize status of the main turbine or turbine control systems." (IR 50-277/97-08 & 50-278/97-08.)
"Several examples of weak control room oversight of activities were noted from the Unit 2 main turbine trip during start-up on January 1, 1998...1) The Control Room Supervisor directed the pulling of control rods to increase reactor coolant system pressure while the turbine condition remained known. 2) Shift turnover and the shift meeting occurred while the turbine was in this unknown condition even though members of the crew knew that the turbine had come off of the turning gear. 3) The crew with the watch during most of this event had not received any just-in -time training such as simulator runs even though this was the first reactor start-up for the Plant Reactor Operator and the Control Room Supervisor." (IR 50-277/98-01, 50-278/98-01.)

January 2, 1998 - "... the unit 2 reactor operator failed to perform the technical specification (TS) surveillance requirements (SR) for verification of proper flow in the recirculation loops. The recirculation loops were not operated outside of the TS requirements during this period. However, it was unclear how station personnel determined the formal TR SRs were met and why operations personnel failed to review the TSs when unclear information was found in the surveillance test." (IR 50-277/97-08 & 50-278/97-08.) These actions violated SR requirements.

January 2, 1998 - Operations personnel failed to take or record the readings for the Surveillance Test for "Daily Jet Pump Operability."

January 3, 1998 - "...operations personnel discovered that the Unit 2 reactor operator (RO) failed to perform the technical specification (TS) surveillance requirement for verification of proper flow in the recirculation loops following start-up" (IR 50-277/99-01; 50-278/99-01.)- January 4, 1998 - "...the main steam line bypass, BPV-1, unexpectedly
opened approximately 25% several times while the Unit 2 reactor was raising reactor power from 96% to 100%. Instrument and control room technicians unknowingly introduced sped error bias in the speed control portion of the EHC [electro-hydraulic control] system after they tightened a loose connection during replacement activities for the EHC pressure control unit. Instrument and control personnel failed to understand what effect tightening the loose connection on the speed control would have on the speed bias signal and EHC system." (IR 50-277/97-08 & 50-278/97-08.)

January 5, 1998 - "...during maintenance on the 2 'C' RHR heat exchanger, technicians found broken glass, an electrical extension cord, and

metal straps on the RHR (shell) side of the heat exchanger. Technicians removed the glass but were unable to remove the cord and metal straps.
After further investigation, PECO determined that the foreign material had been previously identified in the heat exchanger in 1994." (IR 50-277/97-08 & 50-278/97-08.)

January 5, 1998 - "Illinois Power said Monday it contracted an outside nuclear team from PECO Energy Co to manage its Clinton Power Station, which has been shut down since September 1996...Clinton is a 950-megawatt boiling water reactor. Water McFarland, vice president of PECO's Limerick Station, is Illinois Power's new chief nuclear officer. He assumes responsibilities immediately." (R e u t e r s, January 5, 1998.)
"Under the three-year contract, which may be renewed for an additional five years, a core group of PECO Nuclear employees will provide management expertise to Illinois Power." (PECO Energy, 1997 Annual Report, February 2, 1998, p. 4.)

January 12, 1998 - "While transferring a contaminated filter from the spent fuel pool to a shipping cask on January 12, 1998, an area radiation monitor (ARM) alarmed at 20 millirem per hour. Personnel working in the area moved to lower dose areas with the exception of the radiation technician and the overhead crane operator on the bridge. The radiation technician was monitoring radiation levels and informed the operator levels had not significantly changed." (IR 50-277/99-01, 50-278/99-01.)

January 14, 1998 - At Unit 2, "power was reduced to 97% when condenser vacuum decreased after the 2 'C' circulating water pump failed to start and the pump discharge valve failed [to] open during post-maintenance testing." (50-277/97-08 & 50-278/97-08.) (See November 6, 1995 and September 2, 1997, for related incidents.)- January 28, 1998 - "The practice of the control room supervisor leaving
the main control room work station for brief periods without temporary relief from another senior reactor operator demonstrated weak oversight of control room activities.
"On January 28, 1998, the control room supervisor left the main control room work station without temporary relief for several minutes to verify acknowledgment of an expected alarm." The NRC identified a violation of technical specifications. (IR 50-277/98-01, 50-278/98-01.)
"...the NRC identified that a control room supervisor did not visually verify or verbally communicate alarm acknowledgment of an expected alarm that came in on Unit 3 because he was outside his designated work station without temporary relief."
(Severity Level IV violation, IR NOS. 50-277/98-01 AND 50-278/98-01.)

January 29, 1998 - "On January 26, 1998, PECO Energy's Board of Directors voted to reduce the Company's quarterly common stock dividend from

45 cents per share to 25 cents per share, effective with the first quarter dividend, payable on March 31, 1998 to shareholders of record on February 20, 1998. This is a result of the Pennsylvania Public Utility Commission (PUC) orders issued in December and January...

January 30-31, 1998 - "...operators reduced power to about 93% to allow for repairs of the 2C circulating pump discharge valve." (IR 50-277/98-01, 50-278/98-01.)

February 6, 1998 - At Unit 2, "power was reduced to about 90% to investigate trip problems with the 2A reactor feed pump turbine." (IR 50-277/98-01, 50-278/98-01.)

February 13, 1998 - "Unit 3 began the period operating at 94% power. This unit was operating at less than full power due to recirculation system flow rate limitations because of weld cracks on the jet pump risers. On February 13, power was increased to 100%, as allowed by the operating strategy for the jet pump riser cracks." (See March 6, 1998 for follow-up incident.) (IR 50-277/98-01, 50-278/98-01.)

March, 1998 - "The Company reported a net loss for 1997 of $1.5 billion or $6.80 per share. Included in these results was an extraordinary charge of $3.1 billion ($1.8 billion net of taxes), or $8.24 per share, in the fourth quarter to reflect the effects of the December 1997 PUC order (as revised in January 1998) in the Company's restructuring proceeding." (Report to Shareholders, C.A. McNeill, Jr., Chairman, President and CEO, PECO Energy.)- March 1998 - "PECO personnel identified that five Fire Areas in the plant, containing 25 rooms, did not contain automatic fire detection systems...PECO intends to submit an exemption request...for the identified Fire Areas." (IR 50-277/98-10, 50-278/98-10; NOV.)

March 6, 1998 - Power at Unit 3 was reduced to 94%.

March 11, 1998 - PECO Energy Company announced it was counter suing Great Bay Power Corporation "to prevent it from ending a power marketing agreement.

"PECO, which is seeking more than five million in damages for breach of contract and for the loss of goodwill and harm to its reputation, filed the suit in the U.S. District Court of New Hampshire.

"This suit comes a week after Great Bay sought to end the exclusive marketing agreement to sell Great Bay power generated at the Seabrook 1 Nuclear Power Plant in Seabrook, N.H. [Great Bay owns 12.1% of Seabrook.]

"Great Bay also sued PECO last week for breach of contract, charging PECO entered into a number of wholesale agreements in its own name without telling Great Bay or submitting bids on behalf of Great Bay and that PECO 'failed to offer Great Bay's power to customers as required under the marketing agreement' "

(Re u t e r s, March 11, 6:07 Eastern Time.)
June 3, 1998- Great Bay Power Corporation withdrew its lawsuit against PECO. John A. Tillinghast, Great Bay's Chairman said, "We believe PECO acted properly as our marketing agent. And seems clear that the judge in our case is inclined to find that PECO did not breach the marketing agreement....PECO's acceptance of our proposal lets us get started on our own marketing strategy. We appreciate the value PECO has provide Great Bay over the past two years and wish them well in the future." (PECO Energy, Press Release, June 3, 1998.)

March 13, 1998 - Unit 3 was "shutdown for outage 3J12, to perform repairs to the jet pump risers." (Set February 13, 1998 for related information.) (IR 50-277/98-01, 50-278/98-01.)

March 21, 1998 - At Unit-2, "unit load was reduced to perform control rod pattern adjustments, waterbox cleaning, and reactor feed pump turbine testing." (IR 50-277/98-02; 50-278/98-02.)

March 22, 1998 - The NRC noted "reactor engineers did not recommend positive actions to reduce a thermal limit ratio when approaching the Technical Specifications limit, which did not meet operations department expectations for conservative plant operations." (IR 50-277/98-02; 50-278/98-02.)- March 23, 1998 - PECO "identified that they failed to properly implement the improved Technical Specification Surveillance Requirement 3.4.9.4 for the start of the first recirculation pump. Between January 18, 1996, and March 23, 1998, operations personnel were not verifying that the temperature differential between the reactor coolant in the recirculation loop being started and the reactor pressure vessel coolant was within 50 degrees F. On October 27, 1997, the 'B' recirculation pump was started with a differential of 84 degrees F. Although this did not exceed design limits nor impact fuel performance, it was a violation of Technical Specification Surveillance Requirement 3.4.9.4. (Section 08.1). (IR 50-277/98-06; 50-278/98-06; NOV.) (See October 29, 1997, for a precursor event.)

March 25, 1998 - At Unit-3, "foreign material was found in the 3A core spray pump. (IR 50-277/98-02; 50-278/98-02.) (See May 1, 1998 regarding a violation related to this event. (Also, see December 11, 1998, for a related i n c i d e n t .)

March 25, 1998 - A Notice of Violation was issued for cold weather preparations' procedural noncompliances. (IR 50-277/98-11, 50-278/98-11).

March 30, 1998 - "...violations of NRC requirements occurred, namely, (1) the failure to perform certain required tests; and (2) the creation of inaccurate records to indicate that the tests were performed." Charles W. Hehl, NRC, Director, Division of Reactor Projects.)

"... inspectors noted that the control room staff was not aware that maintenance personnel were performing post-maintenance test cycling of vacuum relief valve...during the drywell walkdown. Communications between maintenance and control room personnel were not effective...

"... inspectors noted increased noise in the control room during peak activity periods. During these periods, there were 15 to 20 people in the control room. During these periods order in the control room was challenged. During periods with fewer personnel in the control room and decreased activity, the inspectors observed that operation of the unit became more deliberate." (IR 50-277/98-02; 50-278/98-02.)- March 30, 1998 - A violation was recorded by the NRC form PECO's

failure "during several months to maintain the 2' A' Reactor Feedwater Pump Turbine High Water Level Trip function operable as required by Technical Specification...We concluded during this inspection that your corrective actions for the first two failures were not comprehensive. There were a number of previous opportunities to identify and correct the root cause of these events particularly through at-power verification testing. Also, we noted that the 2' A' feedwater system change of status maintenance to a maintenance rule (a) 1 system was not timely. Although this change met your administrative requirements, we viewed the status change as untimely based on the technical specification significance." (Charles W. Hehl, NRC, Director, Division of Reactor Projects.)

April 16, 1998 - The NRC "observed that the Unit 2' B' stream jet air ejector main steam supply header control room valve...was not in its expected position...This item remains unresolved pending further progress in these investigations..." (IR 50-277/98-02; 50-278/98-02.)

April 27, 1998 - At Unit-2, "unit load was reduced due to an inoperable control rod." (IR 50-277/98-02; 50-278/98-02.)

April 28, 1998 - "The 3A stator water cooling pump tripped during system troubleshooting efforts on April 28, 1998, due to weaknesses both in operations review of the work and with communications regarding restrictions on work scope." (IR 50-277/98-06; 50-278/98-06; NOV.)

May 1, 1998 - "We identified five violations of NRC requirements during this inspection. The first violation involved the failure of a control room supervisor to verify that a Unit 3 expected alarm was acknowledged due to the fact that he was outside of his main control room work station without temporary relief.

"The next two violations were the result of operations personnel failing to perform technical specification surveillance requirements for the verification of proper recirculation loop flow during Unit-2 start-up on January 2, 1998.

"The fourth violation contained several examples of inadequate procedures and control room operators failing to implement operations procedures which

resulted in the unexpected trip of the Unit 2 main turbine on January 1, 1998. The procedures were inadequate since they failed to restore the ElectroHydraulic Control system to the alignment requirement for reactor start-up.
Also, operations personnel failed to adequately implement procedures when they did not recognize the abnormal main turbine status, position of the turbine control valves, or the selection of the speed set for the EHC system for several shifts prior to the main turbine trip."We were concerned with the violations described above, especially the
Unit 2 main turbine trip, because they all showed weak oversight of the control room activities. We previously documented in Inspection Report 50-277 (278)/97-07 where inadequate oversight of operator activities contributed to a scram of the Unit 2 reactor during swapping of a station battery charger.
"The last violation resulted from Unit 3 exceeding the licensed power level up to 0.6% between October 22, 1995 and January 21, 1997. PECO Energy Company operated the reactor at a steady state power level up to 100.6% of rated power. We were concerned that your staff failed to recognize errors in the calibration of feedwater temperature instruments even after deficiencies were identified with the equipment used to calibrate these instruments. The inaccurate feedwater temperature instruments resulted in power levels above the licensed limit for over 15 months." (NRC, Clifford J. Anderson, Chief, Projects Branch 4, Division of Reactor Projects.)
Two "apparent violations" were identified during a special NRC inspection report.
"These violations resulted from: 1) the failure to prescribe and accomplish the ECCS [emergency core cooling system] strainer replacement modification with documented instructions and procedures appropriate to the circumstances to prevent the introduction of foreign materials into the core spray system, and 2) the failure to maintain the 3A core spray pump operable as required..." [See March 25, 1998, for information on the 3A core spray incident.] (NRC, Charles W. Hehl, Director, Division of Reactor Projects.)

May 5, 1998 - "...during testing, operators observed candle-sized flames on the E2 EDG exhaust manifold." (IR 50-277/98-06; 50-278/98-06; NOV.) (See June 9, 1998, for a related incident.)

May 12, 1998 - At Unit 2, "unit load was reduced to withdraw a control rod following repairs to one its scram solenoid pilot valves." (IR 50-277/98-06; 50-278/98-06; NOV.) (See June 1, 1998, for a related incident, and March 22, 2000, for a similar challenge).

May 14, 1998 - "Four licensed operators missed training for the two year requalification period that ended in March 1996 and never made up the missed training within a reasonable time thereafter. This was unresolved pending NRC staff review for enforcement action with respect to 10 CFR 55.59 a (1). (IR 50-277/98-04; 50-278/98-04 and NOV.)- May 14, 1998 - The NRC identified two violations relating to licensee operator requalification training (LORT). “The first violation involved a failure to assure sufficient differences in the job performance measure (JPM) portion of the operating test administered to different crews on different weeks. This violation is of concern because of the potential for precluding the identification of retraining needs. The second violation involves the failure of your operating test to evaluate SROs [senior reactor operators] fulfilling the role of the control room supervisor in their ability to execute the emergency plan. This violation is of concern since the SROs may be called upon to execute the plan in the absence of shift managers.” (IR 50-277/98-04; 50-278/98-04.)

May 14, 1998 - The NRC identified a violation “for failure to include the area of radiation monitoring system within scope of the maintenance rule program...This violation is of concern since scoping problems of this type have been identified through recent operating experience and findings from NRC maintenance rule baseline inspections and the violation represents an apparent failure to incorporate this information into your program.” (IR 50-277/98-04; 50-278/98-04; and NOV.)

May 15, 1998 - “...operations personnel identified that the trip relay for the Main Control Room Emergency Ventilation (MCREV) radiation monitor had not been in the tripped status for approximately 28 hours while the ‘B’ channel radiation monitor was inoperable.” This was a violation of the technical specifications.
“The operations personnel installing the jumper to initiate a Division II isolation trip of the MCREV radiation monitor did not perform, nor did the procedure instruction require, a positive verification that the trip was properly inserted. The corrective actions from the July 10, 1997 event were not comprehensive enough to prevent the subsequent event. (Section 02.1). (IR 50-277/98-06; 50-278; 98-06; NOV.) (Also see September 12, 1997; June 7 & July 17, 1998 for related problems.)

May 16, 1998 - “During a Unit 2 power down evolution on May 16, 1998, operators reduced speed on an incorrect reactor feed pump, resulting in a reactor level excursion and recirculation system runback. The event was indicative of poor operator performance, reflecting weaknesses in communications, self-checking, and peer/supervisory review.” (IR 50-277/98-06; 50-278/98-06; NOV.) (See related incidents on March 17, 199;, March 4, 1996; June 7 and July 13, 1998.) - May 19, 1998 - The NRC issued a “confirmatory order modifying the
license of Peach Bottom Units No. 2 and No. 3 requiring that the Company complete final implementation of corrective actions on the Thermo-Lag 330 issue by completion of the October 1999 refueling of Peach Bottom Unit No. 3”. (PECO Energy Company, Form-10/K-A, p. 10). (See September 12, 1994, October 1, 1996, October 12, 1999, and July 21, 2000, for background information.)

May 22, 1998 - Unit power was reduced at Unit 2 for condenser waterbox cleaning.

May 27, 1998 - "The U.S. Justice Department on Wednesday said it sued Philadelphia-based PECO Energy Co (PE - news) for more than $67 million in damages because the company allegedly reneged on an agreement to buy a share [30% interest in the River Bend nuclear power plant owned by Cajun Electric Power Cooperative, Inc.] of a Louisiana nuclear power plant." (Reuters, Wednesday May 27, 1998, 7:55 pm, Eastern Time.) (See June 5, September 11, and October 3, 1997 and May 27 and June 17, 1998 for background information and related developments). (Cajun update can be found on May 27, 2000).

May 29, 1998 - At Unit 3, "unit load was reduced to clean condenser water boxes." (IR 50-277/98-06; 50-278/98-06; NOV.)

June 1, 1998 - At Unit 2, "unit load was reduced following a scram of a control rod during reactor protection system testing. The control rod had a leaking scram solenoid pilot valve. The unit power was reduced on June 5 to facilitate control rod hydraulic control unit (HCU) on-line maintenance to replace several scram solenoid pilot valves." (IR 50-277/98-06; 50-278/98-06; NOV.) (See May 12, 1998, for a precursor event.)

June 7, 1998 - "...the 3A recirculation pump ran back to 30% speed due to the unexpected loss of a 500 kv line during an electrical storm and the slow opening of the 500 kv breaker. The 3B recirculation pump remained at full speed during this event. Due to the difference in pump speeds of the Unit 3 pumps, the flows in the recirculation loops were significantly mismatched. The recirculation loop flows remained mismatched outside of Technical Specification Surveillance Requirement (SR) 3.4.1.1 for over 12 hours." This was a another violation of Technical Specifications. (IR 50-277/98-06; 50-278/98-06; NOV.) (See May 16 and July 13, 1998, for related incidents.)
Continued on the following page..."Engineering personnel failed to recognize the potential for high vibration
stresses on the 'A' jet pump loops due to the large recirculation flow mismatch following the 3A recirculation pump runback on June 7, 1998. The potential for recirculation flow mismatch to cause excessive vibration of the jet pumps and the jet pump riser braces was described in the Peach Bottom Design Basis Document (DBD) for the recirculation system. This lack of understanding of the effects of this mismatch contributed to the failure of engineering personnel to provide the necessary technical information to operations personnel...
" Also, Unit 3 experienced a runback of the 3A pump in December 1993 due to the loss of power to the same relay that dropped out during this event. Part of the corrective action for this event was to install a modification which would provide a non-interruptible power supply to the recirculation pump runback relays. This corrective action, which could have prevented the 3A

runback on June 7, was never performed. (Section E1.1). (IR 50-277/98-06; 50-278/98-06; NOV.) (Also, see March 17, 1995 and March 4, 1996 for related events.)

June 8, 1998 - "... the 3 start-up transfer became inoperable following a severe electrical storm, but this was not recognized by operators until June 22, 1998. On June 15, the inoperable 3 start-up transformer was aligned to the 2 start-up and emergency source for over nine hours to support off-site maintenance work." The NRC "treated" this event as a Non-Cited Violation. (IR 50-277/98-07, 50-278/98-07.)
An LER (96-005) issued on May 7, 1996, identified a similar problem.

June 9, 1998 - The NRC identified two violations during an inspection. "The first violation involved a high pressure coolant injection (HPCI) system operating procedure [discovered by the NRC on March 22, 1998] that did not provide adequate instructions regrading the HPCI pump turbine vibration monitoring system. The second violation was the failure of health physics personnel to follow radiation area control procedures regrading posting of an open door to a potentially high radiation area.
"We are also concerned about a number of instances of plant valves being identified out of their required or expected position. Although several of these valves were in non-safety related systems, three valves were in safety related systems. We determined that, taken collectively, these items represented a weakness in plant status control." (Clifford J. Anderson, Chief, Projects Branch 4, NRC, Division of Reactor Projects.)- June 9, 1998 - "...plant personnel and the inspectors observed smoking and small flames on the E1 EDG exhaust manifold flanges, and the oil occasionally flashed and self-extinguished as the temperature of the exhaust manifold increased during EDG loading. The smoking and leakage essentially stopped several minutes after the EDGs were fully loaded." (See May 5, 1998, for a precursor event.)
"Some emergency diesel generator (EDG) oil leak reduction strategies were not well-implemented or well-communicated to operations personnel. These factors contributed to oil leaks and flames observed on the E2 and E1 EDG exhaust manifolds in May and June, 1998, respectively." (IR 50-277/98-06; 50-278/98-06; NOV.)

June 12, 1998 - The NRC proposed a $55,000 fine for PECO for two program deficiencies that led to the impaired performance of a Unit 3 emergency cooling pump...The violations were identified during NRC inspections conducted between February 12 and March 3 and from March 30 to April 24 [1998]...Specifically, the violations stem from problems that affected a Unit 3 core spray pump. The component is part of the unit's core spray system, which would be used to keep the reactor core covered and cooled during a loss-of-coolant accident." US NRC, Office of Public Affairs, Region I, King of Prussia, PA, June 12, 1998.) Continued on the following page...

(For more detailed information on these problems, see NOTICE OF VIOLATION AND PROPOSED IMPOSITION OF CIVIL PENALTY - $55,000, June 11, 1998, NRC INSPECTION REPORT NOS. 50-277/98-03 & 50-278/98-06.)

June 22, 1998 - "...a reactor building equipment operator discovered during routine operator rounds that the Unit-3 reactor core isolation cooling system mechanical over speed trip tappet was not fully reset. Station personnel determined that the reactor core isolation cooling system had been inoperable since May 4, 1998 which was the last time the over speed trip function was manipulated and successfully tested." (IR 50-277/98-07, 50-278/98-07.) The NRC "treated" this incident as a Non-Cited Violation.

July 9-10, 1998 - The NRC observed "instrument and plant control personnel failed to comply with the technical specification action time requirements fro placing ; 'A' channel of the main control room emergency ventilation (MCREV) system in trip within six hours of making the channel inoperable...This non-reporting, licensee identified and corrected violation is being treated as a Non-Cited Violation..." (IR 50-277/98-02, 50-278/98-02.)

July 10-11, 1998 - Power was reduced to about 60% at Unit-2 for condenser waterbox cleaning.

July 11, 1998 - Unit load was reduced to 74% at Unit-3 for main steam isolation valve testing.- July 13, 1998 - "A reactor level water excursion on July 13, 1998,
during transfer between feedwater control system computers revealed that instrument and control personnel did not have sufficiently specific written guidance or criteria on computer signal differences for performing the computer transfer. Instrument and control personnel relied on inappropriate assumptions on acceptable computer signal differences." (IR 50-277/98-07, 50-278/98-07.) (See May 16 and June 7, 1998, for related incidents.)

July 17, 1998 - AmerGen Energy announced that it reached an agreement with GPU to purchase TMI-1 for $100 million. The proposed sale includes $23 million for the reactor, and $77 million, payable over five years, for TMI-1's nuclear fuel. (Background information can be found on: September 5 & 11 and October 3, 1997, and May 5 & 27, 1998.)

July 17, 1998 - "...the 2A condensate pump had to be shutdown quickly due to rapidly climbing temperatures on the thrust bearing." (IR 50-277/98-07, 50-278/98-07.)

July 22, 1998 - "... hydrogen water chemistry injection into the unit 2 feedwater system unexpectedly isolated during application of a clearance for the 2A reactor feedwater pump." (IR 50-277/98-07, 50-278/98-07.)

August 6-19, 1998 - During a walkdown, the NRC determined “that the actual wiring did not match the schematic drawings. Although the schematics showed that the wiring for the MOVs [motor operated valves] on both units were the same, the as-found did not match the schematic drawings for 3 CS suction MOVs.” (IR 50-277/98-08, 50-278/98-08.)
“PECO experienced three failures of motor operated valves (MOVs) during 2R12. One other MOV was in a significantly degraded condition when inspected. All of these MOVs were safety-related.” (IR 50-277/98-10; 50-278/98-10; NOV.) (See January 21, 1993, for a related incident.)
- August 10, 1998 - During the calibration of the ‘C’ detector, the [chemistry] technicians inadvertently removed and dropped the “D’ detector. The technicians performing this work did not stop and notify the control room operations personnel or Chemistry Supervision that they had removed the “D” detector and dropped it...The behavior of the technicians to not tell details about the event for several days, and only when asked, was not acceptable. The licensee corrective actions were narrowly focused on the chemistry department and did not include the other departments at the station. Procedural nonadherence has been an issue at the station for the past year.” (IR 50-277/98-10, 50-278/98-10.)
The NRC issued a Violation.- August 12, 19, and 24, 1998 - Access and alarm failures to protected
areas and vital door areas occurred as a result of failures with the #1 security multiplexer. (IR 50-277/98-08, 50-278/98-08.)

August 14, 1998 - At Unit-3, a loss of service water to a main generator hydrogen cooler resulted in a reduction of unit load to 84%.

August 19, 1998 - at Unit-3, “Operators entered the ‘B’ non-regenerative heat exchanger room and found the heat exchanger vent valves partially open, instead of closed, as required. Upon further investigation, operations personnel identified that these valves were left out of position due to poor configuration control of the system while preparing for maintenance activities.” (IR 50-277/98-08, 50-278/98-08.)
A Notice of Violation was issued.

August 20, 1998 - The Reactor Water Cleanup (RWCU) system at Unit-3 was being returned to service, when an automatic isolation “occurred due to a high flow condition.” (IR 50-277/98-08, 50-278/98-08.)
A Notice of Violation was issued.

August 21, 1998 - Unit load was reduced due to a degraded cooling of the 3C main transformer. At Unit 3, “operators commenced a down power maneuver due to cooling of the main transformer. The reduced load prevented a loss of the main transformer and plant transient when the deluge system activated.” (IR 50-277/98-08, 50-278/98-08.)
In other words, “The #6 oil pump had failed due to a burnt wire and when

then operator, following the alarm response card, switched the local control to manual, all of the cooling fans and oil pumps tripped off."

August 22, 1998 - An operator "inadvertently shutdown the 3C drywell chiller. (IR 50-277/98-08, 50-278/98-08.) The NRC concluded, "An engineering evaluation for a similar event that occurred on March 25, 1997, was not effective to preclude the August 22, 1998 event."

August 23, 1998 - "Weaknesses in maintenance planning and work practices led to a significant water leak on the station fire main on August 23, 1998. Water from the leak entered the safety related emergency service water/high pressure service water pump house via underground electrical conduits and degraded penetration seals." (IR 50-277/98-08, 50-278/98-08.)
A Notice of Violation was issued...- August 23, 1998 - "... the motor driven fire pump unexpectedly started
during the post-maintenance testing of the H-1 fire hydrant. Neither the work order or the routine test procedure contained any documentation to inform operators that the motor driven fire pump could staff during the hydrant post maintenance testing nor did these documents contain instructions to fill and vent the fire system after work was performed." (IR 50-277/98-08, 50-278/98-0 8 .)

August 24, 1998 - The torus/drywell vacuum breaker "lost its 'seated ' indication." Six days later, although required by technical specifications, "operations personnel determined that the actions to verify that the vacuum breakers were closed had not been performed..." (IR 50-277/98-08, 50-278/98-08).
The NRC "treated" this problem as a Non-Cited Violation.

September 3, 1998 - In the first eight months of 1998, "PECO has cut its dividend nearly in half, announced 1,200 job cuts, and written off $3.1 billion in assets." (Patriot News, Bu s i n e s s, September 3, 1998. (See June 13, 2001, for more job reductions).

September 15, 1998 - At Unit-2, the reactor water cleanup system automatically isolated. PECO found that this incident was not directly related to an event that occurred on December 1, 1998. (IR 50-278/98-11, 50-278/98-11).

October 6, 1998 - During an alternate decay heat removal test (ADHR), "the inspectors observed the performance of an abnormal operating procedure..." (IR 50-277/98-10, 50-278/98-10; NOV.)

October 12-22, 1998 - Three fuel movement errors occurred during this period. "These errors were caused by a failure to properly verify component location and orientation as required by procedure." The NRC treated this incident as a "no-cited violation." (IR 50-277/98-10, 50-278/98-10; NOV.) (See

October 22 and 24, 1998.)

October 14, 1998 - While restoring the 2B RHR [residual heat removal] subsystem, "operations personnel discovered several hundred gallons of water on the Unit-2 torus room floor. After further investigation, operators discovered that four RHR header vent valves had been left open during the performance of a system fill and vent evolution...The inspectors determined that this event was indicative of on-going challenges at the station in the area of system status and configuration control. Similar issues were cited in Notices of Violation in NRC Inspection Reported 50-277(278)/98-08 and 98-01. The inspector concluded that PECO did not not have sufficient time to fully implemented corrective actions for these previous issues. Therefore, this event was not subject to formal enforcement action." (IR 50-277/98-10, 50-278/98-10; NOV.)
A Notice of Violation was issued...- October 16, 1998 - "...during a routine tour of the reactor building, the
inspectors identified a minor leak on the 2 'D' RHR loop. (IR 50-277/98-10; 50-278/98- 10; NOV.)

October 22, 1998 - "..the refueling floor operators removed a fuel bundle at core location 23-50 (southwest orientation) rather than the the specified 23-52 (southeast orientation.) The LSRO, noting the hole left by the removed fuel bundle, discovered that the wrong bundle had been fully removed for the core." (IR 50-277/98-10; 50-278/98-10; NOV.) (See October 12 and October 24, 1998, for repetitive incidents.)

October 24, 1998 - "...core alterations were suspended for a third time due to a mis-oriented fuel bundle in the spent fuel pool. (IR 50-277/98-10; 50-278/98-10; NOV.) (See October 12 and 22, 1998, for repetitive incidents.)

October 25, 1998 - At unit-3, the "E33 bus was inadvertently tripped during the performance of a surveillance procedure that functionally trip tested E32 and E324 bus over current relays. This resulted in an 'A' channel half scram, a full reactor water clean up isolation, loss of the 'C' standby gas treatment fan, an inboard primary containment isolation system group 3 isolation and subsequent loss of reactor building ventilation, and a half primary containment isolation system group 1 isolation that did not cause any valve motion."
The NRC did not issue any violation. "However, inadequate self-checking and peer checking by the instrument and control technicians performing the surveillance procedure were determined to be the root cause of the event." (IR 50-277/98-10, 50-278/98-10; NOV.)

October 28, 1998 - The NRC identified a violation which "involved the failure of the radiation protection technicians to fully comply with a procedure associated with source checking of instruments used to survey incoming shipments of radioactive material."

Additionally, the NRC noted that there 56 “control room deficiencies” and “critical control room deficiencies” scheduled to be corrected during the most recent refueling outage. (IR 50-277/98-08, 50-278/98-08.)

October 28, 1998 - The use of an improperly sized jumper led to an unplanned core spray loop inoperability and “extended the inoperability period for all four emergency diesel generators (EDG).” (IR 50-277/98-10, 50-278/98-10; NOV.)- November 7, 1998 “...operations personnel in the Unit 2 control room observed that the megawatt electric output did not agree with the reactor core thermal power.” (IR 50-277/98-11, 50-278/98-11.)The NRC “treated” this incident as a Non-Cited Violation. (This was the fifth Non-Cited Violation since June 1998. Please refer to November 30, 1998, and July 27, 1999, for more data on “Non-Cited Violations” .)

November 17, 1998 - “There was one deficiency identified during the November 17, 1998, plume exposure pathway exercise which was resolved on March 16, 1999, during a remedial [emergency preparedness] drill. Also, there were were 27 Areas Requiring Corrective Action (ARCA) identified...” (FEMA Final Exercise Report for the November 17, 1998, Peach Bottom Power Station Plume Exposure Pathway Exercise.)

November 27, 1998 - “...operators shut down Unit 3 to repair a nitrogen leak on an air opened valve inside the drywell.” (See May 11, 2000, for a related incident). (IR 50-277&278/98-11.)

November 30, 1998 - “...inadequacies in a breaker manipulation procedure lead to an unexpected loss of one off-site power source and several emergency safety feature actuations.” (IR 50-277/98-11, 50-278/98-11). The NRC “treated” this incident as a Non-Cited Violation. (This was the sixth NonCited violation since June 1998). (Please refer to November 7, 1998, and April 6 & July 27, 1999, for data on “Non-Cited Violations” .)

December 1, 1998 - The reactor water cleanup system “isolated occurred as operators were opening the system inboard and outboard isolation valves.” According to PECO, his event was not directly related to an incident that occurred at the RWCU on September 15, 1998. (IR 50-277/98-11, 50-278/98-1 1) .

December 6, 1998 - At Unit 3, a control rod worth minimizer rod block occurred during a control rod drift alarm test. (IR 50-277/98-11, 50-278/98-1 1) .

December 11, 1998 - “A fire watch was found asleep in the cable spreading room by inspectors.” (IR 50-277/98-10; 50-278/98-10; NOV.) (See December 18, 1993 and August 4, 1994, for related developments.)

December 11, 1998 - "Contractor personnel performing modification work on the Unit-2 scram air header exhibited poor foreign material control practices, contrary to specific work order instructions. Weaknesses in contractor oversight were identified by these poor practices. (IR 50-277/98-10, 50-278/98-10; NOV.) (See March 25 and May 1, 1998, for related incidents.)- December 19, 1998 - Unit load at Unit 2 "was reduced to 60% (See also January 2, 1999) to repair a leak on the B3 feedwater heater extraction steam line." (IR 50-277/98-11, 50-278/98-11.)

December 27, 1998 - Both Units were at 100% when one (of two) emergency auxiliary transformers failed. This incident precipitated a station blackout and the inoperability of an off-site power source. (IR 50-277/98-11, 50-278/98- 1 1 .)

December 30, 1998 - FEMA's Final Exercise Report For The Spring 1998 identified eight Areas Requiring Corrective Action (ACRA).

December 31, 1998 - PECO reported "a charge of $125 million ($74 million of net income taxes) for its Early Retirement and Separation program relating to 1,157 employees." (PECO Energy Company, Form 10-K/A, 1999, p. 7 7) .

January 2, 1999 - Unit load was reduced again (See December 19, 1998) to 65% to allow repairs to the main steam turbine #3 control valve. (IR 50-279/98-11, 50-278/98-11.) the system inoperable."

January 19, 1999 - "The inspectors reviewed an event in which the Unit 2 HPCI system gland seal condenser lower head gasket developed a significant leak, prompting operators to declare the system inoperable." (IR 50-277/99-01, 5 0 - 2 7 8 / 9 9 - 0 1 .)

January 21, 1999 - "...the station made a four hour non-emergency 10 CFR 50.72 report to the NRC when a damper in the flow path from the Unit 2 reactor building ventilation to the standby gas treatment system (SGTS), failed to open." (IR 50-277/99-01, 50-278/99--01.)

January 29, 1999 - An "outside design basis" event (# 35335) was reported for Unit-2. (See August, 1999, for more information.)

February 1, 1999 - The NRC issued a Violation and stated their " c o n c e r n " :
1) three licensed operators failed to complete your facility licensed operator requalification program for the period April 1994 through March 1996 and the training was not made up until April 1998, in some cases; 2) the failure was due to a program inadequacy (systematic cause) and the inadequacy apparently caused an

inaccurate license renewal application to be submitted to the NRC upon which the NRC issued a renewed operator license. (Curtis J. Cowgill, NRC, Chief, Projects Branch 4, Division of Reactor Projects.)- February 1, 1999 - An NRC inspection team found two examples in which RCIC [reactor core isolation cooling] system design basis information was not properly translated into procedures." A Notice of Violation was issued. (50-277/98-09, 50-278/98-09 & NOV).

February 8, 1999 - During Y2K testing of the Unit-2 rod worth minimizer system, a "seven hour lockup of the plant monitoring system (PMS) computers and interruption of data to PMS-supported systems" occurred. The problem was attributed to "an information systems engineer [who] did not adhere to station policy regarding stopping of testing when unexpected conditions occur." (IR 50-27(278)/99-02.)

February 18, 1999 - During an surveillance test, "the 3 B core spray pump breaker malfunctioned in that it failed to close." (IR 50-277(278)/99-02.)

February 20, 1999 - Unit-2, "unit load was reduced to 60% to facilitate control rod scram time testing, reactor feedwater pump turbine testing, a main steam drain tank valve repair, and a control rod sequence exchange." (IR 50-277(278)/99-02.)

March 25, 1999 - "NRC Inspection Report 50-277 (278)/98-01 cited a violation of the Unit 3 operating license for exceeding the licensed power level by as much as 0.6% for a period of about 18 months. This condition occurred as a result of inaccurately calibrated feedwater temperature instruments." (IR 50-277/99-01, 50-278/99-01.) (See related developments on January 1 and June 4, 1997, and May 1, 1998.)

March 27, 1999 - Unit-2, "unit load was reduced to 62% power to allow condenser waterbox cleaning and reactor feedwater pump turbine work." (50-277(278)/99-02.)

March 3, 1999 - The PUC voted "to give PECO Energy Co. a reproof for running misleading advertisements about electric competition last fall." (Patriot News, March 5, 1999.)

March 3-4, 1999 - Unit -3 was reduced to 92% power for load drop activities and "repair a minor steam leak on the feedwater level switch flange." (50-277/(278)/99-02.)

March 11, 1999 - Documentation of two Security Level IV violations were reported by the NRC: 1) Failure to Energize Trip Relay for Main Control Room Emergency Ventilation; and, 2) Failure to Properly Maintain Procedures for

High Pressure Coolant Injection (HPCI) System Manual Operation.- March 12, 1999 - At unit-3, "RCIC [Reactor Core Isolation Cooling] system isolation occurred during realignment of the system following back seating of an inboard steam isolation valve." (50-277(278)/99-02.)

March 18, 1999 - The potential for a fire from flooding was identified at Units 2 & 3, and classified as an "outside design basis" event. (#35485.) (See August, 1999, for more information.)
In addition, "Between March and October 1998, PECO engineering identified five fire areas, containing cables for safety-related or safe shutdown equipment that did not have automatic fire detections systems as required..." (IR 50-277 & 278/99-05.)

April 6, 1999 - Security staff "detected a disabled a vital door area door alarm in Unit 3. The door alarm function was disabled for approximately six days...This Security Level Violation IV is being treated as a Non-Cited Violation, consistent with Appendix C of the NRC Enforcement Policy. (This was the seventh Non-Cited Violation since June 1998). (See November 30, 1998, for related events.) (NCV-50-278/99-0401)." (IR 50-277/99-04; 50-278/99-04).

April 15, 1999 - A Fitness-for-Duty incident involving controlled substances and three used syringes was reported to the NRC. (See May 10, 1999, for results of laboratory tests.)

April 17, 1999 - "...Unit 3 load was reduced to approximately 83% power for a control rod pattern adjustment and to repair an air leak on a control rod hydraulic control unit." (IR 50-277/99-04; 50-278/99-04).

April 25, 1999 - "...a high temperature alarm (greater than 500 degrees F) was received for the Unit 3 control rod drive (CRD) 26-11." (IR 50-277/9-04; 50-278/99-04).

May 6, 1999 - "During the inspection, the NRC reviewed a violation that your staff identified involving the Unit 2 rod block monitoring system being inoperable for 29 of the 185 control rods. Since this finding involved a Severity Level III Violation of NRC requirements, it could be considered for escalated enforcement including a civil penalty." (Exercise of Enforcement Discretion Related to IR 50-277; 278/99-02.)
"A wiring error dating back to original construction was discovered which resulted in non-conservative inputs to channels of the Unit-2 rod block monitor for 29 of 185 control rods." (Bold face type added.) (50-277(278)/99-02.)- May 6, 1999 - "PECO found a motor brake on the 2'C' RHR [Residual heat Removal] pump torus suction valve that should have been removed during a modification in 1988. The inspectors were concerned that other safety-related MOVs included in the 1988 modification could have motor brakes installed." (Bold faced print added.)

Similar time delayed problems with the 2'C'; RHR occurred on January 5 & August 6-19, 1998. Also, see January 21, 1993 for root cause problems with the 2'C' RHR.

May 10, 1999 - PECO found traces of a controlled substance "in a bathroom inside the protected area" at Peach Bottom. "The results [from a laboratory] indicated the presence of a controlled substance." (IR 50-277/99-04; 50-278/99-04). (For related incidents refer to, November, 1987; January 8, 1988 & February, 1988; and, November, 1989.)

May 15, 1999 - "...Unit 2 load was reduced to approximately 71% for maintenance on an outboard main steam isolation valve."
"...Unit 3 load was reduced to approximately 80% power of a control rod pattern adjustment, then restored to 100% power". (IR 50-277/99-04; 50-278/99-04).

May 25, 1999 - A Unit-3 "reactor operator received a reactor low level alarm and noted that the level was trending downward. The operator took prompt actions in accordance with plant procedures to reduce reactor power and to manually control reactor feed pumps until level had stabilized." (IR 50-277 & 278/99-05.)

June 3, 1999 - Plant personnel identified "the 3B core spray system flow indicator was reading zero flow with the pump running. I&C [Instrumentation and Controls] technicians checked the valve lineup and found the flow transmitter had been improperly left isolated following I&C maintenance the previous day." (IR 50-277 & 278/99-05.)

June 4, 1999 - Load at Unit-2 "was reduced to about 65% power for main condenser waterbox cleaning and various maintenance activities." Power was restored to 100% on June 6, 1999. (IR 50-277 & 278/99-05.)

June 10, 1999 - Plant "operators experienced a temporary loss of the Unit 2 plant monitoring system (PMS) computer. They reduced power slightly to ensure average power limits were not exceeded, since the average power monitoring function of PMS was no longer available." The loss of safety parameter display system, was reported to the NRC (IR 50-277 & 278/99-05.)-
June 11, 1999 - Load was reduced at Unit-3 "to about 65% power for scram time testing and other maintenance activities." Unit-3 achieved full power two days later. (IR 50-277 & 278/99-05.)

June 24, 1999 - Plant personnel "responded effectively to a Unit 3 RCIC high suction pressure alarm. After the high pressure condition was corrected through the use of the alarm response card, shift personnel continued to monitor the RCIC system for abnormal parameters." (IR 50-277 & 278/99-05.)

June 25, 1999 - Load was reduced at Unit-3 "to about 85% power for a rod pattern adjustment and was returned to full power on June 26." (IR 50-277 & 278/99-05.)

June 25, 1999 - PECO's stock price fell $2.50 on June 17 and 18, 1999 per share "after management warned financial analysts second quarter earnings were trailing expectations.
"During a conference call Thursday discussing AmerGen's agreement to purchase the Nine Mile Point nuclear power plant on Lake Ontario in New York State for $163 million, PECO management said the company will have second quarter operator earnings of about 31 cents a share..." (Reuters, Jim Brumm, June 25, 1999.) (See September 11, 1997, for background data on AmerGen, and refer to May 12, 2000, for collapse of the Agreement).

June 28, 1999 - PECO Nuclear transferred radioactive waste material to Chem Nuclear's waste disposal facility in South Carolina "that was not properly characterized...The issue...is more than minor in that, if left uncorrected, it could become a more significant safety concern because accurate waste characterization is necessary to ensure proper near-surface disposal of radioactive waste materials. The issue affected the Public Radiation Safety cornerstone...this is considered an apparent violation." (05000277 & 278/2000-002). (See April 25 & August 3, 2000, for a related incident).
July to September, 1999 - Power was lost to the 351 line on three separate occasions from July to September 1999 due to storm damage. The loss of the 351 line affects a the station blackout (SBO) line and results in a loss of power to the technical support center (TSC). The loss of power to the TSC results in a loss of emergency assessment capability and, if greater, than an hour, an one hour non-emergency report to the NRC if required....In response, PECO initiated a York County Reliability Enhancement Plan to address immediate reliability issues for the 351 and 341 (a backup supply to the 351) lines..." (IR 05000277/99008, 05000278/99008.) - July 7, 1999 - "...operators observed that the 'A' ESW pump flow rate to
the emergency diesel generators (EDGs) was in the In-Service Test (IOST) alert range specified in the surveillance procedure...Engineering placed the 'A' ESW pump on an increased testing frequency and conducted an investigation into possible causes of the degraded flow." (IR 50-277/99-06; 50-278/99-06; and, 72-1027/99-06).

July 10, 1999 - "...Unit 3 load was reduced to approximately 62% for main condenser tube leak repairs." (IR 50-277/99-06; 50-278/99-06; and, 72-1027/99-06).

July 13, 1999 - "...Unit 2 load was reduced to approximately 67% power as a result of the trip of the 2B reactor feed pump and subsequent recirculation system runback." (IR 50-277/99-06; 50-278/99-06; and, 72-1027/99-06).

July 15, 1999 - At Unit 3, "operators removed the fifth stage feedwater heaters from service, restoring full power capability." (50-277/99-06; 50-278/99-06; and 72- 1027/99-06) .
July 27, 1999 - The NRC found two Severity Level IV violations during an inspection, but classified the infractions as" (This was the eighth Non-Cited V i o l a t i o n since June 1998. See November 7 and 30, 1998 and April 6, 1999, for other "Non-Cited Violations.").
"The first NCV involved the inadvertent loss of the Unit 3 Auxiliary Transformer and associated fast transfer of four 4KV emergency busses due to inadequate equipment configuration control management by your operating staff [May 21, 1999.] The second NCV involved nonconformances to Peach Bottom Fire Protection Plan which were self-identified by PECO engineering personnel during comprehensive reviews of the Fire Protection Plan." (NRC, Curtis J. Cowgill, Chief, Projects Branch 4, Division of Reactor Projects.)

August, 1999 - "If a utility has operated a reactor outside of the safety parameters established in its operating license, i.e., "outside design basis," it is required to document it in a daily event report filed with the NRC. The more event reports filed by a nuclear eactor, the less certain that the reactor and its safety systems will operate as designed." (James Riccio, Public Citizen, August, 1999, Executive Summary.) (Refer to October 20 1997 & January 29 and March 18, 1999, for specific "outside design basis" events.)- August 4, 1999 - The NRC reviewed senior reactor operator exams:
"A performance deficiency was identified during the performance of a JPM applicant when an applicant, while operating the refueling bridge under the direction of a fuel handling director (FHD), allowed the mast to make contact with the south fuel prep machine handrail. The mast was in the normal up position with no fuel grappled. Although the contact was minor and no damage resulted, the event indicated a lack of oversight on the part of the FHD and inattentiveness on the part of the applicant."

" An exam security problem was identified by PECO involving exam material previously copied by a PECO exam team member and later discovered in the same copy machine by another PECO exam team member.
"The examiner determined based on the time line developed by PECO, through interviews with those involved, and reenactment of the event, that the event was minor and the exam was not compromised." (IR 50-277,278/99-301.)

September 1, 1999 - "...while installing a switch for a Unit 3 refueling outage recirculation pump trip modification, a contractor technician inadvertently repositioned the 3A reactor protection system (RPS) alternate power supply switch. This resulted in a temporary loss of power to the 3As RPS, causing a half scram and ESF actuation." (050277/99008, 05000278/99008.)

September 23, 1999 - Unicom and PECO announced a "merger of equals with" a combined value of $31.8 billion. "The new holding company will be the nation's largest electric utility based on its approximately 5 million customers and it will have total revenues of $12.4 billion." (PECO Energy, Press release, September 23, 1999.) (See (March 24 and April 1, 2000, for related developments.)

September 20, 1999 - "...while increasing the size of a hole in the reactor control panel to support a Unit 3 refueling outage power range instrumentation modification, a contractor technician drilled into a wire to the Unit 3B reactor manual scram circuit. This caused a blown fuse, a half scram, and the resultant ESF." (IR 050277/99008, 05000278/99008.)

September 30, 1999 - A turbine trip, followed by a scram, occurred at Unit 2. "Following the reactor scram...a heat up rate of 170 degrees in 45 minutes occurred in the 2A recirculation loop. The root cause of this event, as presented in the licensee event report, was in error and will be revised to reflect that the unreliable bottom head drain temperature indication prevented starting the recirculation pump." Deemed a Severity Level IV Violation, the NRC downgraded the event to a Non-Cited Violation. This was the ninth Non-Cited Violation since June 1998. (IR 050277/99008, 05000278/99008.)

October 2, 1999 - An unplanned isolation of the shutdown cooling occurred. (See (April, 200 and September 24 & October 2, 2000, for similar incidents.) (IR 05000277 & 278/2000-012.) -

October 6, 1999 - leakage of reactor coolant system water into the reactor closed cooling water system was caused by cracking in the 2"B' recirculation pump seal cooler. (See March 15, 2000, for problems associated with increased leakage). (IR 05000277 & 278/2000-001).

October 12, 1999 - PECO "confirmed to the NRC that the corrective actions associated with the Thermo-Lag fire barriers at Peach Bottom had been completed." (PECO Energy Company, Form 10-K/A, 1999, p. 10.)(See September 24, 1994, October 11, 1996, May 19, 1998, and July 21, 2000, for related material).

October 20, 1999 - A partially open main steam relief valve caused reactor cavity water to leak to the torus. (IR 050277/99008, 05000278/99008.)

October 20, 1999 - "An engineering modification error caused the flow indication for the 3A recirculation loop to be displayed on the wrong indicator." (IR 050277/99008, 05000278/99008.)

October 21, 1999 - Higher than expected radiation levels were monitored in the reactor cavity after drain-down. The source was the placement of "newly discharged fuel in close proximity to the spent fuel pool gates." (IR 0 5 0 0 0 2 7 7 / 1 9 9 9 0 0 9 , 0 5 0 0 0 2 7 8 / 1 9 9 9 0 0 9 & 0 7 2 0 1 0 2 7 / 1 9 9 0 0 9 .)

November 2, 1999 - "Although PECO engineering was aware that the Unit-2 high-pressure coolant injection (HPCI) steam admission valve could fail to open because of thermal binding when the system was isolated for maintenance, engineering personnel failed to prevent this type of failure during maintenance..." (IR 0500277/1999009, 05000278/1999009 & 0 7 2 0 1 0 2 7 / 1 9 9 0 0 9 .)- November 8, 1999 - during an NRC inspection, two violations relating to
Engineering Support of Facilities and Equipment were identified:

"The failure to adhere to procedural requirements in the performance of ultrasonic testing of safety-related components were identified by the inspectors as a violation of NRC requirements...The failure to include two core spray system welds in the ISI program plan was an violation..."
Both violations were downgraded an rated as Non-Cited Violations.This was the tenth Non-Cited Violation since June 1998.
- November 11, 1999 -A Non-Cited Violation was identified when the "2B CS pump room cooler failed to start during a routine quarterly surveillance test. Operations personnel determined that the room cooler fan switch was not fully turned to the 'run' position which prevented the fan from starting automatically when the pump was started." PECO also filed a LER. This was the eleventh Non-Cited Violation since June 1998.
(IR 05000277/ 1999009, 05000278/ 199009 & 07201027/ 199009.)

November 29, 1999 - "...the inspectors discussed with plant personnel the risk significance of the November 29, 1999, Topaz inverter failure that caused the loss of the alternate shutdown valve control function at the alternate shutdown panel...Although the Unit 3 Core Damage Frequency increased slightly due to this failure, the Sentinel on-line risk assessment still remained in the 'Green' band." (IR 05000277/199009, 05000278/199009 & 0 7 2 0 1 0 2 7 / 1 9 9 0 0 9 .)

December 2, 1999 - "...during a review of an RHR logic system functional test procedure prior to a planned test, operations personnel discovered that the test procedure simultaneously caused all four pumps to be incapable of starting automatically for a period of approximately two hours" (IR 05000277/ 199009, 0500278/ 199009 & 0720/ 199009.)
The NRC issued a Non-Cited Violation.This was the twelfth Non-Cited Vi o l a t i o n since June 1998.

December 19, 1999 - PECO Energy filed papers before the Pennsylvania PUC to acquire Connectiv's (formerly Delmarva Power & Light and Atlantic City Electric) share (15%) of Peach Bottom 2 & 3. The application was posted in the

Pennsylvania Bulletin on February 12, 2000. However, "On September 30, 1999, the Company announced it has reached an agreement to purchase an additional 7.51% ownership interest in Peach Bottom from Atlantic City Electric Company and Delmarva bringing the Company's ownership to 50%." (PECO Energy Company, Form 10-K/A, 1999, p. 11).
(See October 19, 2001, for a related acquisition by PSE&G).- December 27, 1999 - The NRC acceded to industry pressure to keep
information about nuclear plant shutdowns and restarts "confidential" unless the licensee "waives the right." "In the past, the NRC would supply information about most aspects of nuclear licensees' affairs, but with the move toward market competition, it became evident that the policy was having an effect on wholesale prices...The NRC's Mindy Landau said, 'We have seen shutdown information directly affect the prices on the spot market for electricity. ' " (The Energy Report, December 27, 1999.)

December 29, 1999 - "...Unit 2 load was reduced to approximately 70% power to support grid conditions for the millennium roll over." (IR 05000277/1999010, 05000278/1999010 & 07201027/1999 010.)

January 2000 - "...an Instrument and Controls (I&C) technician found that the existing 4KV emergency bus degraded grid relays could not be calibrated to a new, higher voltage setpoint in a revision to technical specifications...Engineering personnel determined that the causes were deficiencies in procedure adherence, attention to detail, and design review during the modification process and they initiated appropriate corrective actions." (IR 0500277/ 199910, 05000278/ 1999010 &07201027/ 1999010.)

January 12, 2000 - "A contract painter inadvertently bumped an E4 emergency diesel generator coolant expansion tank drain valve, resulting in a partial drain down of the coolant expansion tank. The emergency diesel generator remained operable. The problem was similar to a recent previous event."
The NRC "determined" this incident was a "minor violation." (IR 05000277/1999010, 05000278/19990 & 07201027/199010.)

January 19, 2000 - "Procedure errors with a Unit 2 high pressure coolant injection (HPCI) system tests led to a longer-than-planned period of unavailability for the HPCI system. The system manger conducted a thorough investigation of the problem and concluded that incomplete reviews during the revision process failed to identify the procedure errors." (IR 05000277/199010, 05000278/ 19990 & 07201027/ 199010.)

January 21, 2000 - "...Unit 2 load was reduced to approximately 65% for condenser water box cleaning and a control rod pattern adjustment." (IR

0 5 0 0 0 2 7 7 / 1 9 9 9 0 1 0 , 0 5 0 0 0 2 7 8 / 1 9 9 9 0 1 0 & 0 7 2 0 1 0 2 7 / 1 9 9 0 1 0 .)

January 26, 2000 - "...a Unit 3 turbine building equipment operator identified a degrading condition on the 3'B' RPS flexible coupling." (IR 0 5 0 0 0 2 7 7 / 1 9 9 0 1 0 , 0 5 0 0 0 2 7 8 / 1 9 9 9 0 1 0 & 0 7 2 0 1 0 2 7 / 1 9 9 0 1 0 .)- February 6, 2000 - "...during the transfer of a non-safety 4KV circuit breaker on the 2"b" control rod drive (CRD) pump, the breaker did not close as expected due to a mechanical failure of the anti-pumping relay." (IR 05000277 & 278/2000-001) .

February 25, 2000 - "...Unit 3 load was reduced to approximately 63% power to perform a control rod pattern adjustment, scram time and primary containment isolation system testing and replacement of the outboard main stream isolation valve DC solenoid valves". (See May 11, 2000, for a similar challenge). (IR 05000/277 & 278/2000-001).

March 4, 2000 - "...Unit 2 load was rescued to approximately 65%power for condenser water box cleaning." (IR 05000277 & 278/2000-001).

March 15, 2000 - "...the Unit 2 HPCI steam admission valve (MO-2-23--014) failed to open when operations personnel attempted to align the HPCI system for post-maintenance testing. PECO determined that this event was caused by thermal binding of the valve disk in its seat. A similar event had occurred in November 1999 and was documented in the NRC Inspection Report 50-277(278)/9908. Several corrective actions were initiated for the November event, included plans to upgrade the valve motor and placing the valve in a Maintenance Rule (a)(1) status in February 2000. (IR 05000277 & 278/2000-0 0 1) .

March 15, 2000 - "Leakage from the reactor coolant system water into the reactor building closed cooling water system (RBCCW) increased to "approximately 4.125 gallons per hour". (See October 6, 1999, for background information). (IR 05000277 & 278/2000-001).

March 22, 2000 - "...Unit 2 load was reduced to less than 20% power to allow personnel to enter the drywell and repair an instrument nitrogen leak. All Unit 2 inboard main steam isolation valves DC solenoids were replaced during this load drop." (See May 11, 2000, for a similar challenge at Unit 3). (IR 05000277 & 278/2000-001).

March 23, 2000 - "...while the HPCI system was inoperable for surveillance testing, the Unit HPCI MO-16 would not re-open after being taken to the shut position. Troubleshooting revealed that this failure was caused by high resistance associated with a contact in the open logic circuit. Maintenance personnel cleaned the contact and initiated actions to replace it.

“A similar event occurred in November 1998, when the same valve (MO-16) on Unit 2 failed to close due to an auxiliary contact problem. The contacts for this valve were recently removed for analysis during a scheduled maintenance activity on March 15, 2000. The cause of this failure was under investigation (PEP 10009425) at the time of the Unit 3 failure...“...Engineers appropriately recognized the possible recurring nature of this issue and the potential impact on system operability for similar failures on other DC motor-operated valves in the HPCI and reactor core isolation cooling systems. The inspectors noted that auxiliary contact failures have occurred in several safety and non-safety related valve breakers over the past few years. These failure have been documented in NRC Inspection Reports 50-277(278)99006, 98001 and 97005. (IR 05000277 & 278/2000-001).

March 24, 2000 - PECO Energy reached a comprehensive settlement with parties intervening in the proposed Unicom merger. “The Company reached agreement with advocates for residential, small businesses and large industrial customers, and representatives of marketers, environmentalists, municipalities and elected officials.” (PECO Energy, Press Release, March 24, 2000.) (See September 23, 1999 and April 1, 2000, for related developments.)

March 25, 2000 “...Unit 2 load was reduced to approximately 66% power due to problems with the 4’C’ feedwater heater lever control. (IR 05000277 & 278/2000-001).

April, 2000 - An unplanned isolation of the shutdown cooling occurred. (See September 24 & October 2, 2000, for similar incidents.) (IR 05000277 & 278/2000-012.)

April 1, 2000 - “Following the merger announcement, the shares of both firms dropped, indicating the market’s clear disapproval of the merger. PECO fell 4.4 percent and Unicom fell 2.2 percent on the day of the announcement...After 60 days, the shares of both firms were still below the pre-deal prices. PECO has lost over $1 billion in market capitalization. Unicom lost nearly $600 million. PECO shareholders lost more than Unicom, reflecting the market’s more positive initial view of of PECO. The market seems to think that the association with Unicom may decrease PECO’s performance.” (Public Utilities Fortnightly, April 1, 2000.) (See September 23, 1999 & March 24, 2000, for related incidents.)

April 25, 2000 - The NRC “determined that PECO Nuclear did not confirm or verify that the leak testing gauges used for preparation of a Type B shipping cask...conformed to accuracy requirements...The issue of PECO Nuclear’s ability to assure proper closure and leak testing of shipping casks is more than a minor issue since such inabilities could be a precursor to more significant events.”
The NRC deemed this infraction a Non-Cited Violation. This was the

thirteenth Non-Cited Violation since June 1998.(IR 05000277 & 278/2000-002). (See June 28, 1999 & August 3, 2000, for related incidents.) May 2, 2000 - "...a supervisor at the York County '911' center inadvertently activated the York County portion of the alert and notification sirens". (IR 05000277 & 278/2000-002).

May 7, 2000 - "Unit 2 load was reduced to approximately 90% power after the 2 'A' circulating pump was removed from service due to high motor upper guide temperatures." (IR 05000277 & 278/2000-002).

May 10, 2000 - "Unit 3 load was reduced to approximately 35% power after the 3 'B' recirculation pump was removed from service due to low motor oil level". (IR 05000277 & 278/2000-02). (See May 11, 2000, for related inc ident s) .

May 11, 2000 - "Unit 2 load was reduced to approximately 98% due to unexpected speed changes on the 2 'B' recirculation pump while raising or lowering pump speed." (IR 05000277 & 278/2000-002). (See May 15 and 19, 2000, for related incidents.)

May 11, 2000 - "Unit 3 power was further reduced to approximately 19% on to allow entry into the drywell to support adding oil to the 3'B' recirculation pump motor, repair of an instrument nitrogen leak, and replacement of all inboard main steam isolation valves DC solenoids". (IR 05000277 & 278/2000-002). (See November 27, 1998, February 25 and May 11, 200, for related problems. Also, refer to June 1, 1998 and March 22, 2000, for similar challenges at Unit 2).

May 12, 2000 - "Niagara Mohawk Power Corp. said on Friday that agreements to sell its nuclear assets to AmerGen Energy Co. have been mutually ended by the two companies." (See June 25, 1999, for background information.)

May 13, 2000 - The National Weather Service reported that a tornado touched down in the Peach Bottom-area.

May 15, 2000 - "Unit 2 load was reduced to approximately 86% to isolate the 'B' feedwater heater string due to a leak in the 'B2' feedwater heater." (IR 05000277 & 278/2000-002). (See May 11 and 19, 2000, for related incidents).

May 19, 2000 - "Unit 2 was placed in cold shutdown (Mode 4) to facilitate repairs of the 'B2' feedwater heater tube leaks." (IR 05000277 & 278/2000-002). (See May 11 and 15, 2000, for related incidents).

May 22, 2000 - At Unit 2, "a steam leak was discovered in the piping from the 'F' moisture separator to the 'B' low pressure turbine. The turbine was removed from service on May 22 and the leak was repaired. Unit 2 returned to

100% power on May 23." (IR 05000277 & 278/2000-006 & 07201027/2000-006).

May 27, 2000 - The United States Department of Justice, "filed an action claiming breach of contract against the Company in the United States Middle District of Louisiana arising out of the Company's termination of the contract to purchase Cajun's 30% interest in the River Bend nuclear power plant. The action seeks the full purchase price of the 30% interest in the River Bend nuclear power plant, $50 million, plus interest and consequential damages. While the Company cannot predict the outcome of this matter, the Company believes that it validly exercised its right of termination and did not breach the contract." (PECO Energy Company 1999 Annual Report, p. 46). (See June 5, 1997 and May 27, 1998, for background information).

May 28, 2000 - "The most recent packing gland follower cracking event occurred on a similar Unit 3 root isolation valve on May 28 ,2000 and resulted in the leakage of contaminated reactor coolant system water outside of the primary coolant. Leakage of contaminated reactor coolant system water outside of the primary containment is a significant condition adverse to quality." (See August 7, 2000, for more problems with packing gland follower cracking." (IR 05000277 & 278/2000-008)

BLACKOUTS & HIGH PRICES: SUMMER 2000

- April 11, 2000 - The North American Reliability's Council's (NERC) General Counsel, David Cook, testified before a Senate Committee, and "repeated findings of a recent NERC survey that several control area operators in the Eastern Interconnection were 'leaning' on the interconnection during nine peak hours (i.e., selling energy that they didn't have). (Public Utilities Fortnightly, May 15, 2000, p. 16)
- May 9, 2000 - "The Pennsylvania-New Jersey-Maryland (PJM) power pool implemented a five percent voltage reduction on May 9 to ease pressure on the distribution system.

"The action was taken to avoid emergency rolling blackouts where power is interrupted for short durations - typically 20 to 30 minutes." (Up d a t e, The Department of Environmental Protection, May 12, 2000, p. 2).
- May 16, 2000 - The electric utility industry predicted a 17% difference between supply and demand in a service area stretching from Virginia Beach to De t roi t .

"The all time maximum PJM demand of 51,700 MWQ occurred on July 6, 1999." (PECO Energy Company, Form 10 K/A, p.7).

June 28, 2000 - "This summer, (residential customers) probably have fewer choices than they did a few months ago, and the choices they do have are more expensive than they were...Combine strong economic growth with hot weather and the bad luck of having things like a number of power plants being shut down at the same time because of outages, and you certainly have problems." (Sony Popowsky, Consumer Advocate, Investor's Business Daily) .

In June, San Francisco suffered a blackout, and California has mandated usage restrictions for commercial, industrial, and residential customers.

June 9, 2000 - The NRC "approved transferring the operating license for the Oyster Creek nuclear station in New Jersey to AmerGen Energy Co." The New Jersey utilities board, which will meet on June 22, still needs to approve the transfer. ("Reuters", June 9, 2000, 3:12 pm.) (See September 11, 1997, for background information. Refer to August 16, 2000, for follow-up problems).

July 20, 2000 - "U.S. Energy Secretary Bill Richardson on Thursday said the government has agreed to allow PECO Energy Co. to defer up to $80 million in nuclear waste fee payments for its Peach Bottom plant in Pennsylvania, to compensate for the Energy Department's failure to store its waste...The deal allows PECO to reduce the projected charges passed into the Nuclear Waste Fund to reflect costs reasonably incurred by the company due to the department's delay." Press Release, U.S. Department of Energy. July 20, 2000.)

July 21, 2000 - "During the inspection, [April 14-18, 2000] the NRC identified two findings associated with the adequacy of post-fire safe shut down equipment circuit analyses at the station. Both of these issues were determined to be apparent violations...It is our understanding that you do not consider either of these two issues to be violations of 10 CFR 50 or your operating license. Additionally, we recognize that other commercial nuclear power plant operators, represented by the Nuclear Energy Institute (NEI), have adopted a similar position regarding these issues. As such, in accordance with our current enforcement policy...the NRC will defer any further enforcement action relative to these issues until the staff evaluates NEI's proposed resolution methodology." Wayne D. Lanning, NRC, Director, Division of Reactor Safety. (See May 19, 1998 and October 12, 1999, for related events.)

August 3, 2000 - PECO was assessed a "White" level Violation for its "failure to properly classify radioactive waste for shallow land burial...Specifically, the shipment was identified as Class A waste containing 99 curies when it should have been classified as Class B waste containing 407 curies." (NRC, Hubert J. Miller, Regional Administrator). (Refer to June 28, 1999, for background information. See April 25, 2000, for a related incident.)

August 7, 2000 - Unit 3 "automatically shutdown from 100% power when a one inch instrumentation rack root valve packing gland follower failed and caused a false reactor low level input into the RPS [reactor protection system]. The failure occurred when the packing gland follower broke into two pieces allowing package leakage of contaminated reactor coolant system water from the instrumentation piping. The leak was immediately isolated by actuation of the excess flow check valve in the instrumentation piping line. Unit 3 also experienced Groups II and III primary containment isolation valve closures due to the false reactor low level signal."

The NRC issued a Non-Cited Violation.This was the fourteenth NonCited Violation since June 1998.
The NRC also criticized PECO's corrective action program: "Two previous packing gland follower cracking incidents had occurred on similar valves at the facility during the past eighteen months. The most recent packing gland follower cracking event occurred on a similar Unit 3 root isolation valve on May 28, 2000 and resulted in the leakage of contaminated reactor coolant system water outside of the primary coolant. Leakage of contaminated reactor coolant system water outside of the primary containment is a significant condition adverse to quality.The identification of this significant condition adverse to quality was not adequately documented in PECO's corrective action system, and as a result, the cause of the condition was not determined, corrective actuation was not taken to prevent repetition, and generic concerns with potential packing gland follower cracking on other valves were not addressed." (IR 05000277 & 278/2000-008)
The NRC issued a Severity Level IV violation "related to the identification and resolution of problems on leakage of contaminated reactor coolant system water caused by cracking of instrument root valve packing gland followers."

August 14, 2000 - AmerGen reported a valve failure [reactor building isolation valves] at Oyster Creek that forced the plant to shutdown at 82% power. "It's too premature to guess at a date the unit may return. We're still evaluating the problem and will likely replace the valves that failed, " AmerGen Spokeswoman, Debra Piana. ("Reuters", August 16, 2000.) (Please refer to September 11, 1997 and June 9, 2000 for additional information.)

August 22, 2000 - The NRC issued a Non-Cited violation related to "inservice tests for the standby liquid control pumps. A two-minute wait was not mandated, as required in the applicable Code, by the test procedure before pump flow and pressure measurements were recorded. Because of the very low safety significance, the violation was non-cited." This was the fifteenth Non-Cited Violat ion since June 1998. (NRC, Wayne D. Lanning, Director, Division of Reactor Safety, IR 05000277 & 278/-005.)

August 23, 2000 - "Operators reduced power [at Unit 2] to approximately 68% to remove the 'B' feedwater heater string from service due to suspected leaks and on August 24 returned the unit to 83% power." (See September 7 & 13, 2000, for related incidents.) (IR 05000277 & 278/2000-0 1 0 .)- September 7, 2000 - "Operators reduced power [at Unit 2] to approximately 16% in response to pressure perturbations in the 'B' feedwater heater string and on September 8 returned the unit to 75% power." (See August 23 & September 13, 2000, for related incidents.) (IR 05000277 & 2 7 8 / 2 0 0 0 - 0 1 0 .)

September 13, 2000 - Operators reduced power to approximately 16% at [Unit 2] in response to pressure perturbations in the 'B' feedwater heater string and on September 8 returned the unit to 75% power." (See August 23 & September 7, 2000, for related incidents). (IR 05000277 & 278/2000-010.)

September 15, 2000 - "...with Unit-2 at approximately 16% power and 24% flow, operators performed a manual scram to prevent operation in the restricted zone of the power flow map after an unplanned trip of the 2B reactor recirculation pump." (IR 05000277 & 278/2000-012.)

September 16, 2000 - Three workers failed to follow oral and written instructions, and "either worked in proximity of , passed through, or transported radiation shielding materials through elevated radiation fields (up to 13.9 R/hr) in the drywell. As a result, one of the workers did not contact radiation protection personnel upon alarm of the dosimeter, also as specified in written and oral radiation protection instructions.
"This issue was considered to be of very low safety significance...a N o n - cited violation " was issued. This was the sixteenth Non-Cited Violation since June 1998. (IR 05000277 & 278/2000-010.)

August 31, 2000 - Exelon issued an LER after determining that three of four EDGs "were inoperable during the summer of 1999, based on their inability to mitigate a postulated loss-of-coolant-accident plus loss-of-off-site-power design basis accident for a maximum of approximately 25 hours. The licensee attributed the cause of the event to be an original design deficiency on the EDGs, which allowed cross-flows between the jacket water coolers and the intake-air coolers." (IR 50-277/01-06, 50-278/01-06.).

September 24, 2000 - During the 2R13 refueling outage, a "spurious" unplanned isolation of the shutdown cooling occurred. (See October 2, 2000, for similar incidents.) (IR 05000277 & 278/2000-012.)

September 28, 2000 - "...operations personnel determined, during inservice testing, that ESW [Emergency service water] check valve 2-33-514
failed [sic] open. The check valve is designed to prevent reverse flow from the safety-related ESW into the Unit 2 non-safety related water service system. Operators declared both ESW systems inoperable, because ESW flow to the EDGs and emergency core cooling system room coolers and motor oil coolers could be i n a d e q u a t e . . . ""The inspectors and operations personnel noted that, during two periods in
which the ESW system was declared inoperable, operators did not address the operability status of the EDGs or associated Technical Specifications action statements and/or applicable limiting conditions for operation of Unit 2 which was in Mode 5 (refueling) at the time..."
"The inspectors determined that this event required further evaluation in the significance determination process." (See October 1 through November 18, 2000, for an identical problem). (IR 05000277 & 278/2000-010.)

September 30, 2000 - Operators reduced power to approximately 18% in response to a low oil level in the 3B recirculation pump motor. Unit 3 was at approximately 35% power." (IR 05000277 & 278/2000-010.)

October 1 through November 18, 2000 - "Emergency service water (ESW) system check valve 2-33-514 failed [sic] open, allowing safety-related ESW flow to be partially diverted from emergency diesel generators(EDGs) and emergency core cooling system room coolers. The inspectors and the licensee identified that this risk important component had not been included in a preventive maintenance program.
"This issue caused the ESW system and the EDGs to be degraded for a period of up two years. This finding was of very low safety significance because, although the ESW flow rate to the EDGs was below the design basis minimum value engineering personnel determined that the EDGs would have remained available during accident conditions." A Non-Cited Violation was issued."
This was the seventeenth Non-Cited Violation since June 1998. (See September 28, 2000, for a related incident.) (IR 05000277 & 278/2000-012.)

October 2, 2000 - Three unplanned isolations of the shutdown cooling (SDC) occurred. "Engineering personnel stated that these events were caused, in part, by an ILRT (Integrated Leak Rate Test) procedure that did not fully account for the reduced operating margin to the high pressure isolation setpoint..."
"At the time of the isolations during the ILRT, SDC was the only operable decay heat removal system..."
Continued on the following page... "The inspectors identified that there were previous occurrences of SDC
isolations on Unit 2 that were not fully investigated. For example, on October 2, 1999, a similar SDC isolations occurred, but no cause was identified. The pressure switches were found to be in calibration. No PEP corrective action plan document was initiated. Further, in April 2000, engineering personnel initiated an action item to troubleshoot isolations, but no action had been taken prior to the outage. The inspectors brought this issue to the attention of engineering management. Engineers also noted that there were two other not-fullyunderstood SDC isolations on Unit 2 since 1994. The inspectors concluded that engineering personnel had missed opportunities to investigate previous SDC isolations and this constituted a corrective action performance issue."
The inspectors did not identify a violation of NRC requirements.
(See September 24, 2000, for related incident.) (IR 05000277 & 278/2000-012.)

October 4, 2000 - Unit-2 was taken critical.

October 4, 2000 - Unit-2 "operators halted the reactor startup following the discovery of a missed post-maintenance test on a control rod." (IR 05000277 & 278/2000-012.)

October 17, 2000 - Unit-2 "operators reduced power to approximately 65% to repair a condenser tube leak. The unit was restored to 100% on October 18." (IR 05000277 & 278/2000-012.)

October 22, 2000 - "...the failure of the Unit-2 'H' torus/drywell vacuum breaker to fully close during surveillance testing rendered primary containment inoperable...Unit load was reduced to 16% due to an inoperable torus/drywell vacuum breaker...Because of the very low safety significance of this item and because the licensee has included it in their corrective action program (PEP I0011883), this procedure violation is being treated as a Non-Cited Violation." This was the eighteenth Non-Cited Violation since June 1998 (IR 05000277 & 278/2000-012.)

October 23, 2000 - Unit-2 was shut down to repair the torus/drywell vacuum breaker. The reactor was taken critical on October 24 and unit load was 100% on October 26." (IR 05000277 & 278/2000-012.)

November 13, 2000 - "Operators reduced load to 79% [at Unit-2] to repair the 2C circulating water pump traveling screen. The unit was restored to 10% power on the same day." (IR 05000277 & 278/2000-012.)
December 17, 2000 - An LER was issued "when a lightning strike caused the failure of a communications circuit board to a main off gas stack radiation monitor which resulted in a spurious invalid signal causing the isolation." Unit 3 was at approximately 18% power when the lightning strike caused the isolation. (IR 05000277&278/2001-002).

March 23, 2001 - Examinations for reactor operators and senior reactor operators held from February 5-12, 2001,"indicated that a relatively high percentage of the applicants were not well prepared for the exam." (Richard J. Conte, NRC, Chief, Operations Safety Branch, Division of Reactor Safety.)

May 20, 2001- Corbin A. McNeill's base salary after the merger increased from $659,857 to $855,830 and his bonus was increased from $1 million to $1,081, 4572. In addition, McNeill's restricted stock increased from $942,188 to $2.8 million. (See June 13 and September 28, October 24 & December 21, 2001, for information on 900 job cuts, and refer to January 29, 2002, for further job cuts. Also, reference February 26, 2002, for information on McNeill's "retirement package.")

May 29, 2001 - At Unit 3, "... the fifth stage feed water heaters were removed from service for end-of-cycle coast down. Unit 3 ended the inspection period at approximately 98 percent power with the four stage feedwater heaters removed from service." (IR 50-277/01-05, 50-278/01-05 & 07201027/01-05).

June 13, 2001 - Exelon Nuclear "announced its intent today to eliminate

292 Local 15 Union positions, including 138 layoffs in Exelon Nuclear and 154 at
Commonwealth Edison." (Exelon, New Release, June 13, 2001.) (See September 3, 1998, for further Exelon "downsizing"). (Refer to May 20, 2001, for Corbin A. McNeill's pay raise.)

June 22, 2001- After widespread public criticism, AmerGen "notified the Nuclear Regulatory Commission that it intends to delay submitting its application seeking approval for a standardized emergency plan for Three Mile Island, Peach Bottom and Limerick." (Exelon Nuclear, Press Release, June 22, 2001.) (See August 15, 2001 for more information & November 7, 2001, for a related development)

June 30, 2001 - At Unit 2, "...operators commenced an unplanned power reduction to approximately 63 percent to allow repair of an electrohydraulic control system leak at a servo on the No. 2 main turbine control valve. Later that same day, operators returned the unit to 100 percent power." (IR 50-277/01-05, 50-278/01-05 & 07201027/01-05).

June 30, 2001 - "...Exelon Nuclear notified the Nuclear Regulatory Commission (NRC) that it intended to file for renewal of the operating licenses for Peach Bottom Units 2 and 3...
"If approved, Unit' 2's license would be extended from 2013 to 2033 and Unit 3's from 2014 to 2034..."The Nuclear Regulatory Commission is expected to take two years to
thoroughly review the license renewal application before determining whether to grant the license extensions..."
"The total cost of obtaining the renewed licenses for Peach Bottom will be about $18 million, including the NRC review, or about $8 per kilowatt hour...Exelon Nuclear also has notified the NRC that it intends to file for license renewal[s] for its Dresden and Quad Cities Stations in Illinois." (Exelon Nuclear, Press Release, July 2, 2001.)

August, 15, 2001- The NRC's Office of Investigation documented criminal behavior by two of Exelon's Emergency Preparedness personnel. The NRC found that the "technicians fabricated siren testing maintenance records, performed deficient siren tests on the off site EP response sirens and intentionally installed jumper wires in the siren boxes disabling important system functions." (Wayne D. Lanning, NRC, Director of Reactor Safety.) (Refer to August 22, 2001, for background information, and see October 23, 2001, for penalty assessment.). (See June 22 & November 7, 2001, for related developments.) (See October 5-9, 2001, for a related problem at TMI.)

August 22, 2001 - The NRC determined that a white "finding"

(Violation) was warranted for the following infractions relating to the plants Public Address (PA) system and evacuation alarm/siren (EA) system:

1. From 1992 to December 19, 2000, approximately 47% of the PA system's speakers were either inaudible or degraded to the point that personnel were not able to clearly hear instructions.

2. From January 19, 2001 to February 13, 2001, and again from March 20, 2001 to April 17, 2001, the plant PA system was operated only on the backup power breaker, which would have tripped after about 49 seconds of evacuation alarm actuation on the first sequence. (The primary breaker had tripped following the monthly test the beginning of each period.)

3. On February 13 and April 17, 2001, the plant PA/EA system would not properly function in that both the primary and the backup breakers were tripped for periods of 4.5 hours and 1.5 hours resulting in no system capability to provide instruction or sound the evacuation alarm. (Hubert J. Miller, NRC. Regional Administrator.) (See August, 15, 2001, for a related development.)

August 20, 2001 - "...the inspectors observed a health physics technician that was inattentive to his duties when he was assigned to restrict access to a posted high radiation area on the Unit 3 turbine floor...that applies to high radiation areas with dose rates in excess of 100 millirem per hour but less than 1000 millirem per hour at 30 centimeters from the source..." (IR 50-277/01-09, 50-278/01-09).This was the nineteenth Non-Cited Violation since June 1998.

September 6, 2001 - A Non-Cited Violation "of very low safety significance" was recorded for, "The failure to test the Units 2 and 3 HPCI [high pressure coolant injection] torus suction check valves for seat leakage in the reverse flow direction was more than minor because it had a credible impact on safety. Significant leakage in the reverse flow direction could prevent HPCI from performing its function when HPCI is aligned to pump water from the torus. The failure to leak test these valves affected the Mitigating System cornerstone since HPCI performs an accident mitigation function." (IR 50-277/01-06, 50-278/01-06).

This was the twentieth Non-Cited Violation since June 1998.

September 8, 2001- Unit 2 was taken critical and "operated at approximately 100% power for the remainder of the inspection period except for scheduled power changes to support rod pattern adjustments." (IR 50-277/01-09, 50-278/01 -09).

September 14, 2001- Unit 3 "began this inspection period at approximately 81 percent power, in end-of-cycle coastdown, with the fourth and fifth stage feedwater heaters removed from service on. On September 14, 2001, Unit 3 was manually scrammed, in preparation for the 3R13 refueling outage. Unit 3 ended the inspection shutdown in Mode 5 (Refueling)." (IR 50-277/01-09, 50-278/01-09).

September 17, 2001- TMI-Alert filed a Petition for rule making with the NRC requiring the Agency to mandate armed security guards at the entrance to all nuclear rower plants. A final decision is expected in November l, 2002. The Nuclear Energy Institute, Exelon's s "voice in Washington, "recommended" that the Petition be "denied."

September 28, 2001 - With third quarter profit projections down from $1.35 to $1.80 a share, Exelon announced the elimination of 450 jobs. (See June 13, 2001, for earlier job losses.)
Exelon's stock dropped to $44.50 on September 27, 2001. (See May 20, 2001, for Corbin A. McNeill's pay raise, and October 24, December 21, 2001, for related downgrades.)

October 1, 2001 - The NRC reported on Exelon's Emergency Preparedness program:
Although you believe the current EP program remains ready to effectively protect public health and safety, you stated it did not meet Exelon's vision of an industry leading program. Your presentation included changes and improvements to: (1) EP organization/staffing; (2) EP equipment reliability; (3) EP program processes; and 94) the corrective action process. (Richard J. Conte, Chief, NRC, Operations Safety Branch, Division of Reactor safety, October 18, 2001. (See June 22 August 15, 2001 for background information & November 7, 2001, for a related development)

October 5-9, 2001 - At TMI, "Licensee sirens in Lancaster County were inoperable October 5 through October 9, 2001, due to a radio transmitter being deenergized at the county facility. The transmitter is part of the siren actuation system. This issue is unresolved pending further investigation into the lines of ownership and maintenance of the actuation system" (IR 50-289/01-07.) (See August 15, 2001, for a related problem at Peach Bottom.)

October 6, 2001 - The Federal Energy Regulatory Commission (FERC) filed a "show cause" order relating to PECO Power Team's purchase during a power auction that may have benefited from "informational advantage" from Peco. ("Philadelphia Inquirer", October 6, 2001.) On December 19, 2001, according to Exelon, the FERC "terminated its investigation into alleged wrongdoing..." (Exelon Corporation, Press Release, December 19, 2001.)

October 6, 2001 - After the September 11, 2001 terrorist attacks on the World Trade Center, the Pentagon and a downed airliner in Somerset County, Pennsylvania, the NRC has issued a "Security Advisory", and requited 13 "prompt actions which are "safeguarded" and "classified." (See October 17, 2001 & November 2, 2001, for related incidents).

October 8, 2001- The NRC issued another Non-Cited Violation, and concluded that Exelon's "Troubleshooting, Rework, and Testis Process" (TRT) "would not adequately control Unit 3 reactor vessel water levels." (IR 50-277/01-09, 50-278/01-09)
This was the twenty-first Non-Cited Violation since June 1998.

October 8, 2001- Unit 3 was taken critical and "operated at approximately 100% power for the remainder of the inspection period except for scheduled power changes to support rod pattern adjustments." (IR 50-277/01-09, 50-278/01 -09) . - October 12, 2001- "....during the Unit 3 startup from a refueling outage,
when the jet pumps had been cleaned, core flow exceeded 100% (at 106.3%) for a period of ninety minutes before operations personnel initiated actions to reduce core flow to within 100%." (IR 50-277/01-07, 50-278/01-07.)
This was the twenty-second Non-Cited Violation since June 1998.

October 17, 2001 - Due to a "credible threat" against Three Mile Island, the Harrisburg and Lancaster airports were closed for four hours, air travel was restricted in a 20-mile radius, a fighter jets were scrambled around TMI. (See October 6, 2001, & November 2, 2001, for a related events.)
Through the Freedom of Information Act, the York Daily Record (December 21, 2003) found a "twofold" challenge when a threat against Three Mile Island caused the Harrisburg and Lancaster airports to close for four hours: Air travel was restricted in a 20-mile radius and fighter jets were scrambled around TMI.

Officials struggled with whom to call first, next and last. Officials struggled with notifying state and local officials. And officials struggled with when and whether to notify the public...One NRC official had difficulty reaching senior management at TMI...No one contacted enforcement officials in York County about the threat...[PEMA] officials had to push plant officials to staff their emergency operations facility
[in Susquehanna Township which was later relocated to Coatesville].

October 19, 2001 - PSE&G acquired Atlantic City and Electric Company's stake in Peach Bottom. (See December 1, 1999, for a related acquisition by Connectiv).

October 23, 2001 - On August, 15, 2001, the NRC's Office of Investigation documented criminal behavior by two of Exelon's Emergency Preparedness personnel.
In accordance with the Enforcement Policy, a base civil penalty in the amount of $55,000 is considered for Severity Level III violation or problem. Because the Severity Level problem was deliberate, the NRC considered whether credit was warranted for Identification and Corrective Action in accordance with

the civil penalty assessment process in Section VI.C.2 of the Enforcement Policy. In this case, the NRC decided that credit for Identification is warranted because you identified the misconduct and informed the NRC." (Hubert Miller, NRC, Regional Administrator, October 23, 2001).This was the twenty-third Non-Cited Violation since June 1998.
Exelon's total cost avoidance, i.e., "credit" for 23 Non-Cited Violations = $1, 155,000.

October 23, 2001 - At Unit 2, "an automatic reactor shutdown occurred due to a generator lockout and main turbine trip. Following troubleshooting and repairs, the unit was restarted on October 27 and reached 100% power on October 30. (IR 50-277/01-09, 50-278/01-09).

October 24, 2001 - Exelon Corporation's stock was downgraded from "Buy" to "Mkt Perform" by Banc of America and from "Strong Buy" to "Hold" by
UBS Warbug. (See May 20, 2001, for Corbin A. McNeill's pay raise, and September 28 and December 21, 2001, for related downgrades.)

October 30, 2001 - "...the E-2 emergency diesel generator (EDG) tripped on low jacket coolant discharge presurre during routine testing of the EDG...Although Exelon was unable to detemine who closed this valve or exactly when it was closed, they did determine that the valve was closed somewhere in the period between October 12, 2001 and Ocotber 30, 2001...The EDG was successfully tested and returned to service on October 31, 2001" (IR 50-277/01-10, 50-278/01 - 10.)
This was the twenty-fourth Non-Cited Violation since June 1998.
Exelon's total cost avoidance, i.e., "credit" for 24 Non-Cited Violations = $1 ,205,000.

November 2, 2001 - Governor Mark Schweiker reversed an earlier decision, and ordered the National Guard to Pennsylvania's nuclear power plants.The Commonwealth joins over a dozen states with National Guard and/or Coast Guard detatchments depolyed to protect nuclear facilities against terrorist attacks. (See October 6 & 17, 2001, for related incidents). - November 7, 2001 - Exelon met with the NRC to discuss the consolidation
of Emergency Plans for TMI, Peach Bottom and Limerick. Exelon requested the plans be approved and implemented by January 2, 2002. The following personnel (17), including a "Security Coordinator" would be affected:
* LGS and PB Emergency Plan Positions Affected
1 Communicator
2 Dedicated Maintenance Technicians
1 Dose Assessor
2 Dedicated Off-Site Survey member

* TMI Emergency Plan Positions Affected

4 Technicians
1 On-Site OSC Coordinator
1 Dose Assessor
1 Off-Site Field Team Member
1 Communicator
1Security Coordinator
2 Auxiliary Operators.
(Presentation by: William Jefferson, Director, Generation Support, Exelon Nuclear, MidAtlantic Regional Operating Group, May 16, 2001.) (See June 22, August 15, & October 1 2001, for related developments.)

November 8, 2001 - At Unit 3, "...operators commenced a schedlued power reduction to approximatley 19% because a primary containment isiolation valve in the redisual heat removal system in the drywell failed to close when it was tested."(IR 50-277/01-10, 50-278/01-10.)-November 28, 2001 -Exelon Power Team stated that the collapse of Enron will cost the Company "less than $10 million. The current direct exposure (i.e., for current energy sales from Exelon to Enron) is less than $20 million. (Exelon Corporation, Press Release, November 28, 2001.)(See October 8, 1997, for a related development.) Three days later, on December 1, 2001, PPL stated that the collapse of Enron may cost the Company $40 million for energy already purchased. Enron also owns 45% of power plant in New England operated by PPL. (Philadelphia Inquirer, Business, December 1, 2001.)

November 30, 2001 - At Unit 2, "...operators commenced a schedlued power reduction to approximatley 19% to repair an instrument nitrogen leak in the drywell. Following repairs, the unit power was increased and reached 100% on Decmber 2, 2001."
(IR 50-277/01-10, 50-278/01-10.)

December 5, 2001 - Business Day of Joahnnesburg South Africa reported Exelon was negotiating to but 40 Pebble Bed Modular Reactors from Eskom. The order, estimated to be as much as $6 billion, assumes delivery of the reactors to the Untied States by 2007. (See December 10, 2001, for related development.) Refer to April 17, 2002, for information realting to Exelon's decision to pull-out of the project.

December 10, 2001 - Unreco, a uranium supplier, is seeking regulatory approval to build the first new enrichment facility in the US in half a century. The project, estimated to cost $10, is a joint venture of Exelon and duke Power. (Financial Times, December 10, 2001) (See December 5, 2001, for a related d e v e l o pme n t .)

December 21, 2001- Exelon Corporation's stock was downgraded from

"Accumulate" to "Hold" by Jeffries & Co., and Lehman Brothers stated, "We believe an economic recovery is key to the Exelon story, which is highly leveraged to power prices..." (Reuters, December 21, 2001.) (See May 20, 2001, for Corbin A. McNeill's pay raise, and September 28 and October 24, 2001, for related downgrades. Also, refer to January 29, 2002, for further job cuts.)

January 9, 2002 - A well-armed, disgruntled former employee at the San Onfore nuclear power plant in San Clemente was arrested for making threats against the plant.- January 11, 2002 - Siren testing at TMI ecountered numerous problems:
all sirens failed in York County and one siren failed in Lancaster County. AmerGen attributed to computer malfucntions. (August, 15, 2001, and October 5-9, 2001.)

January 9, 2002 - A well-armed, disgruntled former employee at the San Onfore nuclear power plant in San Clemente was arrested for making threats against the plant. (See October, 6, 2001, and January 30 and December 10, 2002, for related incidents.)

January 11, 2002 - Siren testing at TMI encountered numerous problems: all sirens failed in York County and one siren failed in Lancaster County. AmerGen attributed to computer malfunctions. (August, 15, 2001, and October 5-9, 2001.)

January 29, 2002 - Exelon announced it would cut 3,400 or 15% of its work force by the end of 2002. (See May 20, 2001, for Corbin A. McNeill's pay raise, June 13 and September 28, October 24 & December 21, 2001, for information on 900 job cuts. Also, reference February 26, 2002, for information on McNeill's "retirement package.")

January 30, 2002 - President Bush's State of the Union Address including a warning that nuclear power plants may be targeted for a terrorist attack. (See October 6 & 17 and November 7, 2001, and January 9, 2002 for related events.)

February 14, 2002 - Exelon prepared an "inadequate critique" of their "emergency preparedness exercise." (See July 1, 2002.)

February 26, 2002 - Corbin McNeill Jr. announced his retirement, and he is expected to receive $7 million when leaves the Company in April, 2002. He will also receive a bonus payment. McNeill made $2.5 million in 2001.*
"His severance equals triple the sum of his annual base salary plus the average of his bonus over the last two years.
"McNeill is the company's largest individual shareholder. His 1.53 million shares are worth $79.1 million based on yesterday's closing price of $51.70."

(Philadelphia Inquirer, C-1, March 14, 2002.)* Corbin A. McNeill’s base salary, after the merger increased, from
$659,857 to $855,830, and his bonus was increased from $1 million to
$1,081,572. In addition, McNeill's restricted stock increased from $942,188 to
$2.8 million. [May 20, 2001.]
(See June 13 and September 28, October 24 & December 21, 2001, for information on 900 job cuts, and refer to January 29, 2002, for further job cuts. ”)

March 28, 2002 - The NRC admitted that Peach Bottom and the 102 nuclear power plants could not withstand an impact of airplane the size of those that crashed into the Pentagon and World Trade Center on September 11, 2001. (March 28, 2002, Patriot News.) (See October 2001 & October 17, 2001 and January, 9 and 30, 2002, for related incidents.)

April 3, 2002 - “Two men and a male juvenile from Mexico face possible deportation after attempting to enter an unprotected area of the Peach Bottom Atomic Power Station. All three remained in INS custody Wednesday.”(York Daily Record, April 4, 2002.) (See January, 2001, October 6, 2001 & October 17, January, 9 and 30, 2002, and March 21 and May 15, 2002, for related inc ident s .)

April 17, 2002 - Exelon Corp., the country's largest nuclear plant operator, said yesterday that it would end its bid to develop the next generation of nuclear reactors.
The Chicago-based parent of Peco Energy Co. said it would terminate its nearly two-year relationship with Eskom, South Africa's state-owned utility, in building a prototype gas-cooled reactor. Exelon is getting out of the business of designing nuclear plants and will concentrate instead on operating them.
The company spent $20 million on the project, of which it owned 12.5 percent. Exelon said it already had paid for its share as a research-anddevelopment expense. It has not decided what to do with the 12 employees it had working on the project, a spokeswoman said. (Benjamin Y. Lowe, Philadelphia Inqui r e r, April 17, 2002.) (See December 5 & 10, 2001, for background i n f o rma t i o n .)

April 22, 2002 - Exelon's 1st-Quarter Net Income Fell 98%
... as mild winter weather and maintenance costs hurt results.
“The country's largest operator of nuclear power plants reported late Monday net income of $8 million, or two cents a share, compared with net income of $399 million, or $1.23 a share, a year earlier.
“The latest results included a charge of $230 million, or 71 cents a share, from the effect of adopting SFAS 142 for goodwill amortization, while year-earlier results included a tax benefit for the implementation of SFAS 143 for derivatives. Excluding these items, the company said it had operating earnings of 77 cents a share, compared with operating earnings of $387 million, or $1.19

a share." (Mon Apr 22,10:53 PM ET , CHICAGO -- Exelon Corp.)
(See June 13 and September 28, October 24 & December 21, 2001; and, January 29 & February 26, 2002. For information related economic de v e lopment s .)

May 11, 2002- "Exelon Corp. is the subject of a shareholder lawsuit alleging the electric and gas utility made false and misleading statements that artificially inflated its share price. The law firm of Charles J. Piven said it filed a lawsuit on behalf of buyers of Exelon shares between April 24, 2001, and September 27." (Philadelphia Inquirer, D-3, May 11, 2002.)

May 15, 2002 - "A foreign intelligence service recently warned that a nuclear power plant in the Northeast could be the target of a July 4 terrorist attack...Published reports suggested that the target could be Pennsylvania's Three Mile Island, but a second US official with knowledge of the information said no specific facility had been named." (Knight Ridder, May 15, 2002.) (See January, 2001, October 6, 2001 & October 17, January, 9 and 30, 2002, and March 21, for related incidents.)

May 28, 2002 - "Exelon Corp. and three other utilities [Main Yankee Atomic Power Co., Omaha Public Power District & Sacramento Municipal Utility District] lost a $2.2 billion legal challenge to the federal government's nuclearwaste cleanup plan...In 1992, Congress ordered utility companies that use government uranium-enrichment services to pay one-third of the cleanup bill. The U.S. Supreme Court said yesterday that it would not hear an appeals from the companies that argue that the assessments are unconstitutional." ("Associated Press", May 29, 2002.)

June 2, 2002 -An alert began at around 12:30 am, ending at 3:01 am, relating to the activation of the fire suppression system due to EDG failure which released carbon dioxide into a room where two employees were working. No injuries were reported and both Peach Bottom 2 & 3 remained at 100% power. (Exelon Nuclear, News Release, June 2, 2002, 4:10 am.) (See November 26, 2002 for follow-up, and July 11, 2003 for absolution.)

June 12, 2002 - The Bioterrorism Bill signed into law on June 12, 2002 mandates KI stockpiles out to 20 miles.

June 25, 2002 - "...station emergency preparedness personnel discovered that the emergency planning siren base station at the site, was unable to communicate with the off site sirens, due to external radio frequency noise in the area." (IR-50-277/02-05; 50-278/02-05)

July 1, 2002 - The NRC found that on February 12, 2002, Exelon "did not identify that key information needed by the emergency director (ED) to classify the simulated event as a General Emergency was not provided to the ED by members of the Emergency Response Organization (ERO). The finding was preliminary classified as White because the critique failed to identify a problem associated with the implementation of a risk significant planning standard." ...Continued on the following page...Exelon disputed the findings on September 4, 2002.

The NRC reasserted that "the critique problems were more than minor but the Issuance of the White finding is not appropriate because the inadequate critique did not result in a failure to identify a risk significant planning standard (RSPS) problem."

The incident is classified a Non-Cited Violation.

(Final Significance Determination for Green and White Findings and a Notice of Violation at Peach Bottom, IR-50-277/02-07; 50-278/02-07).

This was the twenty-fifth Non-Cited Violation since June 1998. Exelon's total cost avoidance, i.e., "credit" for 25 Non-Cited Violations = $1 ,255,000.

July 21, 2002- At Unit-2, "the fifth stage feed water heaters were removed from service for end-of-cycle coast down." (IR-50-277/02-05; 50-278/02-05). (See August 4, 2002 for related event.)

July 23, 2002- "Exelon did not evaluate in a prompt manner whether it was appropriate to disable the electrical trips of the EDGs from the cardox injection fire protections system after NRC inspectors identified that the trips were still active with the EDG cardox system isolated" (IR-50-277/03-02; 50-278/03-02) (Also refer to IR-50-277/02-04; 50-278/02-04).
(See April 23, 2004 for NCV).

August 4, 2002- At Unit-2, "the fourth stage feed water heaters were removed from service." (IR-50-277/02-05; 50-278/02-05). (See July 21, 2002 for related event.)

August 15, 2002 - Despite a favorable EIS of Exelon's request for a license extension at Peach Bottom-2 & -3, the NRC listed three safety issues that need to be addressed prior to approval: replacement o electric fuse clips; removal of the anti-aging plan; and, replacement of faulty cables.

August 30, 2002- At Unit-3, "power was reduced to approximately 90% prior to shut down the 3 'A' recirculating water pump because of high differential pressures on the circulating water intake screens. The high differential pressures were caused by a sudden surge in the amount of fish (Gizzard Shad) that entered the intake canal and clogged the screens. Unit 3 power was returned to 100 percent following cleaning of the circulating water screens and restating of the 3'A' circulating water pump." (IR-50-277/02-05; 50-278/02-05).

August 31, 2002 - New security budget increased to $2.2 million annually or $550,300 less than John W. Rowe's base salary.- September 5, 2002 -- Three Mile Island Alert filed a formal Petition for Rulemaking with the Nuclear Regulatory Commission to include day-care centers and nursery schools in emergency evacuation planning. The proposed rule would affect all 103 operating nuclear plants in the United States.

September 10, 2002 - The Office of Homeland Security announced that the "yellow" warning had been increased to a heightened state of alert or an "orange" upgrade at 1:00 pm. (Exelon Public Relations.)
- "...Unit 2 was manually scrammed, in preparation for the 2R14 refueling outage" (IR-50-277/02-05; 50-278/02-05).

November, 2002 - " Governor Schweiker "directed the National Guard to join State Police in a joint security mission at the state's nuclear facilities." In December, the Governor extended the joint mission of the National Guard and the State Police at the Commonwealth's five nuclear generating stations until March 4, 2002. (DEP, Update, December 6, 2002.)

September 21, 2002 - A Non Cited Violation was issued for incident "when a chain broke" on a "rigging hoist and the motor, weighing approximately 48,000 pounds, fell approximately ten inches into the pump/motor stand."
This was the twenty-sixth Non-Cited Violation since June 1998. Exelon's total cost avoidance, i.e., "credit" for 26 Non-Cited Violations = $1 ,355,000.

November 26, 2002 - Initially classified as a White, the incident was classified a Non-Cited Violation. (See June 2, 2002, for precursor event.) (Final Significance Determination for Green and White Findings and a Notice of Violation at Peach Bottom, IR-50-277/02-07; 50-278/02-07).
This was the twenty-seventh Non-Cited Violation since June 1998.
Exelon's total cost avoidance, i.e., "credit" for 27 Non-Cited Violations = $1 ,355,000.

December 10, 2002 - A security challenge occurred at an Exelon nuclear power plant outside of Chicago.
"BRAIDWOOD -- A crazed Chicagoan, swearing to be an extraterrestrial alien, crashed his car through the gates of the Braidwood nuclear facility late Monday before speeding away only to be arrested for reckless driving in Wilmington minutes later.
...Continued on the following page... No injuries resulted. Metta said the intruder is alleged to have penetrated the parking area by crashing through closed gates, flashing past a plant checkpoint and then doing "donuts" in the parking lot. ("The Daily Journal", Kankakee IL.)" (See January 9 and December 20, 2002, for related inc ident s .)

December 12, 2002 - TMI sirens malfunctioned in Cumberland and York counties. In Dauphin County, 28 sirens malfunctioned due to the "inadvertent" discharge of the "space bar" by a computer operator. (Refer to June 22, August 15 and October 5-9, 2001 and January 11, March 3 2002, for related problems.)

December 20, 2002 - Another security challenge occurred at an Exelon nuclear power plant outside of Chicago:
"BRAIDWOOD -- She was the second driver to breeze past the guard station at Braidwood's nuclear facility in the span of a week.
"But its unclear if the trespasses arrest of Wilmington's Christina Staley, Tuesday, will result in changes to the nuclear generating station's security apparatus.
"Neal Miller, station director, noted that Ms. Staley, 31, had apparently become disoriented and was looking for some place to turn around when she drove past the security at 9 a.m."
("The Daily Journal", Kankakee IL.)"
(See January 9 and December 10, 2002, for related incidents.)
December 13, 2002 - A security challenge occurred at a nuclear facility north of Peach Bottom, on the Susquehanna River
"At 1450 EST on 12/13/2002, Susquehanna LLC Main Control Room received a request for additional information from the Pennsylvania Emergency Management Agency (PEMA). PEMA received rumors that a HAZMAT team had been dispatched to Susquehanna in response to a spill associated with a potential sabotage event.

December 17, 2002 - "...Unit 2 power was reduced to approximately 16 percent to facilitate leak repairs on the Caldon LEFM flow measurement system. After repairs, Unit 2 returned to 100 percent power in the afternoon of December 21" (IR 50-277/02-06; 50-278/02-06). (See April 30 - May 11, 2003, for a similar problem).

December 21, 2002 - An LER was recorded after "Unit 2 automatically shutdown from 100% power when the main steam isolation valves closed due to a Group I Primary Containment Isolation System (PCIC) actuation" (IR 50-277-03-02; IR-50-278/03-02). "For example, on Dec. 21, 2002, a Peach Bottom Atomic Power Station Unit 2 electro-hydrolic control system circuit card failure triggered a scram, according to the NRC's report. That system controls the wide-range speed control of the turbine, Sheehan said."In other words," Sheehan said, "it serves as a sort of high-tech throttle for the plant's turbine, thereby controlling the plant's power output.""On Dec. 22, 2004, the NRC report said, another part of that same system malfunctioned, causing a loss of reactor pressure and forcing a scram." ("York Sunday News", March 13, 2005)

January 28, 2003 - An NCV was issued relating to Exelon's failure to correct and maintain "preventative maintenance activities and procedures on critical, safety related ventilation dampers since 1988...A contributing cause to the length of time that Exelon did not identify this issue was related to the Problem Identification and Resolution crosscutting area. Peach Bottom plant personnel did not identify the lack of preventative maintenance for safetyrelated dampers following the identification of excessive stroke times...in June 2000 or...failure to stroke on June 16, 2002" (IR 50-277-02-06; IR-50-278/02-06).

This was the twenty-eighth Non-Cited Violation since June 1998. Exelon's total cost avoidance, i.e., "credit" for 28 Non-Cited Violations = $1,405,000.

February 11, 2003 - A Severity Level IV violation was issued by the NRC. Exelon made changes to their emergency plans without prior NRC approval.
"The finding was determined to be more than minor as its significance was related to the impact it would have on the mobilization of the emergency response organization and preclude offsite agencies from being aware of adverse conditions on site" (NCV 50-277; IR-50-278/03-006-01);
This Violation was classified a Non-Cited Violation. This was the twentyninth Non-Cited Violation since June 1998. Exelon's total cost avoidance, i.e., "credit" for 29 Non-Cited Violations = $1,455,000.

February 17, 2003 - PEACH BOTTOM-2 WAS REDUCED TO 45% POWER AFTER A RECIRCULATION PUMP tripped. Exelon spokesman Dave Simon said the trip occurred Feb. 17 at 6:48 a.m. The root cause of the trip has not yet been determined, he added. Simon declined to say how long the unit is expected to be operating at the reduced power level. Peach Bottom-2 was at full power prior to incident (Reuters.) The plant ramped up to full power by February 20, 2003. Reuters: Exelon's Pa. Peach Bottom 2 nuke drops to 41 pct Tuesday February 18, 8:25 am ET NEW YORK, Feb. 18 (Reuters) - Exelon Nuclear's 1,110 megawatt Peach Bottom 2 nuclear unit in Pennsylvania was at 41 percent power early Tuesday, down from full power on Friday, the U.S. Nuclear Regulatory Commission said in its power reactor status report. It was not immediately known why the unit, located in Delta, Pennsylvania, had been reduced. Meanwhile, the adjacent 1,110 MW Unit 3 continued to operate at full power on Monday. The NRC did not issue a reactor status report on Monday due to the U.S. Presidents Day holiday. Exelon Nuclear is a unit of Exelon Corp. of Chicago.

April 12-15, 2003 - At Unit-2, "an automatic reactor shutdown occurred due to high reactor pressure after the 'D' outboard main steam isolation valve (MSIV) collapsed. The MSIV closes as a result of a failed instrument line valve. Unit 2 returned to 100% power on April 15, 2003". On April 12, 2003, "Unit 2 unexpectedly shut down when a single main steam isolation valve failed to close, based on a broken air-supply line. Exelon concluded that the valve's air tubing was vulnerable to a fatigue failure."While the plant did inspect more than 200 pneumatic lines linked to airoperated valves on both Unit 2 and Unit 3, the review did not take into account similar equipment such as instrument lines, according to the report" ("York Sunday News", March 13, 2005.)- April 19, 2003 - A Green Non-Cited Violation was issued "when approximately 25 minutes into a planned load endurance test run for the E2 EDG, a small fire occurred on the EDG manifold" (IR 50-277-200-3003; IR-50-278/200-3003). This was the thirtieth Non-Cited Violation since June 1998. Exelon's total cost avoidance, i.e., "credit" for 30 Non-Cited Violations = $ 1 , 505 ,000.

April 23, 2003 - A Non-Cited Violation was issued for problems associated with the EDG cardox system on July 23, 2002.
This was the thirty-first Non-Cited Violation since June 1998. Exelon's total cost avoidance, i.e., "credi t" for 31 Non-Cited Violations = $1 ,555,000.

April 23, 2003 - A Non-Cited Violation was issued for problems associated with emergency lighting units from November 6, 2002 through March 30, 2003. Eight-hour support batteries for three areas were not provided, i.e. Unit 2 RHR room, Unit 3 RHR room and Unit 3 RB "south isolation valve room." (IR 50-277-03-02; IR-50-278/03-02).

This was the thirty-second Non-Cited Violation since June 1998. Exelon's total cost avoidance, i.e., "credit" for 32 Non-Cited Violations = $1,610,000.

April 30 - May 11, 2003 - Unit-2 power "was reduced to approximately 30 percent to facilitate repairs to the Caldon leading edge flow meter (LEFM) system and for power suppression testing, to identify a leaking fuel assembly. During power ascension to approximately 85 percent, on May 6, following repairs to the Caldon LEFM system and after the leaking fuel assembly was identified and the adjacent control rod was inserted and de-energized, the #3 main turbine control valve started oscillating. Unit power was reduced to approximately 40 percent to facilitate repairs to the main turbine control valve. On May 11, 2003, Unit 2 returned to 10o percent power after the #3 main turbine control valve was repaired" (IR 50-277-200-3003; IR-50-278/200-3003). (See December 17, 2002, for a similar problem).

May 8, 2003 --The NRC RENEWED THE OPERATING LICENSES FOR PEACH BOTTOM-2 AND -3 FOR AN additional 20 years, the agency said today. The licenses will now expire on August 8, 2033 for unit 2 and July 2, 2034 for unit 3. Exelon had submitted the license renewal application on July 2, 2001 (Platts, Nuclear News.)

May 8, 2003 --EXELON LOWERED POWER AT PEACH BOTTOM-2 TO FIX A TURBINE CONTROL VALVE. The problem was discovered at around 3 p.m. yesterday as the unit was powering back up following completion of power suppression testing, company spokesman Dave Simon said. The unit had been operating at around 61% since April 30 while the power suppression testing was being conducted. It reached as high as 86% before being lowered to 42% to repair the control valve. Simon declined to say how long the repairs would take or when the unit would be returned to full power (Platts, Nuclear News.)

May 13, 2003 - During a surveillance test, technician discovered a " wire for the station power supply" was broken. (IR 50-277-03-02; IR-50-278/03-02).

This was the thirty-third Non-Cited Violation since June 1998. Exelon's total cost avoidance, i.e., "credit" for 33 Non-Cited Violations = $1,665,000.

SCRAM: APPENDIX "R" ISSUE AT PEACH BOTTOM 3

- "On May 14, 2003, at approximately 0410, the shift supervisor determined that the Alternate Shutdown Panel on Unit 3 was not operable following discovery of a de-energized power supply. The panel provides the capability to maintain a safe shutdown path for a fire in the cable spreading room, main control room or main control room fan room. Therefore, operators would have been prevented from implementing required actions for a fire in those areas. The apparent cause of the loss of power was a broken wire, which was discovered during routine testing of the panel.

"Power was restored to the Alternate Shutdown Panel at approximately 1030 on May 14, 2003 and further investigation is in progress to determine the cause of the broken wire and full extent and effect of the de-energization of the panel." (U.S. Nuclear Regulatory Commission Operations Center, Event Reports For 05/14/2003 - 05/15/2003.)

"Pa. Nuclear Operator Found Drunk on Job"

May 14, 2003 - An employee at two Pennsylvania nuclear power plants has been suspended for being intoxicated on the job, according to a Nuclear Regulatory Commission report. The employee tested positive as being under the influence of alcohol during a random May 14 drug test at the Limerick Generating Station, according to the report. The test was given at 9:45 a.m., when the employee had already been at work for several hours, the report stated. ...Continued on the following page...The employee had been licensed to operate reactors at the Limerick plant in Montgomery County and the Peach Bottom plant in York County before being suspended by Exelon Nuclear, officials said. The NRC considers nuclear workers with a blood-alcohol content of 0.04 or above to be intoxicated. The state of Pennsylvania considers drivers with a 0.10 reading to be intoxicated and unfit to drive. The NRC is considering whether to issue the company a violation for the incident or revoke the operator's license. (See November 14, 2003, for a related d e v e l o pme n t .)

May 21, 2003 --EXELON'S FORMER CHIEF EXECUTIVE MADE THE TOP 10 LIST OF BEST-PAID U.S. energy executives for 2002, according to a compilation by the Platts Energy Business & Technology (EB&T) magazine. Corbin McNeill, Jr., the ex-chairman and co-CEO of Exelon Corp. had a compensation package of nearly $29.8- million last year, making him the fourth highest paid CEO out of the 250 executives that were examined. McNeill's 2002 package included a severance payment and benefits from a pension benefit plan from PECO Energy. He retired from Exelon in April 2002. The highest-paid executive in 2002, at $46.6-million, was Charles Watson, former CEO of Dynegy Inc., the EB&T listing shows. The survey, which will be published in the June issue of EB&T, considered the executives' salary, bonuses, restricted stock awards, underlying options, value of options exercised, long-term investment pool pay outs, and any other compensation. (See July 9, 2003, for staff cuts).

May 22, 2003 - The NRC identified a Green violation relating to Appendix R, i.e., fire protection. The NRC deemed the issue as being of "very low safety significance" (IR 50-277-03-009; IR-50-278/03-009).

This was the thirty-fourth Non-Cited Violation since June 1998. Exelon's total cost avoidance, i.e., "credit" for 34 Non-Cited Violations = $1 ,720,000.

May 22, 2003 -- THE PENNSYLVANIA NATIONAL GUARD IS INCREASING ITS PRESENCE at the state's nuclear plants, Gov. Edward Rendell (D) announced yesterday. Since shortly after the Sept. 11, 2001 terrorist attacks until the end of last month, Pennsylvania had had a 24-hour Guard presence at the plants, but then had switched to random, unannounced security patrols, Rendell spokesman Michael Lukens said. But under Rendell's order, which went into effect yesterday, the two elements are being combined, Lukens said. ...Continued on the following page...He said the order would remain in effect "indefinitely," and the governor's office would continue to assess it. Rendell's announcement said he took the action in response to the recent elevation of the national threat level to orange, but Lukens said the state's assessment of the need for the Guard would not necessarily be tied to future changes in that threat level. (Platts Nuclear News Flashes. (See October 6 & 17, 2001, January 30, 2002, and November 2, 2002 for related incidents).

May 28, 2003 -A License Event Report was generated after "licensed operators were notified that approximately 4 inches of water [170 gallons] was discovered at the bottom of the 'A' Standby Gas Treatment (SBGT) filter plenum during the performance of annual surveillance (IR 50-277/2003004; IR-50-278/2003004).

June 13, 2003 - LOSS OF BOTH OFFSITE POWER SOURCES TO TECHNICAL SUPPORT CENTER: "During severe thunderstorms in the area power was lost to the onsite technical Support Center (TSC) for approximately 90 minutes. These storms caused both offsite power sources to the TSC to deenergize at 2021. Grid operators began restoration activities immediately and power was restored to the facility at approximately 2200. Investigation is in progress for the cause of the line tripping."

The licensee notified the NRC Resident Inspector.

June 17, 2003 - Pensions: Utility Obligations Add Up,
By Ken Silverstein Director, Energy Industry Analysis Utilities may get socked again. Already, stock values and credit ratings have taken a hit because of the failure to mitigate risks to their unregulated operations. Now, their credit status may get cut even more, given the level of "unfunded" pension liabilities.
If the money in the pension plan to pay retirement obligations falls short, then a "minimum pension liability" must be recorded on the financial statements. In lay terms, it means that if a company were to be liquidated today, then it would be compelled to pay up. The liability recorded could therefore impede the debt-tocapital ratio, which could harm credit quality and even trigger violations of covenants. And while regulated utilities have a chance to recover such costs from their customers, many are now in the midst of rate moratoriums and cannot seek recovery, says Steven Fleishman, analyst with Merrill Lynch in New York City. Others would prefer to avoid a rate case, given that regulators may revisit their entire rate structure and reduce their allowable returns, he adds.
...Continued on the following page...Those with the largest underfunded pensions at year-end 2002, says Merrill Lynch, include Exelon ($2.4 billion), FirstEnergy Corp. ($977 million), Public Service Enterprise Group ($837 million) and American Electric Power ($788 million.) Companies with the largest underfunded pensions as a percentage of equity market value, include CMS Energy (60 percent), Sierra Pacific Resources (30 percent), AES Corp. (29 percent) and CenterPoint Energy (17 percent). FirstEnergy, for instance, has said that its pension liabilities had forced it to cut its 2003 earnings picture. Profits, it says, will grow by 4-5 percent-not the 7-8 percent that it had projected. DTE Energy, meanwhile, said that its pension expenses would be $50-$55 million higher in 2003 than in 2002. (See December 3, 2003, for related GAO Study).

July 9, 2003 --EXELON HAS RESTRUCTURED ITS NUCLEAR OPERATIONS BY ELIMINATING regional operating groups in favor of a single organizational unit. The restructuring was made public today in an NRC Weekly Information Notice, but was announced internally to Exelon employees June 23. As part of the restructuring, Chris Crane was named chief operating officer of Exelon Nuclear, William Levis vice president of mid-Atlantic operations, and Chip Pardee senior vice president of nuclear services. Also, Robert Braun will replace the retiring Joel Dimmette as vice president of nuclear operations. The changes will become effective by Aug. 1, said Exelon spokeswoman Ann Mary Carley. She said that when Exelon Nuclear was formed in 2002, it set up the regional operating groups to accommodate the nuclear organizations of the former PECO Energy and Commonwealth Edison (ComEd), as well as AmerGen, a joint venture between Exelon and British Energy. Exelon was created by the merger of PECO and ComEd parent Unicom Corp. Over time, the two regional groups' policies and procedures have aligned and all 10 Exelon plants are now using the same policies and procedures, Carley said (Also refer to May 21, 2003 --EXELON'S FORMER CHIEF EXECUTIVE MADE THE TOP 10 LIST OF BEST-PAID

U.S. energy executives for 2002, according to a compilation by the Platts Energy Business & Technology (EB&T) magazine.)

July 11, 2003 - The NRC conducted a supplemental inspection to "assess the licensee's evaluation and corrective actions regarding the...June 2, 2002, carbon discharger event". The NRC diluted its previous "White" finding and noted the event "will only be considered in assessing plant performance through the period concluding at the end of the second calendar quarter of 2003..." [In other words, 20 days from the NRC's promulgation the event becomes a "nonevent".] (See November 26, 2002 additional data.) (IR Supplemental Report 50-277-03-11; 50-278/03-011).

This was the thirty-fifth Non-Cited Violation since June 1998. Exelon's total cost avoidance, i.e., "credi t" for 35 Non-Cited Violations = $1 ,775,000.

July 16, 2003 - The NRC's Office of Investigation's (OI) concluded that Exelon was in violation of a License Amend met Restriction that requires notification when a reactor operator (RO) medical status changes. Such a change occurred to an RO on September 13, 2001, and the forenamed operator returned to work between April and December 2002 without notifying the NRC about the reactor operator Fitness for Duty in the control room.

The NRC's investigation began on January 3, 2003. "After careful consideration of the information developed during the investigation, the NRC has concluded that a violation of NRC requirements occurred" (PBAPS, NRC O&I No. 1-2003-002).

This was the thirty-sixth Non-Cited Violation since June 1998. Exelon's total cost avoidance, i.e., "credit" for 36 Non-Cited Violations = $1 ,830,000.

July 22-29, 2003 - Unit 2 experienced an automatic reactor shutdown "due to generator lockout from foreign material causing a short in the bus duct. Unit 2 returned to 100% power on July 29, 2003." (IR 50-277/2003004; IR-50-278/2003004).

On July 22, 2003, "Unit 2 shut down when a piece of broken fan belt entered the reactor's isophase bus duct cooling system. Exelon found that a design weakness existed and decided to install debris guards that would prevent beltmaterial from entering the fan suction."

"Despite Exelon's intention to install fan belt guards within 30 days, the corrective action took two months "with no rationale provided for the delay," according to the inspection report" ("York Sunday News", March 13, 2005).

July 23, 2003 - PEACH BOTTOM-2 REMAINED DOWN TODAY AFTER TRIPPING AUTOMATICALLY yesterday due to an actuation of the main generator protective relay, Exelon spokeswoman Dana Fallano said. She said Exelon is investigating the root cause of the actuation (Source: Platts, Nuclear News) .

July 24, 2003 - The NRC identified a Green violation relating to the inoperability of 'A' train was inoperable between November 200s through may 28, 2003 (IR 50-2772003003 IR-50-278/2003003). This was the thirty-seventh Non-Cited Violation since June 1998. Exelon's total cost avoidance, i.e., "credit" for 37 Non-Cited Violations = $1,885,000.

July 29, 2003 - 11:55:05 AM EST Peach Bottom plant back to full power; Shutdown of nuclear generating unit 2 last week cited as non emergency By LANCASTER INTELLIGENCER JOURNAL
The Peach Bottom Atomic Power Station returned to full power today after an outage of one of its two power generation reactors last week.
Peach Bottom's Unit 2 reactor returned to service at about 10:15 a.m. Saturday. As of yesterday, the unit was operating at approximately 90 percent of capacity, said Dana Fallano, spokeswoman for Exelon Nuclear, which owns the plant. Unit 2 shut down one week ago after generator problems forced an automatic shutdown.

Neil Sheehan, spokesman for the Nuclear Regulatory Commission, said all safety systems functioned properly during the shutdown and any radioactive steam that could have been released was contained and isolated in the reactor vessel. "It seems like a pretty straightforward event," he said.
Exelon reported the shutdown to the NRC at 5:30 p.m. July 22. The commission classified the shutdown as a "non emergency event."
According to Exelon's event report, Unit 2's generator malfunctioned at 1:45 p.m. that afternoon while operating at full power. With no way to output electricity, the plant's main turbine tripped off, which then triggered an automatic reactor shutdown. Exelon employees had no firm answers last week on what caused the generator to malfunction, Sheehan said. Yesterday, Fallano said the generator's protective electronic relay system activated after sensing some type of movement. She said the company is still investigating what type of movement that was. NRC reaction: Sheehan said it's unlikely the NRC will send a team of inspectors to investigate because the problem occurred in the generator, not the reactor vessel, and the shutdown appears to have gone smoothly.

The utility may be concerned, Sheehan said, about losing a reactor during heavy summer demand for electricity. Fallano declined to discuss how much revenue was lost, calling it private, competitive information. When both Peach Bottom reactors are running, the power station supplies enough electricity for 2 million homes.

...Continued on the following page...The event marked the second shutdown at Peach Bottom's Unit 2 in seven months. On average, the nation's 103 commercial reactors automatically shut down only once every other year, according to the NRC.
On Dec. 21, computer failure closed valves that direct steam from Peach Bottom's Unit 2 to the main turbine that generates electricity. The reactor automatically shut down to avoid a steam buildup.

The NRC sent a team of inspectors to the plant and cited Exelon for two safety violations involving human errors and equipment problems that occurred during that shutdown.

Staff writer Charlie Young contributed to this report.

July 30, 2003 - EXELON REPORTED SECOND QUARTER 2003 EARNINGS OF $402-MILLION, an 8.9% increase over the $369-million earned in the same quarter one year ago. The company said an increase in sales, lower interest expense, and lower depreciation and amortization offset weather-related decreases in electricity deliveries and lower energy margins at Energy Delivery. Exelon reported its nuclear fleet, excluding the plants in the AmerGen joint venture (Clinton, Oyster Creek and Three Mile Island-1) generated 29,619 gigawatt-hours in the second quarter, compared to 28,776 GWH in the second quarter of 2002. Capacity factor of the Exelon fleet, including the AmerGen plants, improved to 94% during the second quarter this year from 92.1% in the second quarter last year, Exelon reported. AmerGen is a joint venture between Exelon and British Energy (Source: Platts, Nuclear News) .

August 8, 2003 - The NRC identified a Green violation "concerning the failure to properly correct an equipment deficiency that subsequently resulted in a challenge to the plant and operators. Specifically, a solenoid associated with a reactor feed pump turbine (RFPT) overspeed trip device exhibited degradation during RFPT overspeed testing on two occasions [September 27 and November 27, 2001], however, your staff failed to determine the root cause for this problem until a third problem occurred that resulted in a RFPT trip and plant transient" (IR 50-2772003012 IR-50-278/2003012).
This was the thirty-eighth Non-Cited Violation since June 1998. Exelon's total cost avoidance, i.e., "credit" for 38 Non-Cited Violations = $1 ,940,000.

August 14, 2003 - "...the fifth stage feed water heaters were removed from service for end of cycle coast down." (IR 50-277-200-3004; IR-50-278/200-3 0 0 4) . Exelon Corp debt ratings unchanged by Sithe deal-S&P

(NEW YORK, Aug. 18 - Standard & Poor's Ratings Services said today that its ratings on Exelon Corp. (nyse: EXC - news - people) (A-/Stable/A-2) and its subsidiaries will not be affected by the company's announcement that it will sell 50% of its equity interest in Sithe Energies Inc. Further, subsequent full sale of Sithe, which remains a distinct possibility given the put and call options attached to Sithe ownership, would not affect Exelon's ratings...Exelon's announced equity interest sale demonstrates the company's intention to sell off the disappointing merchant assets it acquired several years ago, a positive for credit quality. However, the fact that Exelon recorded a $200 million writedown related to its original 49.9% investment in Sithe demonstrates the inherent risk associated with the remaining high-risk portion of this business.

(See August 29, 2003 for a related development).

August 24, 2003 - “The fourth stage feed water heaters were removed from service [for end of cycle coast down]”. (IR 50-277-200-3004; IR-50-278/200-3004)

POLL: Security officers expect another blackout in 12 months

August 25, 2003 - CSO Magazine polled 382 chief security officers (CSO) and senior security executives showed 59% blamed the electric industry and not the government for the blackout of 2003.

CSOs showed their lack of confidence in the power industry and grid with 59% predicting another major blackout within 12 months. Over three-quarters said they doubt the electric industry will be modernized in five years. That percentage want a probe by an independent investigator without ties to the industry. Almost half (47%) ask that the probe's results be classified to keep terrorists from learning about US vulnerabilities.

Those surveyed included 156 whose firms felt some direct impact of the outage. Many want the federal government to expand oversight of the electric industry. "Regulations are often regarded as the necessary evil in securing the nation's infrastructure," said Lew McCreary, editor of the Framingham, Mass, publication, but he was surprised that CSOs -- traditionally anti-regulation -- are calling for increased government control in this industry, "having now been faced with a glaring example of so-called market forces at work," the editor cleverly observed.

...Continued on the following page...

The magazine did the survey online Aug 19-21, having sent an email invitation to the web-based survey to 12,200 subscribers. The 382 are the ones that met qualifications and fully completed the survey. The sample was chosen randomly and each subscriber had an equal probability of being selected. Figure a 5% margin of error, the magazine said.

Results are at www.csoonline.com/releases/ 08220385_release.html.
(Story originally published in Restructuring Today 8/25/03)

Raytheon Also Sues BNP Paribas Over Exelon Projects

August 29, 2003 - LEXINGTON, Mass. -(Dow Jones)- Raytheon Co. (NYSE:RTN - News) sued an indirect subsidiary of Exelon Corp. (NYSE:EXC - News) , as well as BNP Paribas SA , about Exelon's decision to turn over the subsidiary to its bank lenders.

Raytheon said it is "seeking to protect Raytheon's rights" in connection with the Exelon Mystic and Exelon Fore River power plant projects in Massachusetts. In a press release, the aerospace and defense company said the suit was filed in Massachusetts' Suffolk County Superior Court.

On July 29, Exelon said it planned to turn Exelon Boston Generating LLC, its indirect subsidiary, over to its bank lenders. It decided to do so after continued evaluation of Boston Generating's power-plant projects and discussions with lenders. Raytheon turned over the Mystic and Fore River projects to owner Exelon -

one in April, one in July. The projects weighed down Raytheon's balance sheet for several years.

Raytheon was forced back into the construction business to complete the projects in Weymouth and Everett, Mass., after Washington Group International Inc. (NasdaqNM:WGI I - News) filed for bankruptcy. Representatives from Exelon and BNP Paribas were not immediately available to respond to the lawsuit. Raytheon named the Mystic and Fore River units as defendants as well.
...Continued on the following page...Raytheon said that since Exelon's announcement, Raytheon has continued to perform final close-out work on the projects. Raytheon said it seeks to "obtain adequate assurances of payment" and protect its rights under its support agreements. Raytheon, through a subsidiary, was the original contractor of the plants.
It sold that subsidiary to Washington Group in 2000, but got project responsibility back in a settlement from Washington Group after Washington filed for bankruptcy in 2002.Exelon seeks to transfer ownership of Boston Generating without the subsidiary filing for bankruptcy. Exelon has about $700 million invested in Boston Generating. Exelon has said it plans to spend nothing further on Boston Generating outside of limited administrative and operational services.Therefore, Raytheon is seeking a declaratory judgment and injunction from the court that will assure it is paid by either Exelon, its subsidiaries and subunits, or its lenders.Exelon has refused to refund about $36 million in prepaid liquidated damages that Raytheon advanced, the court papers said. Raytheon also said that the defendants have no right to draw upon about $73 million in letters of credit that the defense contractor posted for them. Raytheon said it posted the credit to ensure the performance of its contractual obligations. Throughout the court filing, Raytheon says that it spent, during the lifetimes of its guarantee agreements, more than $1 billion for the benefit of Exelon, its subsidiaries and subunits, and the lenders. BNP Paribas' alleged role in the matter dates back to January 2001, when a former owner of the plants, Sithe Generating, secured financing from the French bank to pay for the construction of the Mystic and Fore River facilities. After Washington Group abandoned work on the facilities, BNP and other lenders insisted on credit facility changes. One of those changes was that BNP, court papers indicated, would provide Raytheon with prompt written notice of any continuing events of defaults under the credit agreement. Raytheon said that, from November 2002 -- when Exelon bought Sithe -- to the day Exelon announced it was handing the units over to its lenders, it never received any notices from BNP Paribas. Because of the lack of notice, Raytheon claims it has continued to spend money in good faith and has been damaged by BNP's alleged omissions.
-Thomas Derpinghaus; Dow Jones Newswires; 201-938-5400.
(See August 29, 2003 for a related development). The commission investigated a loss of power at

Peach Bottom's power station in May

By SEAN ADKINS Daily Record

September 4, 2003 - For about nine days in May, an undetected broken wire caused a loss of power to a redundant control station for Peach Bottom Atomic Power Station Unit 3.

A failure to observe work order test instructions after maintenance on the panel prevented plant technicians from immediately discovering the broken wire, according to a U.S. Nuclear Regulatory Commission report.

Damage to the power supply wire occurred during maintenance to the highpressure coolant injection alternative control station — a system used to shut down the plant if the operators are forced to leave the main control room because of a fire, said NRC spokeswoman Diane Screnci.

While the violation is under commission review, the incident did not pose a safety threat since the plant repaired the wire and restored power to the back-up station on May 14, Screnci said.

"There are other ways you could shut down the plant even if you don't have the station active," she said.

Depending on the commission's findings, the infraction could mean additional plant inspections.

In June, Peach Bottom Atomic Power Station was the subject of a supplemental NRC inspection for a violation committed the year before.

Last year, a light bulb dropped from the ceiling onto a circuit board and caused the plant's fire-suppression system to discharge carbon dioxide [Refer to July 11, 2003] into the E-3 emergency diesel generator room in the Diesel Generator Building.

The supplemental inspection found that the plant had taken the proper corrective actions and the power station could return to a routine inspection s chedul e .

While the plant showed that its fire-suppression system was in working order, a malfunction in one of its diesel generators garnered a non-cited commission violation of very low safety significance.

Continued on the following page... In June, NRC inspectors found that Exelon technicians had not adequately tightened the engine top cover flange joint bolts of an emergency diesel generator during a maintenance procedure.

As a result, lube oil leaked from the joint and caused a small fire on the exhaust manifold during a test.

During that same time period, Three Mile Island Unit 1 violated an NRC reporting requirement.

In June, NRC inspectors found that, on three instances, TMI officials found potentially disqualifying medical conditions among its licensed operators but had not reported them to the NRC within the required 30 days.

TMI requested its doctor to confirm with the patient's physicians, which extended past the 30-day NRC reporting period.

Two units at nuke plant shut down; grid disturbance cited

September 15, 2003 - An electrical disturbance on the power grid cut off incoming electricity at the Peach Bottom nuclear power plant and caused both reactors to shut down automatically early Monday, Exelon Nuclear officials said.

Plant officials declared an "unusual event" just after 2:30 a.m.
The plant's four emergency backup diesel generators provided emergency power for about an hour, said Exelon spokesman David Simon. One of the generators malfunctioned, and then another backup source of power was used to power vital equipment, such as lights and emergency feed water pumps, until power was restored later in the morning, Simon said.

... PJM Interconnection, the company that operates the power grid in the Mid-Atlantic, said it was investigating the grid disturbance. PJM spokesman Ray Dodter said the company couldn't yet say what caused the disruption.

Unit-2 was operating at 100% power, and retuned to full power on September 25, 2003.
Unit-3 was operating at 91% power, and remained shut for the 3R14 refueling outage.

September 15, 2003 --THE U.S. COAST GUARD PROPOSED ESTABLISHING A PERMANENT SECURITY ZONE on the waters adjacent to Peach Bottom. According to a notice of proposed rulemaking published in yesterday's Federal Register, the zone "would protect the safety and security of the plant from subversive activity, sabotage, or terrorist attacks initiated from surrounding waters. This action would close water areas around the plant." A temporary final rule issued June 4 established the security zone on the Susquehanna River by restricting any person or vessel from entering or navigating the security zone without Coast Guard permission. The Coast Guard said in the notice that it wants to make the security zone permanent. Comments on the proposed rule are due by Nov. 14. (Source: Platts, Nuclear News) .

October 24, 2003 - Exelon Corp. Posts Quarterly Net Loss of $102 Million - Oct. 24--Commonwealth Edison parent Exelon Corp. reported solid operating profit in the third quarter, but special items -- including a mammoth $573 million charge to write off a disastrous investment in East Coast electricitygenerating projects -- pushed the holding company's bottom-line results into the
red. In the latest quarter, Exelon reported a net loss of $102 million, or 31 cents a share. (Knight Ridder Tribune Business News.)

October 27, 2003 -NRC AGREED TO RELAX TWO REQUIREMENTS IN AN APRIL ORDER ON SECURITY FORCE personnel working hours. NRC Office of Nuclear Reactor Regulation Director James Dyer Oct. 23 issued notices to all reactor licensees that the agency would allow shift turnover time to be excluded from total group work hours that must be tracked. The NRC staff had wanted accounting of all hours worked for tracking overtime, which it says could lead to worker fatigue, but now agrees with the industry that tracking the extra time does impose some additional burden. Industry officials argued the shift change time is usually not more than 15 minutes. The second relaxation allows licensees to increase the work hours during force-on-force exercises from a 48- to 60-hour per week average. Dyer said the staff understands that the simulated exercises put additional demands on the security guards but the mock attacks extend only for a short period of time (Platts, Nuclear News) .

October 29, 2003 --OPERATING POWER REACTOR LICENSEES MUST BE IN FULL COMPLIANCE TODAY with NRC's April 29 order imposing measures to control the work hours for security force personnel. The industry had asked for relief in two areas of the order, and the NRC staff recently approved those requests. The industry will not have to track the time it takes for guards to change shifts in the overall group work hours and will be allowed a 60-hour limit--up from the usual 48 hours per week--in scheduling guards during the week of a force-on- force exercise. Two other April orders, one on security officer training and the other on changes to the design basis threat, require full implementation by Oct. 29, 2004. A Nuclear Energy Institute official said at a conference in Arlington, Va. today that the industry plans to ask the NRC to rescind the three orders after licensees adopt the requirements in their security plans (Platts, Nuclear News) .

November 3, 2003 - S&P placed Exelon on credit watch after the Company announced it wanted to buy Illinois Power from Dynergy. or $2.2 billion, if Illinois legislators grant it single-digit rate increases. The deal was canceled after Exelon determined it could not count on rate increases.

November 4, 2003 - NRC inspectors identified three, "Green” non-cited violations and Severity Level IV violation “associated with a lack of records to support changes made to the emergency plan” (IR 50-277-200-3004; IR-50-278/200-3004).
The Severity Level IV Violation, also Non-Cited, involved changes to Exelon’s Standard Emergency Plan, including Limerick, Peach Bottom and Three Mile island. Exelon changed “emergency plan commitments without documentation” which subsequently impacted “the NRC’s ability to perform its regulatory function...”
Continued on the following page...The three other “Non-Cited” violations include different aspect of plant
operations and training:

Licensed Operator Requalification “Green. A non-cited violation...was identified regarding the licensee’s method used to reactivate senior operator licensees to support refueling. The operators were reactivated without the required direct supervision being present during the shift under-instruction

item. The Limited Senior Reactor Operator (LSRO) Requalification Program for Fuel Handlers is a dual site operator license program that applies to both Limerick and Peach Bottom sites."
Finding 1 -Unit 2 Reactor Core Isolation Coolant System During Unit 2 S c r am "...Exelon did not adequately correct a significant condition adverse to quality identified during a December 21, 02 scram, associated with the inoperability of the Unit 2 reactor core isolation cooling (RCIC) pump in the automatic flow control mode"
Finding 2 -Unit 2 Main Steam Line High Temperature Switch "..during the period of July 2001 through July 2003, Exelon did not adequately correct a condition adverse to quality, specifically a high Unit 2 steam tunnel temperature condition that was not representative of a steam leak".

This was the thirty-ninth, fortieth, forty-first and forty-second Non-Cited Violat ion since June 1998. Exelon's total cost avoidance, i.e., "credi t" for 42 Non-Cited Violations = $2, 160,000.

November 7, 2003 - "NRC: NRC Appoints New Senior Resident Inspector at the Peach Bottom...Craig Smith is the new senior resident inspector at the Peach Bottom Atomic Power Station in Delta, Pa. The two-unit site is operated by Exelon. Most recently, Mr. Smith was a resident inspector at the Three Mile Island nuclear plant in Middletown, Pa." ("NRC Press Release").
However, Eric Epstein, Chairman of TMI-Alert, noted: "Craig Smith was at TMI for five years and hid on the Island except for annual appearances." Mr. Epstein pointed to Mr. Smith's last appearance before the public at the NRC's Annual ROP Assessment meeting on Wednesday, April, 9, 2003.
Continued on the following page...Mr. Smith stated that the number of employees at TMI was 529. When the
NRC was apprised that they were off by 114 employees, they reassured the community it didn't matter how many people worked at TMI based on the color code, PI sequence and late hour. Local residents persisted, and told the NRC that Performance Indicators for Non- Performance does make sense, and we're still old fashioned enough to prefer Zero Tolerance to color-coded lollipops.
- November 8, 2003 - U.S. Warns of Al Qaeda Cargo Plane Plot -
WASHINGTON (AP) -- The latest warning from the Homeland Security Department that al-Qaida may be plotting an attack is renewing calls for stricter security on cargo planes.
The department advised law-enforcement officials Friday night of threats that terrorists may fly cargo planes from another country into such crucial U.S. targets as nuclear plants, bridges or dams, Homeland Security spokesman Brian Roehrkasse (By THE ASSOCIATED PRESS/Published: Filed at 4:29 p.m. ET).

November 13, 2003 - "Exelon Nuclear's Peach Bottom-2 was forced to shut down 196.3 hours due to off-site voltage fluctuations in the elcxtrical grid" (Nucleoniocs Week, p. 17.)yees screened positive for the illegal drug ˜ the largest single six-month j

On drugs, and on the job, Between July 1999 and December 2002, 143 workers at local power plants tested positive for drugs or alcohol.

By SEAN ADKINS , Daily Record staff (November 14, 2003)

Late in the afternoon of Sept. 24, 1999, a Three Mile Island security officer checked a tip about a short-term contractor smoking marijuana on the job. Officer Darlene Ranck escorted George Lonnie McDaniel, 27, to TMI's security office to be questioned for violating the plant's Fitness-for-Duty Program.

Ranck and Officer Greg DeHoff asked McDaniel to empty his pockets.

The Jessup, Ga., resident pulled a small plastic bag of marijuana from his pocket, and plant security officers called the Pennsylvania State Police, according to an affidavit filed with District Justice David H. Judy in Dauphin County.

McDaniel's job at TMI did not grant him access to vital areas of the plant.

Currently, Dauphin County has a fugitive warrant out for McDaniel's arrest. He could not be reached for comment for this article.

Between July 1999 and December 2002, 143 workers and short-term contractors at Three Mile Island and Peach Bottom Atomic Power Station tested positive for drugs or alcohol, according to biannual Fitness-for-Duty reports.

The York Daily Record obtained the reports from the U.S. Nuclear Regulatory Commission through a Freedom of Information Act request.

Drugs listed in the reports include marijuana, cocaine, opiates, amphetamines and alcohol. All the workers tested were people who had or were applying for unescorted access to vital areas of the plants.

Many were short-term workers, such as McDaniel. They travel the nation, from power plant to power plant, to work when reactors are shut down for refueling.

Continued on the following page...State Rep. Bruce Smith, R-Dillsburg, said he was disturbed by the number of positive drug tests reported by TMI officials. “There is no excuse or any way to defend substance abuse at a nuclear power plant,” he said. Smith said he plans to contact the NRC and acquire the plant's Fitness-forDuty reports for his own records. A Daily Record investigation found: XB7; More people might have tested positive, but the NRC does not have a zerotolerance policy when it comes to chemical testing. The commission uses cutoff limits to screen for narcotics and alcohol. For example, the NRC’s limit for alcohol is a blood-alcohol content of 0.04 percent. That is equivalent to three 12-ounce beers in an hour for a 200-pound man. XB7 ; Short-term contractors made up the majority of the workers who tested positive at both Peach Bottom and TMI unit 1 in Londonderry Township, Dauphin County. Short-term contractors generally handle maintenance and repairs that cannot be completed when the plant is on-line. XB7 ; Workers inability to cope with stress following the terrorist attacks may have contributed to the largest single six-month jump in marijuana use among plant workers since July 1999. For both plants, 73 people tested positive for marijuana ˜ the most of any intoxicant. Keeping fit for duty In 1989, the NRC created a policy that each plant should follow an individual fitness-for-duty program. Collecting such data helps ensure that workers complete their jobs free of any physical or mental impairment such as drugs, said Neil Sheehan, commission spokesman. Twice a year, each plant files a report with the commission that details how many workers tested

positive for legal or illegal substances. Continued on the following page...The commission examines the data for trends in drug use among plant workers, Sheehan said. “It acts as a performance indicator of a plant,” he said. If a plant reports two or more fitness-for-duty program failures, the NRC will increase its level of oversight. An example of a program failure could be a worker and plant physician working together to falsify screening results. Program failures could translate into increased inspections and possible fines, Sheehan said. In 2001, the NRC hosted a specific investigation into whether a former commission- licensed chief shift operator at the Nine Mile Point Nuclear Station in New York had deliberately provided false, inaccurate, or incomplete information on health history forms. The investigation uncovered that the operator deliberately failed to provide complete information on the forms in order to mislead an officer.
The fitness-for-duty violation case did not result in a fine, but the NRC could have issued a base civil penalty of $55,000.
Neither Peach Bottom nor TMI Unit 1 has been cited for a fitness-for-duty violation.

Test limits
Rather than have a zero-tolerance drug policy, the NRC relies on cutoff levels to test if a person has abused drugs or alcohol. For example, the NRC’s limit on marijuana is 100 ng/ml ˜ about the equivalent of smoking one joint in a week. At those levels, it is possible that a worker could endanger himself, fellow employees and the community, said Jim Beek, a public information officer for the Substance Abuse and Mental Health Services Administration.

Continued on the following page...A division of the U.S. Department of Health and Human Services, SAMHSA sets guidelines for workplace drug testing for the NRC. The level of impairment depends heavily on a persons sensitivity to a specific drug, Beek said. Since most ‘street drug’ like marijuana and cocaine are not regulated by the U.S. Food and Drug Administration, it can be difficult for experts to determine the strength of the drug, Beek said. “When someone takes a hit off of a joint, you don’t know how or when it might affect them,” he said. “They could end up losing an arm or blowing up Delta, Pa.”From her living room, Marianne Adamski of Goldsboro has a view of TMI’s water cooling towers billowing steam. She said the lack of a zero-tolerance drug policy for plant workers is , “cary.” “They should regulate it much better than that,” Adamski said. “They should be more responsible than that.”The NRC’s use of cutoff levels rather than zero tolerance is based on decades of research, Sheehan said. Studies indicate that drugs in quantities below the cutoff levels are not likely to affect job performance. For example, a plant employee who must report to work at 4 p.m. Monday and has cocktails Sunday night should not be affected by the alcohol once he reports to the plant, Sheehan said. “You might have a small amount of alcohol in your body, but based on evidence, it will not impair your ability to do the job effectively,” Sheehan said.

One expert claims a zero-tolerance drug policy does not account for human digestion and passive exposure involving marijuana. The human body produces alcohol as a process of digestion, said Robert Stephenson, head of the SAMHSA Division of Workplace Programs. That amount of alcohol is below the level of impairment but above zero, Stephenson said.
Marijuana can stick to clothes and hair, he said.
Continued on the following page...
If a person walks through a room where people are smoking marijuana, it may mean that they were exposed to second-hand smoke rather than ingesting the drug. “Zero tolerance means that we won’t tolerate one free bite of the apple,” Stephenson said.
Another hurdle that laboratories must traverse in the quest for a true zero-tolerance drug test is technology.
Many drug cutoff levels exist essentially to test how far down the screening equipment can reach, said Dr. Carla Huitt. “Much of the equipment can't accurately measure down to zero,” said Huitt, medical director of the Industrial Resource Center at Memorial Hospital. “Below the cutoff level, they are just making an assumption that the person is not impaired.”
Regardless of the equipment, doctors cannot determine how an illegal drug will affect one person compared to the next. Marijuana, the most common drug found in plant workers, can remain in the body for up to a month, Huitt said.

Fitness offenders
On a regional level, most nuclear plant workers who tested positive for drugs were short-term contractors who work the sites during refueling. Between July 1999 and December 2002, 91 short-term contractors at Peach Bottom tested positive for drugs. At TMI, 45 temporary employees tested positive. The remaining seven workers who tested positive for drugs at both power plants were licensed employees.
A licensed worker is someone who has been certified by the NRC in their job and works at the plant full time.
Continued on the following page...One reason for the unbalanced figures could be that Peach Bottom has two operating reactors that require double the manpower, compared to the needs of TMI’s lone unit, Sheehan said.Typically, plants temporarily hire hundreds of short-term contractors for repairs and maintenance when reactors are shut down for refueling. For example, short-term contractors have been involved with the installation of a reactor vessel head at TMI since Oct. 18. The plant’s unit 1 reactor is currently shut down.“There really is no need to keep a staff that size on permanently,” said David A. Lochbaum, of the Union of Concerned Scientists in Washington, D.C., a nonprofit environmental group. Power companies have the month-long outages every two years to conduct inspections, change out spent fuel rods, upgrade equipment and perform preventive maintenance that is difficult to complete while a plant is operational.
Since 1990, when the average refueling outage lasted 60 to 75 days, the industry has pushed to reduce the number of days the power plants are down, Lochbaum said. The more time a reactor is offline, the longer a plant goes

without supplying power to the electrical grid ˜ its main business. “They make their money when the plant is running,” Lochbaum said. “Plant operators began to hire additional workers to get the required repairs completed in half the time.” But more workers means more drug screenings and a greater potential for positive chemical tests, Lochbaum said.

Most of the workers who fail the plants' drug tests are new hires who are screened for the first time and have not yet been assigned to the protected area, he said.

Continued on the following page...For those workers who actively take drugs and make it to the protected area of the plant, specific safeguards exist to expose that person’s habits to s e c u r i t y .

Exelon Nuclear operates a computer program that randomly drug tests 50 percent of a plant's staff on an annual basis, said Hugh McNally, regional security manager for Exelon Generation. The process deters people from taking drugs under the assumption that a random test could take place at any time, he said. For example, the computer could randomly select a worker who was tested for drugs on Monday to be screened again on Thursday of the same week. “I could be tested three times in a year,” McNally said. “Personally, I’ve been tested twice in one week”

As part of the plant’s training process, new workers are instructed to recognize the symptoms of narcotics use and must report any changes in behavior they notice in other employees. Failure to do so could result in a worker losing his job, McNally said. “If I smell alcohol on someone’s breath,‰ he said, “I need to report it to my supervisor.”

At the drug test, a worker must list all the prescription medications he may be taking. The employee must fill a container with urine, McNally said. The worker is allowed to complete the four-minute test in a bathroom in private, but the employee is not permitted to run any water or flush the toilet. “We try to have a lot of controls in place so a person can’t beat the system,” he said. An onsite laboratory tests the samples. If a worker’s urine screens positive for drugs, the plant sends the sample to an outside laboratory for complete verification. Exelon temporarily denies the employee access to the protected area of the plant. Once the outside laboratory has confirmed the test, the plant's medical review officer makes a final determination.

Continued on the following page... The commission requires a nuclear plant to restrict a worker's access to protected areas for at least 14 days. “For most people,” Lochbaum said, “that means they lost their job. ‘The plant may request a worker complete drug and alcohol counseling before the employee can return to the plant.

Plant officials make the final determination whether to reinstate the employee’s access to the protected area or to fire the employee, McNally said. Access is automatically denied for three years if a person screens positive a second time, he said.

A failed drug test could hamper a person’s chances for a new job, Lochbaum said. Power companies enter information relating to the failed test into a national database that is monitored by all power plants.

"It's a red flag that you lost unescorted access privileges to the plant, "Lochbaum said. "If you violated their drug policy, you've kissed your job goodbye."

Spike in marijuana use

Between July and December 2001, 10 TMI workers tested positive for marijuana while 20 Peach Bottom emploump since July 1999.

By contrast, no workers at Peach Bottom tested positive for marijuana during the previous six-month period. At TMI Unit 1, three people tested positive for the drug during that period.

Aside from fall refueling outages that require more workers, the jump in drug abuse may be attributed to stress. The Sept. 11, 2001, terrorist attacks happened during the six months when the spike occurred.

Continued on the following page...Generally, an unstable political and economic climate can elevate stress to the point where a person could turn to drugs as a coping mechanism, said Helen Gyimesi, a drug and alcohol prevention specialist for Memorial Hospital. "These are mood-altering drugs," she said. "Working in a place like that after 9/11 could be scary". (See May 14, 2003, for a related incident).

The NRC will increase its inspections after four unplanned shutdowns of the nuclear plant's unit 2 reactor.

By SEAN ADKINS, Daily Record staff, Saturday,

November 15, 2003

For the next year, the U.S. Nuclear Regulatory Commission will increase the frequency of its inspections at Peach Bottom Atomic Power Station's unit 2. Since October 2002, unit 2 has experienced four unplanned reactor shutdowns, said Neil Sheehan, commission spokesman. An NRC rule permits a utility to have three unscheduled reactor shutdowns within 7,000 critical hours of operation or about one year, he said.

If a reactor has more than three unplanned shutdowns, the NRC bumps its level of oversight of the reactor.

Dave Simon, spokesman for Exelon Nuclear, said the issue of the shutdowns will be addressed at a public meeting slated for next week. Exelon Nuclear declared an unusual event Sept. 15 when electrical breakers on the PJM Interconnection power grid failed to isolate a lightning strike in Chester County. The strike generated a power surge on two electrical lines that feed into the plant, forcing the unit 2 and unit 3 reactors into automatic shutdown.

Exelon co-owns and operates Peach Bottom Atomic Power Station and Three Mile Island unit 1 in Dauphin County.

Continued on the following page...On July 22, a fault in the main generator system caused an automatic shutdown of Peach Bottom's unit 2. The unit's computerized reactor protection system received an over-current signal from the generator, which caused a trip of the main turbine and shut down the unit.

On April 12, the power station's unit 2 reactor shut down after an air line

failure. The malfunction resulted in the closure of a main steam line isolation valve, which tripped the automatic shutdown.
An equipment failure that caused multiple bypass valves to open Dec. 21 of last year also led to an unplanned shutdown of Peach Bottom Atomic Power Station's unit 2 reactor.
In response to those four unscheduled reactor shutdowns, the NRC has labeled unit 2 with a white performance indicator. A green indicator is awarded to reactors that require the basic level of inspection. The next level up, a white performance indicator, is assigned to a reactor that requires extra monitoring.
As part of the additional inspections, NRC officials will examine the unit 2 reactor for equipment reliability and operator performance, Sheehan said.
"These shutdowns pose no danger to the public," he said. Mixed findings at plant

Team investigated Sept. shutdown of 2 reactors
By KRISTIN FINAN Dispatch/Sunday News
A special team that analyzed the causes of, and responses to, an automatic shutdown of both reactors in September at the Peach Bottom Atomic Power Station reported mixed findings about the facility's handling of the event.
The U.S. Nuclear Regulatory Commission and representatives from Exelon, the company that operates the plant, presented their early report last night to the public at the Peach Bottom Inn in Delta.
Lightning struck the plant on Sept. 15 and disturbed the local electrical grid. Because Peach Bottom receives energy from the grid as well as provides it, it shut down automatically around 1:30 a.m. when those power sources were reduced.
The six-person team of specialists from the NRC regional office will release a full report by Dec. 18. As it outlined its findings last night, the team said it found both positives and negatives in the way the situation was handled.
Ma l f u n c t i o n : The Peach Bottom facility, which has been generating electricity since 1974, is on the west bank of the Susquehanna River in southeastern York County and serves about 2.5 million homes. It is one of 17 generation units operated by Exelon Nuclear.
A major problem with the September shut- down was a malfunction with a system backup, said NRC spokesman Neil Sheehan. Typically, if there is a problem with a reactor, emergency diesel generators provide more power.
But the reactors shut off after an hour, and one of the diesel generators shut down.
Team members said that while the generator's failure appears to be an equipment problem, they were not yet sure who should have been accountable.
Team members also found degraded conditions within the plant that should have been updated and said concerns voiced by staff members were never investigated.
Continued on the following page...They noted lapses in the monitoring of equipment, procedural problems
concerning what action should be taken after a shutdown and conflicts over which departments should take action about specific issues.

"We have not been as diligent at identifying problems and getting them out on the table as we need to," said Rusty West, Peach Bottom site vice president. "We need to better understand all the equipment anomalies that we have and pursue them with great vigor."
But the team noted that the Peach Bottom staff acted quickly and correctly determined how to respond to the incident, the team reported.
And managers have been diligent about conducting internal investigations and taking proactive actions --- such as cleaning equipment and defining emergency procedures, it said.
But some audience members said the NRC should be doing more to prevent future shutdowns.
Sept. 15 was the plant's fourth automatic reactor shutdown in the past year. On July 22, Peach Bottom's unit 2 reactor lost power after generator problems. The same unit shut down previously on April 12 and Dec. 21.
In response, the NRC recently labeled unit 2 a white performance indicator, meaning it will be monitored more closely, Sheehan said.
But Eric Epstein, chairman of Three Mile Island Alert, a group that monitors local nuclear plants, said the four shutdowns in a year are cause for concern.
"You should be concerned with the trend," Epstein said. "Any time there's a forced shutdown, it means the plant's safety systems are being challenged."

THE NRC’s Inspection team found six “Green: violations as a result of the incidents. All six were deemed Non-Cited violations

This was the forty-third, forty-fourth, forty-fifth, forty- sixth, forty-seventh and forty-eighth Non-Cited Violation since June 1998. Exelon's total cost avoidance, i.e., “credit” for 48 Non-Cited Violations = $2,490,000.

November 25, 2003 - NON EMERGENCY 10 CFR Section: 26.73 - FITNESS FOR DUTY Person (Organization): ANTHONY DIMITRIADIS (R1) Unit SCRAM Code RX CRIT Initial PWR Initial RX Mode Current PWR Current RX Mode 2 N Y 100 Power Operation 100 Power Operation 3 N Y 100 Power Operation 100 Power Operation Event Text FITNESS FOR DUTY NOTIFICATION DURING RANDOM DRUG TESTING
A contract employee tested positive during a random test. The employee's access to the protected area has been terminated. Contact the HOO for additional details. The licensee has informed the NRC Resident Inspector.

December 17, 2003 --PEACH BOTTOM-2 WAS AT 58% POWER AND RAMPING UP THIS MORNING FOLLOWING a reduction yesterday to 44% power in order to perform a planned control rod pattern adjustment, Exelon spokesman Ralph DeSantis said (NUCLEAR NEWS FLASHES)

Peach Bottom has rough week BY REBECCA J. RITZEL
Intelligencer Journal Staff

December 22, 2003 - Peach Bottom Atomic Power Station got a double dose of bad news last week. On Wednesday a routine test went awry, and on Thursday a report arrived in the mail warning that plant operator Exelon Nuclear likely will be cited for five safety violations for reactor shutdowns in September. On Tuesday, plant operators ran a test on the Unit 3 reactor's high-pressure coolant injection system. A turbine exhaust valve stuck open longer than expected, prompting workers to cancel the test midway through, according to a Nuclear Regulatory Commission report. Exelon Nuclear reported to the NRC that the plant was in "accident mitigation" status. The NRC classified the valve problem as a "non-emergency event."

Also last week, the NRC released a report concerning the September reactor shutdowns at Peach Bottom. Both active reactors went off-line Sept. 22 after a lightning strike in Chester County caused widespread power outages. An NRC inspection team visited the York County plant after the incident. In a 38-page letter to Exelon CEO Jack Skolds, an NRC deputy director of reactor safety detailed results of the inspection. The inspectors determined Exelon workers responded "adequately" to the emergency. "Nevertheless," the report says, "the operators were challenged by equipment and procedural problems."Exelon likely will be cited for five safety violations as a result those problems. Of chief concern to the NRC is Exelon's failure to properly maintain its emergency generators according to their instruction manuals. One of the four generators failed during the power outage. On the NRC's color-coded scale, safety violations are classified, in descending order of risk, as "red," "yellow," "white" and "green." Exelon likely will receive a white violation at Unit 2 and a green violation at Unit 3 for the generator problems. Exelon has until Sunday to provide additional evidence before the NRC considers penalizing the utility. The NRC already has decided to cite Exelon for three green violations for other equipment malfunctions that occurred Sept. 22.

The report also includes results from Exelon's own investigation into the lightning strike. PECO, an Exelon subsidiary, owns the power lines where the lightning strike occurred. A joint PECO/Exelon investigation found failures in circuit maintenance and a variety of problems in work practices.

When the lightning strike occurred, a circuit breaker failed to isolate the damaged power line, cutting off power to more than 100,000 PECO customers and shutting down three Exelon and PECO plants - Peach Bottom, Conowingo (Md.) Hydroelectric Station and Muddy Run Pumped Storage Facility in Holtwood.

Exelon determined Peach Bottom could have been isolated from the strike if power substations in Nottingham and Newlinville had been properly upgraded and maintained. Company officials said those upgrades will be included in widespread electrical infrastructure improvements. Report: Funds set aside for nuke cleanup inadequate

By AD CRABLE, Lancaster New Era

Dec 3, 2003, 13:53 EST, Congressional investigators say utilities are not adequately setting aside the hundreds of millions of dollars needed to clean up nuclear reactors at Three Mile Island and Peach Bottom when the plant sites close.

The report by the U.S. General Accounting Office claims that funds that, by law, must be set aside for restoring plant sites to their original condition may be as much as 25 percent lower than needed for TMI's Unit 2 reactor.

Decommissioning for Peach Bottom's closed Unit 1 reactor appears to be 51 to 100 percent underfunded, according to the report.

The cost of closing down and removing TMI Unit 2 was estimated at $433 million in 1997. The cost of decommissioning Peach Bottom Unit 1 was recently estimated at $129 million by plant owner Exelon Nuclear. The report did not say how much actually had been set aside to date in the decommissioning funds for the two reactors.

However, the owners of the two plants, where other reactors remain in use, said today that the decommissioning funding report by the investigative arm of Congress is flawed and that the money will be there when the plant sites end their useful life several decades from now.

Updating a 1999 report that first warned that decommissioning funding at many U.S. nuclear plants was not adequate, the GAO said on Monday that the $27 billion saved by the nuclear industry through 2000 was actually ahead of schedule.

But breaking down the savings by individual plant owners, the study said that owners of 42 of the 125 nuclear plants that have operated in the United States had accumulated fewer funds than needed to be on track to pay for eventual decommissioning, after the plants close.

Continued on the following page..."Under our most likely assumptions, these owners will have to increase the rates at which they accumulate funds to meet their future decommissioning obligations," the 55-page report said. Furthermore, the report criticized the federal Nuclear Regulatory Commission -- the nuclear industry's governmental watchdog -- for not taking action to force utilities to step up funding to address inadequacies.

In 1988, the NRC began requiring owners to certify that sufficient money would be available when needed to decommission their nuclear plants.

Beginning in 1998, utilities were required every two years to show how much money had been set aside and where the money was coming from. Most funds come from ratepayers and investments in trust funds.

The GAO study singled out Exelon Nuclear, the owner of Peach Bottom and the active reactor at TMI, as being behind the curve on set-aside funding. GAO said the trust funds for 11 of the 20 nuclear power plants owned by the company were inadequate.

However, the GAO found that Exelon Nuclear was actually well above other utilities in saving for the future closure of TMI's active Unit 1 reactor and Peach Bottom's two active reactors. And Exelon spokesman Craig Nesbit said the more-than-adequate funding will take care of any deficiency for the other Peach Bottom reactor that closed in 1974. Nesbit criticized the GAO report, saying it looked only at individual units instead of entire plant sites, and did not consider specific decommissioning strategies, such as Exelon's.

He also said the GAO study was "skewed" because it did not take into account that most nuclear plants, such as Peach Bottom and TMI, will be relicensed for another 20 years, which gives utilities more time to save decommissioning funds. "All of Exelon's plants are adequately funded for decommissioning now, and will be in the future," Nesbit said.

Continued on the following page...Though Exelon owns the site, the responsibility for decommissioning the

TMI Unit 2 reactor, closed since a 1979 accident, lies with FirstEnergy Corp., which bought out former TMI owner GPU.

The GAO study indicated the funding shortage is between 1 percent and 25 percent for TMI's Unit 2. FirstEnergy spokesman Scott Shields denied today that there were inadequate funds for restoring the Unit 2 site to its original condition. "We will continue to collect funds for the decommissioning for Unit 2 and we will be fully funded by the time the plant is retired," he said. Shields noted the site can't be cleaned up until Unit 1 is closed. TMI's license expires in 2014 but an extension is expected. Eric Epstein, an expert witness on decommissioning before the Pennsylvania Public Utility Commission and chairman of TMI-Alert, a safeenergy citizens group, is not so confident. He said the GAO study on decommissioning shortcomings is just the tip of the iceberg. Citing the escalating costs of disposing of low-level and high-level nuclear waste, Epstein said "clearly the utilities underestimate and lowball decommissioning costs." Epstein fears utilities will not be making the profits in the future when plants are closed down and will not be able to pay for what it will actually cost to restore nuclear plant sites. People not yet born may have to pay for that shortcoming through higher electric bills, he said. Inadequate funding for future closures was a constant concern expressed by former Lancaster mayor Art Morris when he chaired a citizens advisory panel on the cleanup of TMI in the 1980s."It's just the same old story. It's absolutely remarkable that after all these years of public comment and criticism that the Nuclear Regulatory Commission just sits and does nothing about (inadequate funding)," Morris said today. "The taxpayers will have to pay for it. There needs to be an NRC that stays on top of this and monitors it."

December 22, 2003 - NATIONAL GUARD TROOPS BEGAN PROTECTING PENNSYLVANIA'S NUCLEAR POWER PLANTS at 7 a.m. local time today, according to Gov. Edward Rendell (D). Troops will remain at the plants as long as the threat level remains at "orange," indicating a high risk of a terrorist incident, Rendell said. Deployment of the state National Guard to the nuclear plants was among the steps the state government took to protect Pennsylvania infrastructure in response to the raising of the Homeland Security Threat Level yesterday. The nuclear plants in Pennsylvania are Beaver Valley, Limerick, Peach Bottom, Susquehanna and Three Mile Island. NRC spokesman Dave McIntyre said he was not aware of other states deploying National Guard troops to nuclear plants in response to the increased threat level (NUCLEAR NEWS FLASHES.)

- NEW YORK,

Jan 13, 2004 (Reuters) - Exelon Corp., the No. 1 U.S. nuclear plant operator, on Tuesday said it named nuclear industry veteran Christopher Crane as chief nuclear officer and president of its key nuclear division. Crane replaces John Skolds, who was named president of Exelon's energy delivery unit -- a position left vacant by the resignation of Michael Bemis.
Jan 13, 2004 Reuters
Power News Sat,

Jan. 17, 2004 Authorities: Pilot who buzzed area was drunk
By NICOLE WEISENSEEEGAN
The pilot who terrorized the airways with his erratic flying for four hours Thursday night - circling the Limerick nuclear plant and buzzing Philadelphia International Airport - was drunk, authorities said yesterday.
When he emerged from his single-engine plane, he was staggering, his eyeswere bloodshot, and his pants were unbuttoned and unzipped, authorities said.
Tests showed that the pilot, John Salamone, owner of a Pottstown concrete company, had a blood-alcohol level of 0.13, over the legal limit of .08. Until tests are complete, however, he has not been charged with DUI, according to Montgomery County District Attorney Bruce L. Castor Jr. Salamone, 44, owner of J. Vincent Concrete Contractors , was released into the custody of his brotherin-law. The single-engine plane he was flying is registered to his firm, records show.
Jim Peters, a Federal Aviation Administration spokesman, said his agency had opened an investigation into Salamone but have not yanked his license. "At the end we will make a recommendation about what to do," he said. That could mean anything from no action to a civil penalty, or suspension or revocation of his license. Salamone did not return phone calls requesting comment.
Salamone took off from Pottstown-Limerick Airport between 6:15 and 6:30 p.m., Peters said. He first flew over Center City, then headed toward Philadelphia International Airport, prompting controllers to order six aircraft that were on final descent to clear out of the way, Peters said.
Salamone then headed to South Jersey and attempted tried to land at an airport outside Glassboro before returning to Philadelphia airspace.
He declined to land in Philadelphia, and then headed to Limerick, where he landed briefly there, before taking off toward the nuclear plant. He finally landed again at Limerick airport and was arrested, authorities said. York Daily Record: NRC watching Peach Bottom -

The power station was issued violations after a September reactor shutdown.

By SEAN ADKINS Daily Record staff Tuesday,
February 10, 2004 - The U.S. Nuclear Regulatory Commission will be more vigilant of Peach Bottom Atomic Power Station’s Unit 2 reactor as result of a

second-tier safety violation. The commission has penalized the Unit 2 reactor with a “white” finding related to the failure of an emergency diesel generator during an unscheduled Sept. 15 reactor shutdown. A white violation refers to an event at the plant that is considered as of low to moderate safety significance.
Since the generator failure affected both of the plant’s units, NRC officials tacked on a green violation in regard to the power station’s Unit 3 reactor.
A green violation is an event characterized as being of very low safety significance, said Neil Sheehan, spokesman for the NRC. The commission decided on a green violation because fewer safety-related electrical loads powered by the emergency generator exist for Unit 3.“This will help us better know where we need to focus an increased level of attention going forward,” Sheehan said.
A bolt of lighting struck a Chester County power pole Sept. 15, generating an electrical surge along power lines that feed into Peach Bottom Atomic Power Station.
The strike led to the automatic shutdown of the plant, which triggered the formation of a special, augmented NRC inspection team.
As part of its findings, the team found that faulty protection circuitry and a loose wire failed to contain the surge that disabled the plant.
Exelon has replaced all damaged fuses and tightened necessary wires to help ensure a similar event will not shut down the power station.
Within moments of the September shutdown, the plant’s four diesel generators kicked on to power the station’s vital equipment and offices.About an hour later, one of the reserve generators seized. Exelon declared a “discretionary” unusual event — the lowest of the NRC’s emergency categories.
“This is not a common thing,” Sheehan said. “These generators should operate smoothly.”
The commission’s inspection team found that deficient procedures were followed during the 1992 installation of generator adapter gaskets. Combustion gas leaked into the jacket water cooling system — a problem that led to the automatic tripping of the generator Sept. 15.
In March and April 2003, Exelon took corrective actions to repair the observed low jacket water pressure conditions. The NRC said the...problem was not resolved.
Last June, commission inspectors documented that lube oil had leaked from loose flange joint bolts on an emergency diesel generator at the plant. That leak caused a small fire in the exhaust manifold during a test.
The NRC responded to the fire by issuing a green violation.
Exelon has less than a month to reply to the commission’s white finding.
The company will not appeal the determination, said Craig Nesbit, a company spokesman.
Exelon agrees with the NRC’s findings, he said.

February 22, 2004 Event Text
MANUAL SCRAM AT PEACH BOTTOM 2 DUE TO
DECREASING CONDENSER VACUUM
"Peach Bottom Unit 2 reactor was manually scrammed due to degrading

main condenser vacuum. The reactor was manually scrammed prior to reaching the automatic scram setpoint. All plant systems responded as expected with no significant issues noted. A Group II and Group III Primary Containment Isolation was received due to reactor water level passing through 1 inch. All isolation systems responded as required and repositioned to their expected positions." The licensee also indicated that all control rods properly inserted into the core. The method of decay heat removal was using the main condenser. The licensee initiated a post scram review to identify and correct the source of degrading vacuum. The licensee also indicated the manual scram was initiated at 25 inches and lowering of condenser vacuum.

The licensee notified the NRC Resident Inspector.YDR: Reactor shutdown no threat - Mechanical problems caused Peach Bottom's
Unit 2 reactor to be shut down Sunday.

By SEAN ADKINS Daily Record staff Wednesday,

February 25, 2004 -Operators manually shut down Peach Bottom Atomic Power Station's Unit 2 reactor Sunday after a series of mechanical problems.

Last week, control room workers monitored an air leak in the reactor's condenser — equipment used to turn steam into water. The condenser pumps that water back to the reactor.

On Tuesday, plant officials determined the leak came from an expansion joint caused by routine wear and tear of the system, said Dana Melia, spokeswoman for Exelon Generation. Exelon co-owns and operates the power stat ion."That type of wear and tear is typical of any steam plant," Melia said.

That leak caused a loss of vacuum — a piece of equipment found inside the condenser, she said.

The shutdown caused no threat to public health or the plant's ability to distribute electricity, Melia said.

Peach Bottom Atomic Power Station's Unit 3 was not affected by its neighbor's shutdown and continues to function at full power.

The second unit's reactor is designed to go into automatic shutdown if the vacuum level drops to a specific set point, Melia said.

On Sunday, operators elected to manually take the reactor offline and bring the unit to a cold shutdown."(A shutdown) is safer when it's manual rather than automatic," Melia said. "You have more control over it."All equipment used to carry out the shutdown functioned as it should, Melia said."They did what they were supposed to do," said Diane Screnci, spokeswoman for the U.S. Nuclear Regulatory Commission. "The plant's systems responded as expected."Soon after the 3:11 p.m. shutdown, the plant notified its resident NRC inspector of the unit's problems.

The commission is having its inspector look into the cause of the shutdown, Screnci said. As for Exelon, officials are investigating the cause of the leakage and what steps are necessary to bring the plant's second reactor back online, Melia said. "We are trying to determine why it happened," she said.

Plant officials will use the shutdown as an opportunity to conduct routine

maintenance of the site such as the checking of valves.
While Melia did not say when the reactor would return to service, Screnci said the time frame is more “a matter of days rather than months.”
YDR: NRC still watching Peach Bottom -
Four unplanned shutdowns in about a year netted the reactor a ‘white' violation, which gets it extra oversight.

By SEAN ADKINS Daily Record staff Saturday,
April 10, 2004 -At bottom: · IF YOU GO A low to moderate safety violation discovered last year means additional regulatory oversight for Peach Bottom Atomic Power Station's Unit 2.

The unit will face a Nuclear Regulatory Commission supplemental inspection later this year as a result of deficient performance based on its number of unplanned shutdowns.

The commission will follow a normal inspection schedule for the power station's third unit through Sept. 30, 2005. Based on the assessment of an NRC inspection team, the commission cited Unit 2 with a "white" violation for the failure of the emergency diesel generator.

Following a Sept. 15 unplanned shutdown of Units 2 and 3, a reserve generator seized.

The generator, one of four, helps power the plant's vital equipment and Offices. A commission inspection team later found that deficient procedures were followed during the 1992 installation of generator adapter gaskets. Gas leaked into the equipment's jacket water cooling system — a problem that led to the automatic tripping of the generator Sept. 15. The NRC team determined that corrective actions Exelon took to repair the observed low jacket water pressure conditions in March and April 2003 were inadequate. The problem was not r e sol v ed.

Since that time, the plant has created corrective actions to ensure the operation of the generators, said Pete Resler, spokesman for Exelon Nuclear, which co-owns and operates the power station.

For example, the plant has revised maintenance, testing and inspection procedures for the diesel generators.

Training materials regarding the generators have been updated, Resler said.

Aside from the low to moderate safety breach, five "green" violations at Unit 2 in 2003 caught the attention of the commission.

A green violation is characterized as being of very low safety significance.

Some of the green infractions include problems with the second unit's safe shutdown emergency lights and the emergency diesel generator fire protection system. "These findings highlight a need for Exelon to improve this area," according to a March 3 letter sent by the NRC to the utility.

Commission officials will make another trip to Peach Bottom Atomic Power Station's Unit 2 in September to review the causes behind the reactor's four unplanned shutdowns per 7,000 critical hours, or roughly one year of operation.

The shutdowns occurred between the fourth quarter of 2002 and the fourth quarter of 2003, said Diane Screnci, spokeswoman with the NRC.

The fourth shutdown that occurred during the third quarter of 2003 netted the second reactor a white performance indicator, she said.

Increased oversight was maintained by the NRC at Peach Bottom-2, "which will face a Nuclear Regulatory Commission supplemental inspection later this year as a result of deficient performance based on its number of unplanned shutdowns. The commission will follow a normal inspection schedule for the power station's third unit through Sept. 30, 2005 (York Daily Record.) Unplanned shutdowns and equipment failure were to blame.

By SEAN ADKINS Daily Record staff Thursday,

April 15, 2004 -With little more than a projection screen between them, officials with both the Nuclear Regulatory Commission and Exelon Generation met Wednesday night at the Peach Bottom Inn to walk through the agency's annual safety performance assessment of Peach Bottom Atomic Power Station.

Based on a 2003 low-to-moderate safety violation, commission officials will host a supplemental inspection of Unit 2 to ensure the reliability of the plant's diesel generators.

In September, NRC staff will investigate through an additional inspection the reason behind Unit 2's four unplanned shutdowns per 7,000 critical hours, or roughly one year of operation. The unscheduled shutdowns occurred between the fourth quarter of 2002 and the fourth quarter of 2003.

The fourth shutdown that occurred during the third quarter of 2003 netted the second reactor a white performance indicator — a violation of low to safety s igni f i canc e .

Between Jan. 1 and Dec. 31, 2003, both Peach Bottom Atomic Power Station's Unit 2 and 3 reactors racked up 17 green violations — an infraction of very low safety significance, said Brian Holian, deputy director of reactor projects for the NRC's Region 1.

Some of the green infractions include problems with the second unit's safe shutdown emergency lights and the emergency diesel generator fire protection system. "Seventeen green violations," Holian said, "it's a hefty amount. But you have to remember it's a twin reactor plant and that's for both units."

Bill Levis, vice president of mid-Atlantic operations for Exelon, said the company views the violations as an indicator that the plant did not meet expectations. "We can clearly do better than that," he said.

The commission will follow a normal inspection schedule for the power station's third unit through Sept. 30, 2005.

On Sept. 15, one of the plant's four emergency diesel generators seized. The equipment's failure occurred in the hours following an unplanned shutdown of both reactors.

A commission inspection team later found that deficient procedures were followed during the 1992 installation of generator adapter gaskets. Gas leaked into the equipment's jacket water cooling system — a problem that led to the automatic tripping of the generator.

Typically, the plant runs all four diesel generators for at least two hours every two weeks to check for reliability, said Craig W. Smith, senior resident NRC inspector at Peach Bottom Atomic Power Station. The NRC team determined that corrective actions Exelon took to repair the observed low jacket water pressure conditions in March and April 2003 were inadequate. The problem was not resolved.

"We didn't do enough fast enough," Levis said. "We recognize our obligation to public health and safety. We take that very seriously." Since the generator failure, the plant has instituted a monitoring system that tracks the amount of gas that could leak into the generator's cooling system, said Paul Davison, director of engineering for the power station.

Following the failure, the plant checked all the generator adapter gaskets and installed new equipment as needed, he said.

Other tests that were in place prior to the generator shutdown scan for temperature, engine reliability and vibration control.

"We will follow all this up with inspections," Holian said. "The proof will be in the pudding."

July 2, 2004:
GOVERNOR RENDELL ANNOUNCES ENHANCED SECURITY MEASURES AT NUCLEAR POWER PLANTS
National Guard, State Police to Provide a 24-hour Presence and Random, Unannounced Patrols During Independence Day Holiday

HARRI SBURG: Governor Edward G. Rendell today said the Pennsylvania National Guard and the Pennsylvania State Police will provide both a 24-hour presence and random, unannounced security patrols at the Commonwealth's five nuclear power plants. The enhanced security measures will be provided in a coordinated fashion with the plant operators and their security teams, and will remain in force at least through the conclusion of the Independence Day holiday.
"My Homeland Security Team continues to coordinate on a regular basis with the U.S. Department of Homeland Security, the Federal Bureau of Investigation, the U.S. Department of Defense, and the Nuclear Regulatory Commission in order to discuss and share relevant intelligence information and threat analysis," Governor Rendell said."Although there currently exists no credible threat against any
Pennsylvania nuclear power facility, in an abundance of caution I have asked the National Guard and State Police to immediately commence enhanced security measures at our nuclear power stations. At a minimum, we will maintain this deployment status through the holiday weekend."
The state's nuclear power plants are Beaver Valley in Shippingport Borough, Beaver County; Susquehanna in Salem Township, Luzerne County; Limerick in Limerick Township, Montgomery County; Peach Bottom in Delta Borough, York County; and Three Mile Island in Londonderry Township, Dauphin County.

Groups want action on nuke fuel storage

Watchdogs prod federal regulators to shore up spent-fuel pools against possible terrorism. Peach Bottom is among plants affected.

August 11, 2004

Day: Wednesday Page: B-1 Byline: Ad Crable

LANCASTER NEW ERA - Used, deadly uranium fuel stored at the Peach Bottom and 31 other similarly designed nuclear reactors around the United States is especially vulnerable to terrorist attack, watchdog groups charge. "Nuclear reactors are pre-deployed weapons of mass destruction," said Deb Katz, executive director of Citizens Awareness Network, one of three-dozen public interest groups signing the petition, including Greenpeace, Union of Concerned Scientists and the locally based Three Mile Island Alert.

The groups filed a petition for action with the U.S. Nuclear Regulatory Commission, calling on the agency to immediately address structural vulnerabilities to terrorism at the plants. "It is the NRC's job to protect our health and safety and assure public confidence in the regulatory process. Presently, NRC's efforts are inadequate," said Eric Epstein of TMI Alert and a candidate for the state Senate. While alleging that all 103 commercial nuclear plants in the country are vulnerable to accidents or "acts of malice or insanity," the 33-page petition particularly points the finger at spent-fuel pools at Mark I and II boiling water reactors, such as that found at Peach Bottom.

At those nuclear plants, used uranium fuel rods are placed in pools of water high above the ground, covered by only a lightweight roof and walls, the groups say. The arrangement, they say, makes the pool vulnerable to terrorist attacks from planes or on the ground. "If a pool is breached, there is no surrounding structure or backfill to inhibit the drainage of water. Its cooling system is vulnerable to attack at several points. The exterior configuration of the reactor building facilitates accurate aiming - for example, of an explosiveladen aircraft - by a knowledgeable attacker," the petition states.

The group says breaching of spent-fuel pools "could cause great public harm" with widespread radiation fallout.

The groups outline a number of steps they feel the NRC should take, including beefing up on-site security; re-equipping spent-fuel pools with lowdensity racks so that spent fuel would not ignite if water were lost from the pool; establishing ways to recover from loss of water; and improving emergency response plans for surrounding communities.

The petition comes shortly after concerns about spent-fuel vulnerability were voiced by some members of Congress.

Craig Nesbit, spokesman for Peach Bottom operator Exelon Energy, said this morning that "there is nothing substandard about any of Exelon's plant designs."

The NRC has no comment on the petition while the agency is processing it to see if it meets the NRC standards for action, spokeswoman Diane Screnci said.

A spokeswoman for the Nuclear Energy Institute, a nuclear industry

group, said she had not yet seen the petition.
In another development affecting Peach Bottom, the federal Department of Energy announced it would pay Exelon at least $300 million for costs associated with storage of spent fuel at its nuclear plants.

The DOE had promised in the early 1980s to accept used fuel from U.S. reactors for disposal, beginning in 1998. Amid extensive controversy, however, a national repository has not yet been built. Exelon and 64 other companies sued DOE for not taking the fuel. By SEAN ADKINS Daily Record/Sunday News,

September 1, 2004 -The Nuclear Regulatory Commission has requested that officials at Peach Bottom Atomic Power Station Unit 2 submit in writing plans to address inadequate corrective actions for known equipment problems.

The cross-cutting issue includes two "green" violations of very low safety significance listed within the commission's mid-cycle performance review and inspection plan of the power station.

That review stretched from July 1, 2003, to June 30. The NRC released the review Monday.

Next month, a team from the NRC will travel to the plant to run an additional inspection on Unit 2 to determine how Exelon has responded to "white" performance indicators found in the third quarter of 2003 and the first quarter of 2004.

Exelon co-owns and operates Peach Bottom Atomic Power Station.

The power station's Unit 3 performance requires no additional NRC oversight. That unit will follow a normal inspection schedule through March 31, 2006.

The supplemental inspection will investigate the reason behind Unit 2's four unplanned shutdowns per 7,000 critical hours, or roughly a year of operation.

The unscheduled shutdowns occurred between the fourth quarter of 2002 and the fourth quarter of 2003. One of the unplanned shutdowns included the failure of one of the plant's four emergency diesel generators. Following the shutdown, a commission inspection team found that deficient procedures were run during the 1992 installation of generator adapter gaskets. Gas leaked into the equipment's jacket water cooling system — a problem that led to the automatic tripping of the generator.

The NRC determined that the problem warranted a "white" finding, or a violation of low to moderate safety significance.

Earlier this year, the plant formed a root-cause analysis team from the power station's maintenance and engineering divisions to deal with the failed diesel generator, said Dana Melia, an Exelon spokeswoman. The plant put its self-critical analysis into action in June and further modified its plan last month, she said. The actions focused on the maintenance of the generator and other reliability conditions, Melia said. The NRC will look at all the plant's actions during its September inspection.

Power station officials are now forming a second root-cause team to deal with the plant's ongoing problems with cross-cutting issues, Melia said.

Cross-cutting issues are events that affects many different areas of plant performance, said Neil Sheehan of the NRC. "The substantive cross-cutting issue was based on several inspection findings in which corrective action for a known equipment problem was either insufficient or delayed for implementation," according to the mid-cycle review.

The most recent findings deal with problems related to Unit 2's high-pressure coolant injection oil system and high-pressure service water valves, Sheehan said. Both problems resulted in green violations.

The high-pressure coolant injection oil system is a reserve safety operation put into play to shut down the plant quickly, Sheehan said.

The oil is used to lubricate the system that injects coolant into the reactor vessel to keep the fuel cool at times of emergency, he said.

In June, plant officials found that oil flow to a part of the system had been interrupted. As a result, damage to the turbine bearing and rotor rendered the machine inoperable. The plant had to replace the bearing and rotor. The system was unavailable.

The second green violation dealt with corrective actions of high-pressure service water valves that pull water from the Susquehanna River that is used to cool down various plant components, Sheehan said.

How the plant will respond to the violations will be part of the letter sent to the NRC in October, Melia said.

September 12, 2004- State plan to handle nuke crisis challenged
Preschools, hospitals and nursing homes are unprepared, 2 residents say
BY GARRY LENTON Of The Patriot-News
State and federal authorities are investigating allegations that Pennsylvania is unprepared to evacuate preschool children and nursing home and hospital patients during a nuclear accident.
The federal government requires that the state have a plan for moving people who cannot care for themselves and live within 10 miles of a nuclear plant. Two Harrisburg area residents allege that the state has been out of compliance with federal safety requirements for nearly two decades.
Gov. Ed Rendell's office and the Federal Emergency Management Agency took on the review of the state's plan after receiving a letter last week from Larry Christian and Eric Epstein, chairman of the watchdog group Three Mile Island Alert, detailing these issues. The Nuclear Regulatory Commission also received the letter.
If the accusations are deemed true, it would call into question the validity of the operating licenses for the five nuclear power stations in Pennsylvania. Federal law requires the NRC to determine that the public will be protected in a radiological emergency before it grants a license to open a nuclear plant.

December 22, 2004 Event Text
REACTOR SCRAM AND ECCS INJECTION FOLLOWING OPENING OF TURBINE BYPASS VALVES
"At approximately 04:55 on December 22, 2004, Unit 2 experienced a malfunction of Electro-Hydraulic Control (EHC) system resulting in opening of

main turbine bypass valves and resultant loss of reactor pressure. The reactor automatically shutdown on RPS with the completion of a Group I isolation signal (Reactor pressure 850 prig and Reactor mode switch in RUN) resulting in a closure of the Main Steam Isolation Valves (MSIVs). Reactor level lowered to (ECCS) initiation set-point of -48 inches and High Pressure Coolant Injection (HPCI) system and Reactor Core Isolation Coolant (RCIC) system automatically initiated and restored level. When reactor level lowered below the 1 inch setpoint, Group II and III Primary Containment Isolation System (PCIS) signals initiated. All Unit parameters are stable and RPS/PCIS/ECCS systems performed as designed. MSIVs remain closed. Reactor level and pressure are stable with HPCI and RCIC systems in control. Group I, II, and III isolations have been reset. The EHC malfunction is presently under investigation by Station Management." All systems functioned as required. The reactor water level is now at 23 inches and stable and the licensee is conducting a slow depressurization to Mode 4 to investigate the EHC system malfunction. The level transients experience during the scram would be expected with the closure of the MSIVs.
The licensee has notified the NRC Resident Inspector.

Peach Bottom-2, already under increased NRC supervision, scrams again

REACTOR SCRAM AND ECCS INJECTION FOLLOWING OPENING OF TURBINE BYPASS VALVES

"At approximately 04:55 on December 22, 2004, Unit 2 experienced a malfunction of Electro-Hydraulic Control (EHC) system resulting in opening of main turbine bypass valves and resultant loss of reactor pressure...All Unit parameters are stable and RPS/PCIS/ECCS systems performed as designed...The EHC malfunction is presently under investigation by Station Management... The reactor water level is now at 23 inches and stable and the licensee is conducting a slow depressurization to Mode 4 to investigate the EHC system malfunction...The licensee has notified the NRC Resident Inspector." (NRC, Region I,Power Reactor Event Number: 41277.)

Continued on the following page... By TOM JOYCE Daily Record/Sunday News Saturday,

December 25, 2004 -Peach Bottom Atomic Power Station's Unit 2 reactor had an emergency shutdown early Wednesday morning. It was down for about 48 hours, and started up again on Friday morning, according to Craig Nesbit, a spokesman for Exelon, the company that owns the plant.

No radiation leaked during the shutdown, Nesbit said. In fact, the shutdown didn't occur in a portion of the plant that contains radiological parts. According to Nesbit, the problem occurred when a circuit card malfunctioned in the electronic hydraulic control system.

The plant shut down, as it's designed to do in such circumstances. Nesbit characterized it as an engineering issue rather than a safety issue.

The time-consuming part was figuring out precisely where the malfunction occurred. "It's a relatively simple operation, but it takes a few days," Nesbit said.

The plant has experienced several emergency shutdowns in the past two years, Nesbit said. Plant officials are now conducting a "root cause investigation" to see if the problems are all the result of an underlying problem, or simply isolated occurrences. "A root cause investigation is a very detailed and intense look at the root cause of the problem," Nesbit said.

The Nuclear Regulatory Commission could not be reached for comment. On Friday, the Lancaster Intelligencer-Journal reported that an NRC spokesman said the commission is concerned about the frequency of Peach Bottom's shutdowns .

In August, the NRC sent Exelon's CEO a letter warning the company to improve its routine maintenance work for the remainder of 2004 or face increased federal oversight. And in September, the NRC sent a special inspection team to see what Exelon was doing to prevent emergency shutdowns at Unit 2.

Feb. 7, 2005- Peach Bottom Unit 2 shuts down for valve replacement

Chicago-based energy company Exelon Corp.'s 1,110-megawatt Unit 2 reactor at the Peach Bottom nuclear station in Pennsylvania exited a work outage and ramped up to full power by early Monday, the U.S. Nuclear Regulatory Commission said in its power reactor status report.
The company shut the unit on Feb. 2 to replace a safety relief valve.
The 2,220 MW Peach Bottom station is located in Peach Bottom, Pennsylvania, about 75 miles southwest of Philadelphia. There are two 1,110 MW units 2 and 3 at Peach Bottom. Unit 3, meanwhile, continued to operate at full power.
One megawatt powers about 1,000 homes, according to the North American average.
Exelon Nuclear, a unit of Exelon's Exelon Generation subsidiary, operates the station for its owners: Exelon (50 percent) and New Jersey-based energy company Public Service Enterprise Group Inc. (PSEG) (50 percent).
In December 2004, Exelon agreed to acquire PSEG. Pending regulatory and shareholder approvals, the companies expect to complete the deal in 2006.
-Report from Rueters

Feb. 9, 2005 -Peach Bottom Unit 2 back in production

Chicago-based energy company Exelon Corp.'s 1,110-megawatt Unit 2 at the Peach Bottom nuclear station in Pennsylvania ramped up to 94 percent of capacity by early Wednesday, the U.S. Nuclear Regulatory Commission said in its power reactor status report.

On Tuesday, the unit was operating at 64 percent of capacity as it increased power following a planned control rod pattern adjustment.

The company performed the rod pattern adjustment to optimize the efficiency of the fuel in the reactor after the reactor exited an outage started on Feb. 2 to replace a safety relief valve.
The 2,220 MW Peach Bottom station is located in Peach Bottom, Pennsylvania, about 75 miles southwest of Philadelphia. There are two 1,110 MW units 2 and 3 at Peach Bottom. Unit 3, meanwhile, continued to operate at full power.One megawatt powers about 1,000 homes, according to the North American average.
-Report from Rueters

Feb. 11, 2005- Nuclear plant guard rule could be year away
TMI watchdog group decries 'glacier' pace The Harrisburg-based nuclear watchdog group Three Mile Island Alert has been waiting since Sept. 12, 2001, for the U.S. Nuclear Regulatory Commission to decide whether nuclear plant owners must post armed guards at their front gates.
TMIA will have to wait another year for its answer, according to an NRC memo released to Wednesday. The memo outlines a schedule the NRC plans to follow as it considers rule changes for security at the nation's 63 nuclear power stations.
The memo, from Luis A. Reyes, executive director for operations, anticipates that recommendations that could mandate guards at plant entrances will be presented to the commissioners next February.
If the NRC adheres to the schedule, the recommendation would come nearly five years after TMIA petitioned the agency for the change.
A statement issued by the watchdog group yesterday called the NRC's failure to act on its request irresponsible and unreasonable. "For nearly four and a half years the NRC has misled [TMIA] about its deliberations on the petition," the statement said. "When requesting status updates, the NRC perpetually stated that a decision on the petition would be made within three to six months."
TMIA asked the NRC to require plant operators to keep at least one armed guard at each plant entrance. The petition, which was drafted weeks before the terror attacks of 9/11, argued that the guards would serve as a physical and visual deterrent against attacks.

Since 9/11, the NRC has issued security requirements aimed at making the plants less vulnerable to attack. Changes include the addition of guard towers, truck barriers, deeper background checks and high-tech fencing. Most, if not all, plant owners post guards at their front gates.
For months after the terror attacks, Pennsylvania was among several states to assigned National Guard troops to the plants. NRC officials have denied allegations of foot dragging. Petitions such as TMIA's, which require rule changes, take a long time to complete, officials said.
The Nuclear Energy Institute, which represents plant owners and operators, opposes the petition. It told the NRC that guards should be posted only when the level of security threat makes it prudent.
On July 29, 2005, the NRC a issued White Violation relating to another staffing deficiency at Three Mile Island where “approximately 50% of the emergency responders,” including “key responders” were “overdue” for their annual training for “an

approximate five month period. (Please refer to Thursday, July 14, 2005, for background material).
-Report by Garry Lenton of the Patriot-News

March 30, 2005- NRC reviews Peach Bottom, plant a leader in shutdowns

Attendees seemed more in the dark last night after a 90-minute session aimed at shedding light on Peach Bottom Atomic Power Station's performance last year.
Exelon and Nuclear Regulatory Commission officials didn't exactly wow the crowd of about 40 with a slide show highlighting corporate progress, touting a 25 percent reduction in radioactive exposure to employees and diagramming federal "matrixes" and "cornerstone" safety guidelines.
One attendee asked why the commission couldn't just grade performances A to F, drop bureaucraticese and spell out problems that affect the public.
The bottom line: The NRC found that Peach Bottom improved in 2004 with two shutdowns of its Unit 2 reactor compared to three in 2003.
The shut downs placed Peach Bottom in the **top three nationwide** for unexpected shutdowns right behind Indian Point 2 in New York and Saint Lucie Unit 2 in Florida.
Five shutdowns in Unit 2 over two years is a lot when compared to the national average of less than one shutdown annually at the country's 103 commercial plants, said Eric Epstein of Three Mile Island Alert, a Harrisburg-based nonprofit citizens' organization.
The NRC said the shutdowns, called "scrams," were low-level safety risks but noteworthy nonetheless.
Want better procedures: Federal officials also warned the plant, operated by Exelon Corp., that its procedure in finding and reporting causes for shutdowns needs improvement. "They said our focus regarding inspections was too narrow," said Robert Braun, Exelon's site vice president at Peach Bottom. "We'll apply what they told us, which was to broaden our investigation."
Braun said that the shutdowns pose no threat to the public but only affect the company's bottom line. He further touted adherence to safety guidelines saying the plant was taking a "proactive approach." That tack, he said, would help plant workers discover problems such as the cause of a Unit 2 shutdown in July 2003.
A piece of broken fan belt that had been lost "a number of years ago" entered a cooling system and caused the shutdown. The debris wasn't found when the belt broke, but "years later it came back to haunt the plant," Braun said. "We continue to improve our existing processes," he added.
Epstein questions numbers: Epstein asked corporate and federal officials how many workers were employed at Peach Bottom, whether they had decreased in the past five years and if so, would that affect plant performance and the reduction in radiation exposure. NRC Chief of Projects Branch 4 Mohamed Shanbaky said the plant was in federal compliance with the number of employees needed for high-profile jobs such as reactor operators.
Shanbaky further said the NRC doesn't focus on the overall number of employees but rather whether federal rules are obeyed and safety regulations adhered to.

"This meeting was the NRC's assessment for 2004," said April Schlipp, Exelon spokeswoman, who added that there have been no staffing changes since the 2003 assessment. "We've been able to improve for the past two years; that's really the most relevant here."
Beth Birchall, a Lancaster County resident, sat in the back of the Peach Bottom Inn banquet room shaking her head.
"They seemed prepared," she said. "But there wasn't a lot of information."
The NRC has scheduled quarterly, team and regional inspections of the plant in 2005.
-Report by Kathy Stevens of the York Dispatch

May 27, 2005 -Many emergency sirens would not work if power lines were down

In the event of a nuclear accident or an act of terrorism at a U.S. nuclear power station simultaneously occurring with an electrical grid failure, only 27 percent of the nation's 62 nuclear power emergency planning zones using public notification siren systems are prepared to fully operate their emergency sirens independent of the main power lines," emergency enforcement petition filed by Nuclear Information & Resource Service, Three Mile Island Alert and numerous citizens' groups.
While the Nuclear Regulatory Commission revealed that some but not all of the sites without backup power are preparing to create battery backups, the NRC actually denied the petition, and argued that the concerned citizens should instead use a petition for rulemaking process that can take as long as two years.
Peach Bottom is grid-dependent for sirens.

July 2005- Peach Bottom Investigation: NRC probes shutdown at Peach Bottom

Officials with the Nuclear Regulatory Commission will follow up on the cause of a turbine trip that led to the automatic shutdown of Peach Bottom Atomic Power Station's Unit 2 reactor on July 10, 2004.
At the time of the shutdown, the unit's reactor coolant system experienced a high pressure condition that caused both recirculation pumps to trip. As a result, three safety-relief valves lifted and reseated.
By Tuesday morning, the reactor had returned to 67 percent power.
In September 2004, the NRC staff, through an additional inspection, investigated the reasons behind Unit 2's four unplanned shutdowns per 7,000 critical hours, or roughly one year of operation. The unscheduled shutdowns occurred between the fourth quarter of 2002 and the fourth quarter of 2003.
On December 22, 2004, Peach Bottom Atomic Power Station's Unit 2 reactor had another emergency shutdown and was off-line for 48 hours.

Circuit Breaker Replacement Primary Bushings Not Tested
to American National Standards Institute (ANSI) Standards

While investigating the dedication process of a different circuit breaker component, GE Energy-Nuclear (GE) discovered that ANSI testing had not been

accomplished for the AM breaker primary bushings used in Magne-Blast circuit breakers. The replacement primary bushings were provided by GE Supply PSC, Sharon Hills, Pa., and supplied to Watts Bar and Peach Bottom, units 2 and 3, by GE as safety-related components. The NRC issued a report to inform all licensees of this issue since additional licensees may have obtained these devices through other dedicating entities.
Previously, the GE product department produced Magne-Blast circuit breakers and switchgear, that was qualified to the appropriate ANSI C37 standards. When the GE breaker plant operation facility was closed, GE contracted with a vendor to manufacture primary bushings. The contractor uses a similar but not identical insulating material, and has variations in the manufacturing process for the bushing construction. GE dedication specifications addressed the replacement insulation material, but not the variation in the manufacturing process. An implicit assumption in the GE dedication specification was that testing in compliance with the applicable ANSI standard had been completed.
GE has determined that design tests in accordance with certain ANSI C37 Industry Standards for Switchgear were not performed prior to implementation of bushing design changes for Parts Q0845D0123G001, and Q0845D0124G001 andG003, which have been delivered to Peach Bottom 2, 3 and Watts Bar 1 for use as replacement primary bushings in Magne-Blast circuit breakers.
For primary bushings purchased under the identified purchase orders and placed in inventory, GE recommended that the primary bushings in inventory not be installed until after successful completion of the ANSI standards testing.
For primary bushings purchased under the identified purchase orders and installed in Magne-Blast circuit breakers, GE recommended that no corrective or preventive action be taken, pending completion of the ANSI standards testing.
- From reports by York Daily Record and NRC documents

July 21, 2005 - Inspection finds only 'Green' problems

An inspection of the Peach Bottom Atomic Power Station resulted in two findings of "very low safety significance" that were categorized as Green by the NRC. Neither finding was cited, according to the report.
A report on the inspection by the Nuclear Regulatory Commission stated that Peach Bottom staff identified "inadequate procurement of quality services for the commercial grade dedication of the Unit 3 high pressure coolant injection(HPCI) electronic flow controller." The report explained the internal power supply was not properly identified for replacement to "preclude any age-related degradation" and failed while installed in the Unit 3 HPCI.
The report said this failure affects the ability to ensure "the availability, reliability and capability of system that respond to an initiating event to prevent undesirable circumstances." A single train system was unavailable for less than three days because of this loss of safety function, the report said.
Another finding showed that procedure instructions prepared but not in a timely manner, upon discovery of an inoperable component and leakage of a component boundary for

Unit 2. The leak was repaired and Unit 2 returned to service, the report said, explaining why, though the finding was considered "greater than minor" that there was no citation.
-Report by Marlene Lang

Aug. 30, 2005 -Peach Bottom's mid-cycle performance review receives a 'White' rating for three shutdowns in 12 quarters

The Peach Bottom Atomic Power Station Unit 2 had what the NRC terms "three scrams" with a "loss of normal heat removal" all within 12 calendar quarters, the plant earned itself an unusual White Performance Indicator (PI).
A SCRAM is an industry acronym representing a nuclear reactor shutdown (Skived Coke Rod Adversive Motion).
All of the other findings by inspectors were classified as Green, and considered of "very low safety significance."
-Report by Marlene Lang

Sept. 12, 2005 - NRC inspectors: No findings of significance at Peach Bottom

The Nuclear Regulatory Commission released a report on its most recent inspcection of the Peach Bottom Atomic Power Station, saying no findings of significance were identified, but adding that minor problems were found.
The report went on to explain that "causal evaluations for equipment issues and events reasonably identified the causes of the problem and developed appropriately corrective actions." The report added, "However, for some of the issues affecting human performance, the evaluations were not of sufficient depth to identify the base root cause; therefore, the corrective actions did not prevent further human performance errors of a similar nature."
In two cases, read the report, "operability determinations did not consider all the applicable information to support the final conclusion that the equipment was operable."
Corrective actions were typically implemented in a timely manner, the inspectors said, but added that they found in one case, "corrective actions were not adequate to correct the problem, and did not prevent reoccurrence."
-Report by Marlene Lang

Sept. 13, 2005 -Peach Bottom 2 nuke exits outage

Exelon Corp.'s 1,112-megawatt Unit 2 reactor at the Peach Bottom nuclear power station in Pennsylvania exited an outage and ramped up to 43 percent of capacity by early Tuesday, the U.S. Nuclear Regulatory Commission said in a report.
- Report by Reuters

Sept. 19, 2005 -In a failure to follow procedures, plant operators entered the Unit 3 reactor's drywell after a reactor shutdown but did not, before entering, collect and analyze a radiation sample for airborne particulate and iodine, as required by code.

The failure could have resulted in worker radiation exposure at unsafe dose levels, said a Nuclear Regulatory Commission report made in January, 2006.
Because the two individuals who entered did not sustain any significant dose, no citation was made and the finding was labeled Green.

Sept. 30, 2005 -Fire barrier systems inadequate in real fires, says NIRS

At a public meeting, Nuclear Regulatory Commission staff "announced their recommendation to the Commission to drop a proposed rule making that would substitute controversial "manual actions" for federally required nuclear power station fire protection requirements on electrical cablling (physical fires, minimal cable separation with automated detection and suppression) vital to shutting down the reactor in the event of a significant fire," according to an industry newsletter.
According to Nuclear Information & Resource Service (NIRS), "Since 1992, NIRS has identified widespread nuclear industry violations where fire barrier systems, ... have dramatically failed standardized industry fire tests and would likely fail to protect reactor safety systems in the event of a real fire."
The NRC subsequently declared the fire barriers "inoperable" for protecting electrical power circuits, control and instrumentation cables used in the event of fire to remotely operate reactor shutdown.
As a result, the NIRS explained in the Oct. 14, 2005 issue of Nuclear Monitor, "the majority of the U.S. nuclear power industry was found to be in violation of safety standards as prescribed under current Code of Federal Regulation."
The report went on to say that "the federal agency (NRC) failed to take effective enforcement action and require that operators become compliant with the current fire protection law by installing qualified fire barriers or maintaining minimal separation requirements between electrical circuits for reactor safety-related equipment.

Oct. 31, 2005 -NRC announces inspection

The NRC informed Exelon Nuclear that it would perform a triennial fire protection baseline inspection in January and February of 2006. A letter stated the NRC would make an information gathering visit the week of Jan. 9 and would perform the onsite inspection the weeks of Jan. 23-28 and Feb. 6-10.

Nov. 1, 2005- Inspectors find three federal code violations, issue no citations

An airborne radiation sampler was not sampling correctly, NRC inspectors discovered during an integrated inspection of the Peach Bottom Atomic Power Station.
The inspection, which was completed Sept. 30, turned up three issues, none of which resulted in a citation.
The radioiodine and particulate sampler is required to be in one of the highest annual average ground level D/Q areas. The report also said that Exelon had failed to conduct vegetation or milk sampling of highest calculated annal average ground level D/Q at the nearest offsite garden. The report did not explain what "D/Q" was an abbreviation for.

The report said the failure could affect "protection of public health and safety from exposure to radioactive materials released into the public domain." However, the finding was considered of "very low safety significance" because "calculations of public dose commitments did not identify andy significant public dose or environmental impacts."
NRC inspectors also found that emergency workers required to use respiratory equipment had not maintained their qualifications. The violation affects readiness, the report stated, which in turn could put public health and safety at risk in a radiological emergency. The matter was deemed of "very low safety significance." Owner Exelon was not cited.
Exelon was not cited, either, after its Peach Bottom staff failed to "implement established procedures adherence standards during recovery from an aborted routine test." Operators did not perform the appropriate portions of the restoration section, did not initiate a temporary procedure change, and did not seek technical support after receiving an unexpected test result, according to the report. The error contributed to a reactor trip, but did not result in a citation because the error did not increase the likelihood of equipment or functions being unavailable, the NRC report stated.
-Report by Marlene Lang

Jan. 22, 2006- Fire watch technician pleads guilty to falsifying records

A contracted employee at the Peach Bottom Atomic Power Station pleaded guilty Jan. 9 to the falsification of records used to safely operate the dual-reactor nuclear power plant. Between Jan. 17, 2005, and March 20, 2005, Tracy David, formerly of Bartlett Service Inc., failed to conduct hourly fire watch inspections in multiple sections of the plant including the emergency diesel generator room and the cable spreading room. Contacted by telephone, David - a resident of Quarryville, Pa., according to court documents - declined to be interviewed for this story. Based in Plymouth, Mass., Bartlett Services is a subcontractor for the Peach Bottom Atomic Power Station. On 199 occasions, David claimed that she had completed her rounds of fire watch inspections while on duty at the plant, said Neil Sheehan, spokesman for the U.S. Nuclear Regulatory Commission. Last year, both the NRC and plant officials ran independent investigations that uncovered evidence that showed that David had falsified her fire watch inspections and had not completed her rounds.When interviewed by representatives of the NRC's Office of Investigations, David commented that one reason for her accused offense was that she had been disgruntled after being passed over for a promotion, Sheehan said."There were a significant number of fire watches that were missed," he said. "But (the plant) still had fire suppression systems in place."Regardless of the seriousness of the charges, the commission found that the safety significance was low since no fires werereported and each room on David's route was equipped with automatic fire-detection systems, Sheehan said. A fire watch technician walks a predetermined route, checking sections of the plant for smoke or other signs of fire, said Paul Gunter, director of the reactor watchdog project for the Nuclear Information and Resource Service. The technician keeps=records of hourly checks to ensure that each room has beenmonitored at a particular time."The job is pretty monotonous," said April Schilpp, a spokeswoman for the plant. Gunter said his organization has tracked fire protection violations at nuclear power plants since the early 1990s. For many years, Gunter's group has argued for improved fire barriers and other systems rather than rely on

fire watches."(Plants) should put in adequate fire protection features," he said. "You put humans into the picture, there will be an error. Especially with roving fire watches."
The manual fire watch checks serve as a compensatory measure as ordered by the NRC. The commission requires that fire watches be conducted for any room inside a plant that has its fire detectors on automatic but its fire suppression system on manual. At times, a plant may switch its fire suppression equipment to manual if the system proves too sensitive, Sheehan said. Should a fire watch patrol worker spot signs of smoke, the worker would immediately notify the on-site firefighting brigade, he said."It is a very important function," Sheehan said. Along her route, David's duty's took her to the plant's cable spreading room and to the emergency diesel generator room - the site of a small June 2003 fire.

Peach Bottom Atomic Power Station is equipped with four emergency diesel generators that kick on when the plant loses power.

The generators serve as a source of backup energy. They power the plant's vital equipment including systems used to safely shut down the power station, Sheehan said.

In June 2003, NRC inspectors found that plant technicians had not adequately tightened the engine top cover flange joint bolts of an emergency diesel generator during a maintenance procedure.

As a result, lube oil leaked from the joint and caused a small fire on the exhaust manifold during a test.

While no fires occurred during David's shifts, an internal investigation carried out by Peach Bottom Atomic Power Station officials did raise eyebrows concerning David's actions while on the job.

In February, while on duty, David's personal dosimeter sounded when it should not have gone off, Schilpp said. Typically worn around the neck, a dosimeter is a pager-sized piece of equipment that measures and detects radiation.

As part of the plant procedure, when a worker's dosimeter sounds, that person must leave the room and locate a plant technician, Schilpp said.

A quick check found that David had come from an area of the plant that was not part of her route, Schilpp said.

"She was not supposed to anywhere near that area," Schilpp said. "At that point, (the plant) started to question other things."

As part of the investigation, plant officials checked previous dosimeter readings and found that, in some cases, David's scans did not match what they should have been for her predetermined route.

Plant investigators tracked David by her badge, which is needed as a key to enter specific areas of the site.

"The evidence was overwhelming that things were not going right," Schilpp said. "We saw a pattern emerge."

At the onset of its own investigation, the plant alerted the NRC to the situation, she said.
"We self-identified the problem," Schilpp said. "We want people to be doing the things we ask them to do and to fulfill the obligations of our license."
Site officials confronted David with their evidence and conducted an interview to make sure the plant had not been deficient in explaining to the contracted employee what her job had entailed.
"She told us that she fully understood the job," Schilpp said, adding, "We don't want this to happen again."
Peach Bottom notified Bartlett Services that David had not been doing her job as assigned and had falsified fire watch records.
Bartlett Services removed David from her fire watch position at Peach Bottom Atomic Power Station in late March. On April 15, the NRC opened its own investigation.
Since the commission is not a legal or judicial agency, the NRC notified the U.S. Department of Justice of its investigation. The Department of Justice, in turn, accepted the case for potential action.
"If we have findings of a criminal or deliberate nature," Sheehan said, "we refer those to the (U.S. Department of Justice)."
At the guilty plea proceedings held earlier this month, David acknowledged that she had falsified her fire watch records, said Martin Carlson, the assistant U.S. attorney assigned to the case.
A sentencing date for David has not yet been set.
-Report by Sean Adkins of the York Daily Record/Sunday News

Jan. 25, 2006- An integrated inspection of Exelon Nuclear's Peach Bottom Atomic Power Plant documented two violations, neither of which resulted in citation of Exelon by the Nuclear Regulatory Commission.
In a failure to follow procedures, plant operators on Sept. 19, 2005, entered the Unit 3 reactor's drywell after a reactor shutdown but did not, before entering, collect and analyze a radiation sample for airborne particulate and iodine, as required by code.
The failure could have resulted in worker radiation exposure at unsafe dose levels, the report said.
Because the two individuals who entered did not sustain any significant dose, no citation was made and the finding was labeled Green.
Nor was a citation made when NRC inspectors discovered that following a valve replacement, high pressure service water (HPWS) was not adequately tested. The report stated that "The post-maintenance test did not account for the known degraded condition of the 3B residual heat removal heat exchanger HPSW outlet throttle valve. Improper test

control on two occasions did not identify that high pressure service water flow through the section was below the established "design basis" flow.
The finding was categorized as Green, the report explained, because it did not result in a loss of function.
-Report by Marlene Lang

Feb. 10, 2006 - Fire inspection finds nothing significant

A fire protection inspection of the Peach Bottom Atomic Power Station resulted in "no significant findings" by federal inspectors.
A report on the inspection, from the Nuclear Regulatory Commission, dated March 9, 2006, stated that the purpose of the triennial fire protection inspection was to assess whether Peach Bottom owner Exelon had implemented and adequate fire protection program and that "post-fire safe shutdown capabilities have been established and are being properly maintained."
-Report by Marlene Lang

Feb. 19, 2006- Peach Bottom reactor operating after shut down

The operators of Three Mile Island, Peach Bottom and Limerick nuclear power plants are checking their systems for leaks of water laced with tritium, a radioactive isotope linked to cancer.
Chicago-based Exelon Energy Co., which owns the plants, ordered the inspections after water contaminated with tritium was found in the groundwater or in test wells at three of its plants in Illinois. Exelon owns 10 nuclear plants.
The company ordered each plant to conduct inspections of systems that carry tritium-laced water. The inspections will include pipes, pumps, valves, tanks and other equipment, said Ralph DeSantis, a spokesman for AmerGen Energy, the operator of TMI and a subsidiary of Exelon.
Tritium, a radioactive isotope of hydrogen, is a byproduct of the nuclear reaction. In large doses, it has been linked to cancer.
"Our purpose is to ensure that we have a full understanding of the health of our systems that handle tritium and that we have satisfied ourselves ... that our equipment has a high degree of integrity," said Charles Pardee, Exelon's nuclear chief operating officer.
TMI officials have been monitoring tritium since shortly after the 1979 accident that destroyed the Unit 2 reactor. About a dozen monitoring wells are checked at TMI quarterly, DeSantis said.
Higher-than-usual tritium levels were found in a test well at TMI last fall, said David Allard, the director of the state Department of Environmental Protection's Radiation Control Program. The amounts never exceeded 19,000 picocuries per liter of water. The U.S. Environmental Protection Agency allows up to 20,000 picocuries per liter in drinking water. There is no standard for groundwater.
The leak was traced to a sump pump and corrected, Allard said.
Tritium-laced water is routinely released into the Susquehanna River by TMI, where it is diluted.

The DEP monitors the river at Steelton and Columbia. "I'd be very surprised if we ever saw any tritium," Allard said.
Eric Epstein, the chairman of the watchdog group Three Mile Island Alert, called on Exelon to be more aggressive with its well testing.
The EPA describes tritium as one of the least dangerous radioactive substances because it emits weak radiation and usually leaves the body within a month.
-Report by Garry Lenton of the Patriot-News

Feb. 27, 2006 -Fire cause power reduction, 'no threat'

A electrical fired occurred at Peach Bottom's Unit 3 transformer, forcing the plant to reduce power to 50 percent.
Exelon and government officials said the fire posed no threat to the public, as it happened in a non-nuclear area of the plant, shortly after 9 a.m. It was extinguished by 10:32 a.m., officials said.
The fire was traced to a transformer cabinet in the turbine building of the Unit 3 reactor, said April Schlipp, spokeswoman for the plant's owner, Exelon Nuclear.
-Report by Garry Lenton

Feb. 28, 2006 - NRC examing TMI security

The U.S. Nuclear Regulatory Commission plans to investigate the management of the security force at Three Mile Island, focusing on fitness-for-duty issues such as fatigue and sleeping on the job.
The probe, announced in a certified letter delivered to a Patriot-News reporter, was prompted by a story published Jan. 29.
The story reported on a memo in which John Young, head of the Wackenhut security, scolded security supervisors for failing to note that veteran officers were telling new hires safe places to sleep undetected while on duty. Wackenhut is a private security firm hired by plant owner Exelon Nuclear to guard the nuclear station.
The memo also said officers were telling new hires ways to short-cut patrol duties.
Of additional concern to the NRC were reports that security officers were being allowed to work excessive hours. The newspaper documented one person who worked more than 150 hours during a 14-day period, and averaged more than 54 hours a week for more than 10 months.
Since March 2004, AmerGen Energy, the operator of TMI, investigated and disciplined five workers for "inattentiveness to duty." The phrase is used by the industry and regulators to cover an array of conditions, including sleeping. Three of those workers were security officers.
Guards, speaking on the condition of anonymity, said fatigue from long hours and boredom were to blame for the inattentiveness.
Guards work 12-hour shifts at TMI. Federal regulations limit those hours to 16 out of 24; 26 hours out of 48; and 72 out of seven days.
The agency said it will not announce the findings of the probe.

"Due to the nature of the security-related issues ... we are not providing you with further information on this matter," wrote David J. Vito, senior allegation coordinator for the NRC.
-Report by Garry Lenton of the Patriot-News

March 1, 2006- Drop-in inspections planned by state

Prompted by reports of sleeping or inattentive employees at Three Mile Island, the state said it will conduct surprise inspections at least twice a month at Pennsylvania's five nuclear power plants.
The first round of inspections last month found no instances of inattentiveness on the part of control roomoperators or plant security, Gov. Ed Rendell said yesterday.

The state Department of Environmental Protection will continue the inspections through the end of the year. Then the DEP will decide whether to continue the practice, said Ronald Ruman, a department spokesman.
The inspections came shortly after The Patriot-News reported on five cases of inattentiveness at TMI that occurred since March 2004.
Report by Garry Lenton of the Patriot-News

March 2, 2006- NRC notes three shutdowns of Unit 2

Peach Bottom's annual assessment of it nuclear reactors noted that the Atomic Power Station's Unit 2 reactor was shut down three times in 12 quarters, "with a loss of normal heat removal," a rate which resulted in a "White" level performance indicator. White is the second least significant, just above Green.
-Report by Marlene Lang

March 15, 2006 -NRC responds: Incidents unrelated

The NRC's Senior Allegation Coordinator responded to TMIA's Eric Epstein, in a letter, saying that two incidents of workers falsifying records at the Peach Bottom plant were unrelated and did not represent a pervasive problem.
One incident involved a fire-watch report in January 2006. Another, in October 2001, involved falsification of maintenance tests on sirens.
-Report by Marlene Lang

May 3, 2006 - Nuclear Regulatory Commission inspectors found Peach Bottom was not adequately testing it E-2 emergency diesel generator (EDG) air coolant auxiliary pump following shaft packing replacement, according to a report on an inspection completed March 31, 2006.
A post-maintenance test did not account for the higher pressure that occurs in the EDG cooling subsystem when the EDG is operating and the cooling system is pressurized by the attached cooling pump, the NRC report explained. Ten gallons of water leaked on the

floor in the area of the EDG, as a result, and the leak occurred over a 22-hour period on Dec. 27 and 28, 2005.
The report further stated that personnel had "an inadequate understanding of the air coolant auxiliary pump design and the pump's interrelation with the EDG operation," though the information was available to the testers.
The finding was label Green and owner Exelon was not cited, though a plan was made to correct the problem, the report said.
Inspectors also reviewed an event that happen on Jan. 1, 2006, in which a Unit 2 reactor control rod drive (CRD) system flow transmitter failed by "drifting low." This resulted in an increased control rod drive flow as the flow control valve open in an attempt to compensate for the low flow in the CRD system and according to the report, the condition was not immediately identified. Core thermal power increased and operators reduced power while the situation was evaluated. It turned out that the system was not at in overpower condition.
Also noted in the report, on Feb 13, 2006, operators forgot to complete required technical specification tests after a slow start of an emergency diesel generator. They remember three hours later to do the tests, the report stated.
None of the incidents resulted in citations, as they were considered of low safety significance.
-Report by Marlene Lang

May 12, 2006 - The NRC evaluated Emergency Preparedness exercises held April 25 at Peach Bottom's Unit 2 and Unit 3, reporting no findings of significance.

May 17, 2006- After employee falsified records, plant stays in compliance, with firing

The federal Nuclear Regulatory Commission gave its lowest form of enforcement notice to the nuclear power plant in Peach Bottom Township after an investigation into falsified plant records.
Peach Bottom Atomic Power Station sidestepped a more severe infraction from the regulatory agency by identifying and immediately acting on the violation by a contracted employee, the federal commission said in a letter dated May 12.
As part of a backup verification to its fire safety system, Exelon Corp. contracts with Bartlett Service of Massachusetts to enter certain rooms and verify there is no fire or risk of a fire.
Between January and March of 2005, Exelon determined an employee of Bartlett – whom the commission did not name – falsified records on the fire watch logs on almost 200 occasions.
When Exelon realized what had happened, the employee was fired, and the company started its own investigation, along with notifying the proper authorities of the violation.
In the letter to Exelon, the commission said it considered a more severe infraction, but settled on a "non-cited violation." As a result, the power plant must take corrective action to improve the fire watch performance and prevent the violation from happening again

– which the commission noted Exelon had already done a year prior.
"You restored compliance immediately after identification of the violation by terminating the employee," the commission said in the letter, "and by conducting a prompt investigation to review the access records for other contractor fire watch staff that concluded that the individual's action was an isolated case."
The violation was classified at Severity Level IV, the lowest severity level. In comparison, commission spokeswoman Diane Screnci said a Severity Level III violation would have included the consideration of a fine.
Exelon agreed with the level of severity set by the commission, said April Schilpp, a spokeswoman for the Peach Bottom power plant.
-Report by Charles Schillinger of the York Dispatch

June 1, 2006- Inspection turns up one test issue

An NRC inspection completed on April 21, 2006 turned up one low-significance finding, according to a report released June 1.
Inspectors reported that Peach Bottom operators failed to ensure that test procedures for the high pressure coolant injection (HPCI) and the reactor core isolation cooling (RCIC) pump had acceptance criteria incorporating limits from design documents. Failing to stay within the limits for which the pump was designed could degrade the pump to a lower limit could interfere with proper flow and discharge pressure. The finding was not cited and a correction plan was made, the report stated.
-Report by Marlene Lang

June 30, 2006 - The NRC completed an integrated inspection of the Peach Bottom Atomic Power Station with four findings, all rated "Green," and all not cited.
One finding by inspectors involved barrier integrity, according to a report on the inspection, dated July 26, 2006.
Exelon was to compare task performance between its plants at Limerick and Peach Bottom, according to company procedures established in 1991, the report stated.
Inspectors found that three out of five job performance measures for Limerick Senior Reactor Operators who handled fuel differed significantly in the way they were performed. The NRC report said the differences should have been explored, but were not, and that the failure could have affected physical design barriers that protects the public from radionuclide releases. The finding was not cited.
In another Green finding, personnel failed to properly implement procedures for a high pressure coolant injection (HPCI) turbine exhaust drain piping.
This failure, the report explained, preventd an HPCI containment isolation valve closure on April 5, 2006. The matter was considered of very low safety significance because it did not represent an actual open pathway in the physical integrity of the barrier.
There was also a finding that affected emergency preparedness. Inspectors found a ready-for-use self-contained emergency apparatus in the main control room which had a partially separated regulator air diffuser. The finding was categorized as Green.

In a violation of NRC requirements that one residual heat removal (RHR) shutdown cooling system (for high water level) be operable and in operation during a shutdown, and this was not the case in instances in September 2002 and 2003. No citation was made as there were no actual safety consequences caused by the failure.
-Report by Marlene Lang

July 24, 2006 - NRC responds to fire watch concerns: There is no chronic problem

A Nuclear Regulatory Commission official responded to Eric Epstein's June 12, 2006 letter, in which Epstein ask whether the NRC believed there were a chronic problem at Peach Bottom regarding missed fire watches.
The NRC stated they did a historic review of missed fire watches at the plant and that no chronic problem was found.
Epstein was also told that there was no adverse issue with documentation falsification, after an inquiry.
Epstein asked about a matrix being used to reach these conclusions and the NRC stated it did not use a "matrix" but instead made inspections and reviews.
-Report by Marlene Lang

Aug. 16, 2006- 'Unusual Event' Declared, Terminated at Peach Bottom Plant in York County

Exelon Nuclear's Peach Bottom Atomic Power Station's fire brigade extinguished a small fire onsite yesterday after a backup emergency diesel generator's exhaust gasket on the roof of the diesel generator building unexpectedly caught fire.
The fire occurred during routine testing of one of the station's four diesel generators. The fire prompted the declaration of an Unusual Event at 6:14 p.m. Tuesday, in accordance with station procedures, due to a fire in the Protected Area that was not extinguished within 15 minutes. The fire was extinguished at 6:35, and the Event was terminated at 8:40 p.m. No offsite fire responders were needed to extinguish the fire.
There was no threat to the safe operation of the plant, and there was no danger to station personnel.
An Unusual Event is the lowest of four emergency classifications established by the U.S. Nuclear Regulatory Commission. There was no danger to the public during the event and no special action by the public was needed.
Exelon Nuclear notified all appropriate federal, state and local emergency response officials of the Unusual Event.

Oct. 11, 2006 -Reactor back in service

A nuclear power plant reactor in southern York County returned to service yesterday morning after a cracked pipe in the cooling system forced owner Exelon Nuclear to shut the reactor down Saturday night.
The shutdown was the second at the Peach Bottom Nuclear Station in 15 months and the third since 2003.

The reactor, which had been off line for three weeks for refueling and maintenance, was only two hours into its restart when an equipment operator noticed a leak in a pipe used to test the cooling system, said April Schilpp, spokeswoman for the plant.
-Report by Garry Lenton of the Patriot-News

Oct. 20, 2006 - Peach Bottom among nuclear power plants included in study

The Peach Bottom nuclear power plant in Pennsylvania and Seabrook Station in New Hampshire has been chosen as one of six nuclear power plants nationwide to be part of a study of the consequences of an accident that would release radioactivity into the atmosphere.
The other nuclear plants being reviewed are Diablo Canyon in California; Duane Arnold in Iowa; Fermi in Michigan; and Salem in New Jersey. The study is expected to take three years.
"The sites were picked based on the demographics of the surrounding communities and the type of containment used," said Scott Brunnell of the Nuclear Regulatory Commission.
The study will bring together information about how accidents could occur within containment buildings; how containment could be breached; how radioactive plumes could travel; and how effective emergency planning would be**,** Brunnell said.
Ultimately, the criteria developed as a result of this study would be applied to all U.S. nuclear power plants, Brunnell said.
Seabrook Station spokesman Alan Griffith said that all nuclear plants would eventually be reviewed. He said this is an effort on the part of the NRC to update its methodology.
"It will be beneficial to the community because the NRC will be taking a look at emergency planning," Griffith said. "Ultimately, it will be good for all of us."
-Report by the Portsmouth Herald

Feb. 5. 2007- Operators compensate for low system settings

An integrated inspection by the NRC found Peach Bottom workers failed to follow procedure for equipment evaluations involving pressure pulsations going into standby liquid control (SLC) systems in which relief valves were degraded.
According to a report, on Nov. 21, 2006, engineering personnel documented the incorrect setting of SLC pump relief valves. During the rebuild of Peach Bottom's Unit 3 on Nov. 1, 2004, an SLC pump discharge relief valve was incorrectly adjusted from its design setpoint. There were similar setting questions about Unit 2 and engineers determined that Units 2 and 3 SLC systems were degraded and set low, but still operable, with "two compensatory actions" to maintain pressure. The report noted the relief valves were scheduled to be replaced during each unit's next refueling outage.
The finding was considered of very low safety significance and was not cited.
-Report by Marlene Lang

Feb. 28, 2007- Power plant fire not a threat, officials say

An electrical fire at the Peach Bottom nuclear station in southern York County yesterday posed no threat to the plant's operating nuclear reactors, according to company and government officials.
The fire, discovered shortly after 9 a.m. in a non-nuclear area, was extinguished by 10:32 a.m. and there were no injuries, officials said.
The fire was traced to a transformer cabinet in the turbine building of the Unit 3 reactor, said April Schilpp, spokeswoman for the plant's owner, Exelon Nuclear. As a precaution, officials shut down the turbine and cut power to 50 percent.
Company officials were assessing the damages, but they were expected to be minor.
"It should not prevent the plant from operating normally," Schilpp said.
U.S. Nuclear Regulatory Commission spokeswoman Diane Screnci said the plant was stable and that its inspectors were in the plant control room monitoring the situation.
The fire is the ninth at Peach Bottom since 1986, and the second in the Unit-3 turbine buildings, according to a chronology put together by the watchdog group Three Mile Island Alert using NRC documents.
The most recent was a small fire in an emergency backup diesel generator in August, 2004.
"Fires at nuclear power plants are never a welcome development," said TMIA Chairman Eric Epstein. "Older plants with aging parts, like Peach Bottom, require heightened vigilance. The root cause needs to be identified and defeated."
-Report by Garry Lenton of the Patriot-News

March 17, 2007- Fire was electrical

The Pennsylvania Department of Natural Resources reported that it was a breaker that caught on fire at the Peach Bottom plant in February. A spokesman said the fire was electrical in nature.
"They replaced the breaker and verified proper connections and amperages to prevent a recurrence. I have not yet seen the utility's root cause evaluation, but Dennis Dyckman of my staff is following up on this with the plant," according to Rich Janati, of the DEP.

March 20, 2007- A former security manager for Wackenhut Coporation reportedly sent a letter to the Project on Government Oversight, who passed it on the the Office of the Inspector General on March 27. The writer of the letter claimed that Peach Bottom security officers were fatigued from working excessive overtime or 12-hour shifts and would cover for each other so they could take naps of 10 minutes or more during shifts.
According to an NRC memo released Aug. 22, 2008, the letter also indicate the past efforts by the NRC to identify personnel sleeping on duty had failed, and alleged that NRC and Exelon were aware that officers were sleeping while on duty, and said security officers feared retaliation for raising safety concerns.
The memo stated the letter was provided to the Nuclear Regulatory Commission resident inspector at Peach Bottom in March 2007, and that at that time the concerns it relayed were evaluated under the NRC allegation program by the NRC's Region I office, which oversees Peach Bottom.

In August 2007, Region I concluded the concerns were not substantiated and the allegation filed was closed, according to an NRC document.
-Report by Marlene Lang

2007

March 2007- John Jasinski sends the Nuclear Regulatory Commission a letter alleging guards are sleeping throughout the nuclear plant in York County, Pa. The NRC refers the concern to plant owner Exelon and security provider Wackenhut.

March 13, 2007- NRC: 2002 miscue accidental

In 2002, a plant security officer falsified fire watch logs at Peach Bottom Atomic Power Station.
A contracted security officer at Peach Bottom Atomic Power Station - who logged a fire watch he didn't actually perform - did not willfully falsify fire watch records, according to a U.S. Nuclear Regulatory Commission investigation.
In April 2002, a Wackenhut contract security officer did not conduct a required fire watch but indicated on a log sheet that the action had been completed, according to NRC Office of Investigations records.
While investigating an unrelated matter in July 2006, commission investigators learned about the 2002 missed fire watch, said Neil Sheehan, a commission spokesman.
Investigators discovered that the officer believed his missed fire watch would be conducted by another officer during a scheduled tour of that same area. However, the second officer was assigned to cover the area once every four hours and not every hour as required to cover fire watches.

April 11, 2007 -Security guards to receive back wages

The Miami-based company that employs guards at Peach Bottom Atomic Power Station has agreed to pay $129,953 in back wages to 157 workers at the nuclear-powered plant.
A U.S. Department of Labor's Wages and Hour Division investigation found that Wackenhut Corp. paid guards their regular rates of pay regardless of how many hours they worked.
A federal act states that employees must be paid time and a half should they work more than 40 hours per week.
In the case of Wackenhut Corp., the company required security guards to arm themselves prior to the start of their shift, said Leni Uddyback-Forston, a spokeswoman for the U.S. Department of Labor. "The arming-up process could take five to 15 minutes per employee each day" she said. "They were not being compensated for that time."
Also, regular changes to Wackenhut's work schedule resulted in some guards being paid for four hours at their regular rate instead of overtime pay, Uddyback-Forston said.

Wackenhut officers guard both Three Mile Island in Dauphin County and Peach Bottom Atomic Power Station.
A representative from Wackenhut Nuclear Services said he could not comment on the reimbursement of the Peach Bottom Atomic Power Station guards.
Wackenhut has paid more than 90 percent of the back wages owed, Uddyback-Forston said.
The company is in the process of reimbursing the remaining 26 of 157 guards affected, she said.
-Report by Sean Adkins of the York Dispatch

April 19, 2007- Plant owners request 'reduction' to code

Exelon Generation Company and AmerGen Energy Company asked the Nuclear Regulatory Commision for approval of a change to the required Quality Assurance Topical Report, required under federal code. The companies explained the requested changes to the fire protection program represents a "reduction in committment."
The NRC said it would need more information to complete a review of the request.
Federal code requires the NRC Safety Review Committee to inspect and audit the fire protection program, and the NRC asked the companies to describe how the topical report in question "establishes a requirement for the inspection and audit of the fire protection program."
Twelve nuclear power plants would be included in the requested code change.
-Marlene Lang

April 26, 2007- Work hours to be limited for some nuclear plant workers

Security workers and others in critical jobs at the nation's nuclear plants will no longer be allowed to log excessive overtime hours under new rules approved by the U.S. Nuclear Regulatory Commission.
The change in the NRC's "fitness for duty" requirements is meant to reduce fatigue among plant employees and improve safety and security.
Exelon Nuclear, owner of Three Mile Island, Peach Bottom and Limerick nuclear stations in Pennsylvania, and seven other plants nationwide, expects to increase security staffing to reduce overtime.
"Any area where you have 24/7 coverage is most likely to be impacted," said Craig Nesbit, a spokesman for the company.
The regulations, which should go into effect this year, end a policy that allowed plant operators to meet work-hour limits by averaging the hours of dozens of employees. The process allowed some employees to log hundreds of hours of overtime a month. The new rule bases hourly limits on individuals.
The work-hour limits apply to security, maintenance and operations staffers, such as control room operators.
The rule is common sense, said Dave Lochbaum, a nuclear safety expert with the Union of Concerned Scientists, a Washington, D.C.-based watchdog group.
"Groups don't get tired. People do," he said.

David Desaulniers, an NRC staffer who helped shepherd the rule change through a seven-year administrative review, said the revision will improve plant safety.
"I think that what the commission has approved will be a substantial step forward in addressing worker fatigue issues in the future," said Desaulniers, senior human factors analyst for the agency.
The shortcomings of group averaging were evident at TMI, where some security officers employed by Wackenhut Nuclear Services logged 72-hour weeks for six weeks straight last year.
In 2005, TMI officials cited three security workers for being inattentive or sleeping on the job. Each incident occurred during the night shift. Security officers contacted by The Patriot-News at the time said the incidents were not surprising given the overtime officers were being compelled to work.
The NRC rule, which must undergo review by the federal Office of Management and budget before it goes into effect, also:
• Increases the minimum break between shifts from eight hours to 10.
• Establishes training requirements for fatigue management.
• Limits the reasons plant operators may waive the hourly limits.
• Revises drug- and alcohol-testing requirements.

A veteran security officer at TMI employed by Wackenhut welcomed the changes. "It will definitely keep things from getting really bad again like they were in '02 and '03," said the officer, who spoke on the condition that he not be identified.
Another officer, also requesting anonymity, said the change would significantly reduce fatigue. But he remained skeptical of how much leeway employers would have to waive the rules under special circumstances.
Though the NRC establishes the regulations, it does not require plants to obtain agency approval before authorizing a worker to go over the limit.
Eric Epstein, chairman of the Harrisburg-based watchdog group Three Mile Island Alert, had similar concerns. "I believe the standards are contingent upon voluntary compliance," he said. "I see nothing that suggests there will be more aggressive oversight of a new fitness-for-duty program."
-Report by Garry Lenton of the Patriot-News

April 30, 2007- NRC calls fudged fire checks "minor"

The NRC wrote Peach Bottom to report on an investigation of Jan. 19, 2006 incident in which an employer deliberately did not make the fire protection surveillance rounds required, and falsified reports to say the checks were made.
The NRC told Peach Bottom owner Exelon, "Because you are responsible for the actions of your employees, and because the violation was willful, the violation was evaluated under the NRC ... process. .... The NRC considered that the violation, absent willfullness, would be of minor safety significance because the fire safety equipment was maintained in a functional condition."
The report went on to say: "However, the NRC escalated the severity level of Severity Level IV because the violation was a deliberate act."
-Report by Marlene Lang

May 3, 2007 -NRC alerts power plants of fires
Operators told to review fire protection plans

The Nuclear Regulatory Commission informed power plant operators of two fire incidents, and their causes.
On Aug. 15, 2006, at the Peach Bottom Atomic Power Station, combustible, improperly installed roofing materials on an emergency diesel generator caught fire where it came into contact with a steel penetration sleeve which the generator's exhaust passes through.
According to a letter from the NRC to nuclear plant operators, the fire smoldered for about 35 minutes, from the time it was fire identified until it was put out by the plant's in-house fire brigade.
Peach Bottom found that some of the roofing materials were improperly installed back in 1997-98, and were abutting the steel sleeve. The report explained that during an extended run of the emergency generator the steel sleeve "heated to the point that it caused the adjacent roofing materials to ignite." The exhaust stack operates at about 900 degrees Fahrenheit, but asphalt roofing paper burns at about 400 degrees.

Another fire occurred Aug. 18, 2006 at the Beaver Valley Power Station, Unit 1 reactor, during ventilation duct installation, through a concrete wall which served as a contamination barrier. A worker had stuffed combustible cotton rags around the venting, and sealed it with duct tape. When welding began, heat transfer through a metal sleeve box ignited the duct tape and rags.
According to the NRC report, the burning rags and melting plastic fell through the concrete wall opening into the cable vault. Drops of hot burning plastic fell into conduit-protected cables.
There was no continuous fire watch on the cable vault side of the fire barrier, but smoke from the burning plastic activated a smoke detector. The fire burned about six minutes, and was put out by hand, by a worker, the report said.

Nuclear power plants were told to review their fire protection plans with this information in mind. No specific requirements were made, or specific actions required of plants.

May 8, 2007 - Worker faking records was isolated case

Peach Bottom Atomic Power Station has not been cited even though a plant worker falsified records on two occasions, according to the U.S. Nuclear Regulatory Commission.
An NRC investigation substantiated that a low-level worker deliberately falsified fire-protection-surveillance records without the knowledge of plant management, according to an NRC document dated April 30.
Plant officials ran an investigation into the matter and fired the worker, the document states.
Exelon Nuclear checked the records of other operators to determine if anyone else was involved in the falsification of the records. The commission determined that the violation resulted from the isolated actions of one worker.

-Report by Sean Adkins of the York Dispatch

May 15, 2007- NRC finds partial-flow line under full-line use

Peach Bottom Atomic Power Station credited individuals with performing the functions of a "senior operator" who were not actually senior operators (SOs). Technical specifications and federal code require a certain number of hours and functions to be done by SOs. NRC inspectors discovered that another classification of worker was performing tasks which SOs were to be doing, as required under the plant's license.
The finding was classified as Green, with "very low safety significance." Owner Exelon was not cited, according to the NRC report of an inspection that ended March 31, 2007.
The report also noted that a partial-flow flush line (part of a high pressure coolant injection (HPCI)/reactor core cooling line), was being used for full-flow testing. The use, for which the line was not designed, resulted in cracked piping to the torus, which had to be replaced, according to the NRC report.
The finding was called "more than minor" and the report said the issue had been complex to evaluate. The matter was given Green categorization as "the probability of a large early release remained low."
Inspectors also found that procedures for effluent monitoring were inadequately established and maintained. Procedures were not adequate to detect "non-representative sampling of the 'B' train of the main stack particulate effluents sampling system."
The finding potentially affects public health and safety, but was considered of very low safety significance because it did not involve radioactive material. The NRC report also noted that personnel were not trained properly in the procedures.
None of the violations were cited, according to the NRC.
-Report by Marlene Lang

June 26, 2007 -NRC finds 2 violations, untimely corrections, makes no citations

An NRC inspection completed on April 21, 2006 reported that in March 2006 Peach Bottom operators failed to ensure that test procedures for the high pressure coolant injection (HPCI) and the reactor core isolation cooling (RCIC) pump had acceptance criteria incorporating limits from design documents. Failing to stay within the limits for which the pump was designed could degrade the pump to a lower limit could interfere with proper flow and discharge pressure. The subsequent inspection, completed May 18, 2007, found that the March 2006 problem was not corrected.
The NRC inspectors reported that Peach Bottom owner Exelon had not revised the procedure "and had continued to conduct the surveillance test 13 times since the issue was discovered by the NRC."
An Exelon evaluation found the pumps currently met the design basis requirements and were operable, according to the report. "Exelon failed to take prompt corrective actions to address a safety issue in a timely manner," commensurate with safety significance and complexity," the report stated.
The matter did not result in citation because it did not represent a loss of system safety function.

A second violation also did not receive citation. Peach Bottom failed to correct a condition deemed "adverse to quality" for 22 months. The condition was associated with pressure boundary leakage, the NRC report explained. In July 2005 the NRC noted the plant had not promptly evaluated a steam leak on a high pressure coolant injection valve. The NRC report said Exelon "did not take corrective actions to address a safety issue in a timely manner."

July 30, 2007 -Inspection notes failures to follow procedures

The NRC followed up on a fire and other problems at the Peach Bottom Atomic Power Station in a three-month inspection that ended June 30.
No citations were made for three incidents, two of which involved violations of NRC requirements, according to the Nuclear Regulatory Commission report.
An incorrect size matchup on a breaker caused a fire at the '4T4' 480 volt load center, NRC inspectors explained in a report that followed up on the "Unusual Event."
The February 2007 fire was a result of human error, according to the report, which explained that "an incorrect frame size breaker was installed into a cubicle for which it was not sized. This mismatch caused an electrical fault that led to a fire and a plant transient that upset plant stability." Operators responded to the fire and "equipment losses" by cutting reactor power to half its normal rate.
NRC inspectors determined the "root cause" of the fire to be "that standards, policies, and administrative controls were not used." Maintenance technicians did not strictly follow instructions to verify the frame size during the overhaul of a spare breaker.
The finding was labeled Green and "of very low safety significance" because it did not increase the likelihood of a plant shutdown or the likelihood that mitigation equipment functions would not be available.
The report also noted that a missed procedure step in a surveillance test resulted in an unplanned overloading of an emergency diesel generator on March 15, 2007. This also was due to human error, according to the NRC report, which explained that workers did not follow procedure when the overload happened.
Other emergency generators remained operable. The generator that was overloading was out of service for less than the specified outage time allowed, of seven days. The finding was labeled Green and Exelon was not cited.
In a third Green finding, the NRC said operators failed to follow procedures while manipulating a diesel-driven fire pump cooling water valve on May 23, 2007. The improper manipulation led to misalignment of the fire pump cooling water that subsequently damaged the entire engine during operations without cooling water, the report explained. The fire pump was rendered inoperable by the damage to the engine. The report said operators were not provided complete and accurate instruction for cleaning the cooling water strainer, which contributed to the situation. The finding was considered of very low safety significance.
Exelon was not cited.
-Report by Marlene Lang

Aug. 31, 2007 -Performance review by NRC give good marks

The Nuclear Regulatory Commission announced the completion of its performance review of the Peach Bottom Atomic Power Station for the first half of 2007. The report said the plant operated in such a way as not to require any additional NRC oversight beyond the regularly scheduled inspections. Those inspections were outlined in the letter to Exelon president Christopher Crane.
-Report by Marlene Lang

August 2007- File closed on allegation

NRC's Region I office which oversees Peach Bottom closed the file on the allegations made in a letter by a Wackenhut Corp. supervisor that security officers were working too long and taking naps on duty, saying the accusation was unsubstantiated.
-Report by Marlene Lang

September 2007 -News station WCBS in New York provided the NRC Region I office with a videotape that depicted inattentive security officers on duty at the Peach Bottom Atomic Power Station. "The videotape was broadcast on national television and resulted in considerable congressional and public concern," an NRC memo noted in Aug. 2008.

Baltimore Examiner summary of Peach Bottom sleeping guards incidents

March: John Jasinski sends the Nuclear Regulatory Commission a letter alleging guards are sleeping throughout the nuclear plant in York County, Pa. The NRC refers the concern to plant owner Exelon and security provider Wackenhut.

Sept. 10, 2007- WCBS in New York informs the NRC that it has a videotape of guards asleep or nodding off in a “ready room” near the nuclear reactor.

Sept. 21, 2007- An NRC inspection confirms only the 10 guards caught on tape were sleeping — only one of four shifts is implicated.

Nov. 1, 2007- Exelon terminates its contract with Wackenhut and takes over the plant’s security. Whistle-blower Kerry Beal, on leave during the investigation, is not among the Wackenhut guards rehired by Exelon.

Nov. 5, 2007- NRC inspectors follow up at Peach Bottom to ensure Exelon is correcting the problem.
December 2007-2008: NRC pledges to monitor Peach Bottom.
Baltimore Examiner, December 12, 2007

Nov. 28, 2007 -Security issues prompt more inspections for Peach Bottom

Between March and August of 2007, Kerry Beal videotaped 10 of his fellow Wackenhut Corp. officers at the Peach Bottom plant napping in a secure location of the plant while on the job.
Beal reportedly tried to report the incidents within his chain of command on duty, but then turned the tape over to WCBS news in New York.
The incident prompted Exelon to fire Wackenhut from serving at the Peach Bottom plant. Exelon will conduct more inspections and is reviewing whether to continue contracts with Wackenhut for security at Exelon's other nine nuclear power plants.
An NRC investigation also found officers has slept on duty at least four times between February and August 2007. However, the NRC determined that the plant's security program was not significantly degraded as a resulted.
Increased NRC inspections will review the plant's transition to an in-house security force.
-Report by Garry Lenton of the Patriot News

Feb. 5, 2008- Peach Bottom plant repairs safety valve

Peach Bottom Atomic Power Station operators shut down Unit 3 this morning to repair a safety valve.
The valve prevents steam lines to the electric turbine from becoming over-pressurized, said Bernadette Lauer, power station spokeswoman.
In a release, Lauer said the plant's operators are investigating the cause of the equipment malfunction. There was no risk to the public, she said.
Unit 2 continues to operate at full power. Units 2 and 3 are boiling water reactors, and Unit 2 is capable of generating approximately 1,138 net megawatts and Unit 3 is capable of generating approximately1,140 net megawatts.
-Report by York Daily Record/Sunday News

Feb. 8, 2008 -Peach Bottom Atomic Power Station's Unit 3 reactor came back online at 3:30 p.m. Thursday after workers had replaced a safety relief valve that had malfunctioned earlier this week.
Peach Bottom's Unit 2 reactor continued to operate at full power without interruption during the Unit 3 shutdown.
-Report by Sean Adkins of the York Dispatch

Feb. 14, 2008- Inspection finds one violation

An integrated inspection by the NRC found one violation deemed of low safety significance at the Peach Bottom Atomic Power Station, according to a report by the Nuclear Regulatory Commission. Exelon was not cited for the "failure to include the reactor building equipment and floor drain plugs in the scope of the Maintenance Rule

program." Because of this, the station "did not recognize that appropriate preventive maintenance was not being performed," the report stated.
Inspectors noted that the finding indicated a failure of problem identification and resolution, because the procedures did not contain lessons learned from a similar event in February 2007.
-Report by Marlene Lang

March 3, 2008 -Annual Assessment calls for heightened oversight of guards, security

The NRC has called for "additional regulatory oversight" of Peach Bottom's performance, as a result of security officer inattentiveness revealed in the last quarter of 2007. The inspection covered all of 2007 and the plant was found to have performed satisfactorily in areas related to reactor and radiation safety.
However, enhanced oversight will include additional inspections in the areas of security force performance monitoring, corrective actions, safety conscious work environment (SCWE) and completion of commitments.
The Nuclear Regulatory Commission's report on the annual inspection told Exelon that "behaviors and interactions within the security organization did not encourage the free flow of information related to raising safety issues."
This presumably was a reference to media reports that the Wackenhut Corp. security officer who videotaped his fellow officers sleeping on the job, claimed he had tried to report the problem within the work environment and was met with no action, before he gave the recording to local media.
The plant receive a White rating for the violations.
-Report by Marlene Lang
Here is a brief recount of the events which led to the heightened oversight:

December 2007-2008: NRC pledges to monitor Peach Bottom.
Baltimore Examiner, December 12, 2007

April 9, 2008- NRC announcing meeting with Exelon over safety issues

Officials of the Nuclear Regulatory Commission will meet with Exelon Generation Co. representatives to discuss the results of an NRC inspection that focused on "safety conscious work environment" (SCWE). The inspection and the meeting are in response to incidents related to Wackenhut Corp. security offiicers who were found sleeping on the job and the related issue of why incidents were not reported before a worker took a videotape to the media. Wackenhut has provided security guards on a contract basis to several of Exelon's plants, but since the incident, Peach Bottom and others have turned to in-house security.
The NRC requires that license holders, like Exelon, "maintain an environment in which safety issues are promptly identified and effectively resolved and employees feel free to raise safety concerns," according to an NRC announcement of the April 15 meeting.
In another NRC press release the same day, the agency proposed a $130,000 civil penalty against a nuclear power plant in Florida, 30 miles south of Miami, after a 2006 investigation found Wackenhut-employed security officers there sleeping on duty over a

period of two years. The release said that on April 6, 2006, a security officer was seen by an NRC inspector sleeping while on duty at a post in a vital area of the reactor.
-Report by Marlene Lang

May 6, 2008- Fire bridgade 'deficient'

An integrated inspection of the Peach Bottom Atomic Power Station by the NRC ended March 31, 2008 and resulted in one "more than minor" finding that was not cited. According to the report, numerous fire brigade deficiencies were not discussed at a post-drill critique or documented in a fire drill record, resulting in fire brigade deficiencies. Among the undocumented deficiencies: the brigade opened a hot door to a fire area with no protective equipment on; the supervisor gave orders to sway, rather than shut down, lubricating oil pumps during the fire, failing to take the most conservative action as required. This failure went unrecognized by other team members and evaluators. Also, the fire brigade was not aware of the status of the sprinkler system, to ensure that it was actuated, and the team failed to set the ventilation system to remove smoke from the room, until prompted by the drill instructor.
The crew with observed "deficiencies" was one of five on site, and the only one with problems.
The violation was not cited.
-Report by Marlene Lang

May 9, 2008- Emergency exercises assessed, need improvement: FEMA

A regional administrator for FEMA informed Maryland's Director of Emergency Management that the Federal Emergency Management Agency (FEMA) and the Department of Homeland Security held radiological emergency preparedness exercised at Peach Bottom Atomic Power Station on April 22, 2008 and that four deficiencies occurred during the exercises.
One deficiency was that Harford County, Md., emergency operations were not coordinated with other jurisdictions and were not preceded by siren activation.
There were similar coordination problems with Cecil County, Md., where problems arose related to communication with media during an emergency. Maryland municipalities participate in the exercises because of their proximity to the Peach Bottom plant in southern York County, Pa.
-Report by Marlene Lang

May 21, 2008- Inspectors: Required battery test was not being performed

In an NRC Component Design Bases inspection completed April 11, 2008, one violation was identified at the Peach Bottom Atomic Power Station.
According to the NRC's report, Exelon, owner of Peach Bottom, did not verify that certain battery connection resistances were within the limits of technical specifications.
The report stated that Exelon had exempted the inter-tier connections (those between cells using cables vise steel bars) from the testing requirement. When Exelon did perform

the exempted test, it was discovered that one of four cables on a Unit 2 battery was about the specified limit.
An evaluation of the violation showed the degraded connection would not have prevented the battery from fulfilling its safety function, the report stated.
Because safety function was not lost, the finding was given a Green rating and was not cited.
-Report by Marlene Lang

May 27, 2008- Work environment study complete

After heightened oversight and additional inspections following incidents of sleeping guard, the NRC reported on its inspection of 'safety conscious work environment,' (SCWE). Exelon was to resolve work environment issues related to inattentive security guard issue identified in Sept. 2007.
According to the NRC report on the special inspection, 150 employees of the Peach Bottom plant participated in discussions on work environment issues. Inspectors determined that the SCWE survey was conducted in a manner that encouraged candid and honest responses and that survey results compared "favorably with industry norms." Exelon determined that there were some negative perceptions of the Employee Concerns Program among workers, regard confidentiality and effectiveness.
There were also perceptions of inconsistent standards and direction during refueling outages, and Exelon was to address this and other "perceptions" about adverse reaction for raising issues. During focus group meetings, inspectors could not find any instances where retaliation had happened as a result of someone raising safety issues, the report stated.
The report noted that Exelon had already begun the transition to an in-house security force.
The report said Exelon's self-assesment "resulted in a reasonabley complete understanding of the SCWE" at Peach Bottom.
-Report by Marlene Lang

June 5, 2008- Radioactivity dose assessment not adequate, NRC says

Exelon violated federal code by not providing a means to continually assess the impact of the release of radioactive materials, in its 'dose assessment' program. According the a Nuclear Regulatory Commission report on an evaluation of an April 23 emergency preparedness exercise.
The assessment procedures and programs at the Peach Bottom plant limited assessment to only those conditions in which "the fuel clad barrier was lost or potentially lost," with instruction to operators telling them, in fact, not to take dose assessment protective action in cases where there was no loss or potential loss of the fuel clad. the report explained.
The report stated, The (NRC) inspectors observed during the April 23, 2008 exercise that before the fuel clad barrier had been declared potentially lost, a plant release was in progress while radiation readings in the Unit 2 drywell exceeded 600 rad/hour."
Inspectors noted that otherwise, asssessments were being conducted as prescribed.

The finding was classified as Green and of very low safety significance and was not cited, the report stated.
-Report by Marlene Lang

June 25, 2008 -NRC inspectors found three violations of "very low safety significance" in a team inspection completed May 16 at Peach Bottom.
The findings were rated Green and Exelon was not cited. NRC documents specifying the nature of the violations were not available.

July 15, 2008- NRC checks on progress in sleeping guard remedies

The Nuclear Regulatory Commission continued its follow-up response to inattentive security officers and issues related to "safety conscious work environment" (SCWE) with an inspection at the Peach Bottom plant. The June 6, 2008 visit was to determine Exelon's progress in meeting the commitments it made to address the issues.
The inspection looked into the transition from a contracted to an in-house security force, a review of Peach Bottom's evaluation of the "root cause" of the problem and its effectiveness and an inspection of activities related to work environment issues (SCWE).
The NRC reported that no findings of significance turned up in the inspection and all actions to which Peach Bottom committed were considered closed, with two exceptions.
Exelon would have to perform safety conscious work environment surveys at its other plants, and those survey results would have to be discussed.
It also remains for Exelon to submit written confirmation that all items have been completed.
-Report by Marlene Lang

Aug. 12, 2008 -Material found in sprinkler system valve

An integrated inspection of Peach Bottom Atomic Power Station completed on June 30, 2008 by the NRC noted only on finding of "very low safety significance."
The Green level finding was made by maintenance personnel who discovered foreign material inside a supply valve to an automatic 13KV switchgear sprinkler system. The system is important to the plant's fire protection program. The material was removed.
Exelon was not cited.
-Report by Marlene Lang

Aug. 22, 2008- Regional NRC office under review for response to sleeping guards
Office of Inspector General find Region I assessment 'inconsistent'

The NRC Office of the Inspector General reviewed whether its Region I office responded adequately in handling the letter it received in March 2007 alleging security officers were sleeping on the job at Peach Bottom, and concluded the Region I office was inconsistent in its response.
(For background, see Chronology entries beginning March 20, 2007.)
According to a memo from the Inspector General to the Region I office of the NRC, the regional staff received the letter on March 27 and held a board meeting to evaluate it on

March 29 and again on April 11, 2007. Prior to the two board meetings, an NRC engineer had been assigned to check out the relevant history of allegations at Peach Bottom. The engineer returned an e-mail report on March 28, stating there had been three previous allegations in 2005 related to Peach Bottom security; one about overtime and fatigue, one concerning retaliation against security officers and one allegation of security officers sleeping in the towers.

None of the allegations were substantiated, the engineer reported, also noting that there were some inconsistencies in the stories of the sleeping officers because it would be impossible to observe anyone sleeping inside the towers from outside.

The review also discussed an interview the Inspector General's office made of the Wackenhut security manager who made the original report of the inattentiveness. That manager said there was a fear of retaliation among guards, and said he had reported that fear to Exelon and Wackenhut. He also said he told Exelon that conditions in the "ready room" at the Peach Bottom plant were "not conducive to remaining alert." The ready room is an area where officers not on patrol may relax, but are ready to respond as needed.

The manager said he had suggested in a March 2007 letter approaches for catching the sleeping guards.

The Wackenhut manager claimed he had forwarded his concerns to the NRC on behalf of the security officers because they afraid of retaliation if they raised concerns, according to the memo.

NRC's Region I office referred the March 2007 concerns to Exelon in a letter on April 30, 2007. Three concerns were emphasized: 1) guard sleeping on duty, 2) guards fearing retaliation if they reported safety concerns, and 3) that Exelon was aware of the officers sleeping on duty and was not taking action.

Exelon responded in a letter on May 30, 2007, saying the concerns were not substantiated, based on several points. 1) Exelon had measures in place to reduce potential for inattentiveness, such as random radio checks, requirements for officers to walk around every 15 minutes, random observations of officers in the tower post, and supervisor visits twice per 12-hour shift. 2) Interviews did not confirm the allegations, 3) reviews of corrective actions reports did not show reluctance to report safety problems, and 4) officer work hour averages were lower than NRC limits.

The NRC Inspector General office noted that the NRC's May 30, 2007 letter did not contain any documents to support its evaluation of the safety concerns. The memo also explained that the two Exelon investigators who reviewed the March 2007 concerns concluded that the allegations were unsubstantiated. The Inspector General also noted that those Exelon investigators said at the time that they would have liked to have had more information from the Region I office about the concerns. But Region I said, in the past, Exelon had asked for more information when needed.

In May 2007, the Region I Division of Reactor Projects recommended the allegation file be closed, the memo said. The Region I Division of Reactor Safety delved into Exelon's response in a bit more detail, looking at how the random checks were implemented, how often, how many officer were checked and how checks were documented. That director concluded, also in May, that Exelon's response to the safety concerns was reasonable and sufficient in both depth and scope.

However, an engineer for the Division of Reactor Projects noted that Exelon might have interviewed a larger number of personnel, and said that he was unaware, at the time he made his review of Exelon's response to the concerns, that no security officers were interviewed from the team with the allegedly inattentive officers.
NRC's Region I Division of Reactor Safety pointed out that Exelon never explained exactly what was meant by "random observations," whether that meant post checks or visual observation and noted that observation of the Bullet Resistant Enclosure (BRE) tower guards was "not feasible." Others on the Region I staff agreed it would be hard to "sneak up" on BRE guard to check on inattentiveness.
The NRC's Office of the Inspector General found that the NRC's Region I office was "inconsistent" in its assessment of the safety significance of the two allegations, made within six months of each other, expressing similar concerns about inattentive security officers at the Peach Bottom Atomic Power Station. The inconsistencies were in relation to allegations that officers feared retaliation if they reported safety concerns, and the allegation that Exelon was aware that officers were inattentive on duty but did not take action to address the matter.
The Inspector General's report noted that the Region I staff did not question the information they were given by Exelon and did not probe or attempt to verify it.
The NRC memo said that Region I staff could have contacted the former Wackenhut security manager to obtain more specifics, could have provided Exelon with more detailed information, could have provided the information to the NRC's resident inspectors at Peach Bottom for increased monitoring of guard activities, and could have assigned Region I security inspectors to look into the March 2007 concerns during a baseline inspection that took place from April 30 to May 4, 2007.
-Report by Marlene Lang

Aug. 28, 2008- Inspection procedures complete regarding inattentive guards
NRC: Matter closed

The Nuclear Regulatory Commission completed its inspection and review of Peach Bottom's "inattentive security guard events" and concluded that "the licensee (Exelon) has adequately addressed the commitments/actions described in (Confirmatory Action Letter) 1-07-005; the NRC has reasonable assurance that the Peach Bottom facility will continue to be operated safely; and adequate corrective actions have been taken to prevent reoccurrence of the underlying issues that led to the inattentive security officer events."
A letter to Exelon from the NRC said that the company would be expected to fulfill its commitment to conduct "safety conscious work environment" (SCWE) surveys of security organizations at all it nuclear reactor sites it identify any actions that need to be taken, and to inform the NRC by Oct. 31, 2008 of survey completion so that a meeting can be scheduled to discuss the results.
Additionally, the NRC gave Exelon a "White" level safety finding related to the incidents and for having "an ineffective behavior observation program."
-Report by Marlene Lang

Aug. 29, 2008- Supplemental inspection finds nothing 'significant'

Inspectors conclude management of guards was 'inadequate'

An NRC inspection, completed July 25, 2008, examined Exelon's response at Peach Bottom to a previous "White" level finding related to inattentive security officers. The report on the supplemental inspection stated no findings of significance were identified. The report also stated that Exelon's comprehensive evaluation of the security officer inattentiveness issue determined three root causes. They were: 1) Inadequate Exelon management oversight and leadership of Wackenhut Nuclear Security management to ensure appropriate security force perfomance. 2) Wackenhut Nuclear Security failed to provide adequate oversight of security force performance, and 3) an adverse culture of inattentiveness and non-compliance with the behavior observation program existed within the Peach Bottom Atomic Power Station security organization.
The report stated Exelon had addressed the issue acceptably, but the matter would be considered in assessing plant performance in future assessments, through the third quarter of 2008.
-Report by Marlene Lang

Sept. 10, 2008- WCBS in New York informs the NRC that it has a videotape of guards asleep or nodding off in a “ready room” near the nuclear reactor.

Sept. 21, 2008- An NRC inspection confirms only the 10 guards caught on tape were sleeping — only one of four shifts is implicated.

Oct. 10, 2008 -Water leak in containment area not analyzed

NRC inspectors found Peach Bottom Atomic Power Station Unit 1 reactor had failed to perform periodic radiological analysis of water in the containment vessel, as required by federal code.
An inspection conducted in July and August 2008 found that water that had accumulated in the containment vessel on the 87-foot, 9-inch elevation under a removable floor plate in a hallway was not analyzed. The water "intruded" into the Unit 1 containment vessel and the radioactive waste building, the report stated. The water accumulated was less than the code specification limit of 500 gallons. According to the report, the water had been there since "at least January 2005."
The finding was considered a Level IV violation, but was not cited, as Exelon "initiated a plan to restore compliance."
Inspectors also found that Peach Bottom had failed to properly keep records related to decommissioning, not maintaining or referencing the location of all required records "important to the safe and effective decommissioning of the facility." The site file contained a list of "spills and released from 1976 to 2004" but it did not contain other required records and their locations, as code demands.
Owner Exelon was not cited for the Level IV violation.
-Report by Marlene Lang

Nov. 1, 2008- Exelon terminates its contract with Wackenhut and takes over the plant's security. Whistle-blower Kerry Beal, on leave during the investigation, is not among the Wackenhut guards rehired by Exelon.

Nov. 5, 2008- NRC inspectors follow up at Peach Bottom to ensure Exelon is correcting the problem.

A Sept. 30, 2008 inspection of the Peach Bottom Atomic Power Station, Units 2 and 3 by the Nuclear Regulatory Commission found three violations by owner Exelon Generation Company LLC, though no citation were made.
In a self-revealing non-cited violation, a failure to follow procedure was revealed after an emergency service water leak (ESW) was discovered on the E-1 emergency diesel generator (EDG), according to the NRC's report, dated Nov. 13, 2008. The report said the leak "resulted in safety-related equipment being adversely affected."
The NRC determined the finding was of "very low safety significance," or Green level, because it did not represent an actual loss of system safety function.
Also, a transformer fire and petroleum spill were not properly reported to the NRC, according to the NRC report. A Level IV Severity event, NRC inspectors noted the NRC was not notified by the Peach Bottom Power Station of the reportable event on July 23 and 24, 2008. Inspectors found a planned press release and notification of other government agencies concerning the transformer fire and petroleum spill. The NRC report state "the failure to make a required report could adversely impact the NRC's ability to carry out its regulatory mission," and that the event was related to public health and safety as it contributed to the loss of the plant's three offsite power sources. The event was also noted as an environmental protection issue because "it involved the spill of more than minor quantity of oil the required reporting to the state of Pennsylvania."
Because the NRC had been "informally notified," the NRC determined the finding was a non-citation violation.
NRC inspectors also found the Peach Bottom plant did not conduct a sufficient quality assurance program, adequate to identify incorrect gamma spectroscopy analyses of a principal gamma emitting radionuclide used to scale hard-to-detect radionuclides for purposes of waste classification in accordance with 1- CFR 61.55. The report noted, "The failure to conduct a sufficiently robust quality assurance program ... is a performance deficiency that was reasonably within the licensee's ability to foresee and correct." The NRC called the finding "more than minor" because it affect the plant's "cornerstone objective" by failing to identify incorrectly anylyzed samples used to classify radioactive waste for land disposal.
The finding was considered of "low safety significance" because no radiation limits were exceeded, there was no breach of packaging and no certificate of compliance finding, no low-level burial ground non-conformance, and no failure to make notifications or provide emergency notification.
- Report by Marlene Lang

November 13, 2008- NRC inspects Peach Bottom plant, finds three violations, makes no citations

Dec. 10, 2008- Hunters trespass on power plant property
Several hunters were found to be trespassing on company property in the vicinity of the north substation of the Peach Bottom Atomic Power Station.
The incident was classified as an Event of Potential Public Interest (EPPI) by officials, who issued a report for Units 2 and 3 around 1 p.m. on Dec. 10.
The state Department of Environmental Protection Bureau of Radiation Protection was notified along with Military and Veteran Affairs, the Public Utility Commission, state police, officials of Chester, York and Lancaster counties and PEMA's central office.
-Report by Marlene Lang

May 12, 2009- NRC inspection finds plant departed from code in analyzing spent fuel pools

NRC inspectors who completed a quarterly inspection of the Peach Bottom Atomic Power Station on March 31, 2009 found three violations at the plant.

Two were rated “Green” findings but a third was considered a Severity Level IV violation, but none were cited, according to the NRC report of the inspection.

In one case, NRC inspectors reported that inadequate work instructions resulted in a momentary shorting of a terminal lead during maintenance, causing an inadvertent one-hour shutdown of reactor Unit 3. A containment isolation valve signaled the shutdown.

The report explained, “Work instructions allowed technicians to lift and manipulate energized leads on a safety-related pressure switch, without providing any guidance as to the risk and consequences that inadvertent grounding of those leads could cause.”

The report also stated that the failure “could reasonably be viewed as a precursor to a major event.” The valves in question “failed closed,” the report stated, and “did not represent an actual open pathway in the physical integrity of reactor containment.”

The failure to “provide appropriate risk insights” to workers was a human performance and work control issue, according to the inspectors’ report.

This finding was rated Green and was not cited.

In another “Green” inspection finding, a partial shutdown of the Unit 3 reactor occurred on Jan. 26, 2009 when the ‘A’ Wide-Range Neutron Monitoring (WRNM) became inoperable due to “inadequate procedural guidance regarding adjustments to the mean square voltage offset during the outage.”

The same NRC report described workers’ failure to make a “smooth transition” when shutting down the Unit 3 reactor to replace a main transformer, triggering a partial shutdown or “half-scram,” in industry terms.

The full explanation of the incident explained that the neutron monitor read a certain noise as mean square voltage (MSV) fluctuation within the reactor core. To compensate, the MSV was adjusted to a value of 8E9, though the MSV offset cannot be set higher than 3E8. According to the report, a system manager had specifically said this, but personnel performing the work did not “address the comments,” and this mis-adjustment caused the failed “smooth transition” and a sudden shift in the WRNM, which in turn generated the shutdown signal.

An NRC analysis of the incident concluded that the “deficiency,” or cause of the incident was the use of only two, instead of the required three operable WRNMs, on the Reactor Protection System (RPS) trip, when transferring to “Mode 2.”

The Severity Level IV code violation was noted because the Peach Bottom plant had used a spent fuel pool criticality analysis methodology that was not previously approved by the NRC, departing from the code-prescribed method and failing to obtain NRC approval or a license amendment to do so.

The methodology relates to degraded Boroflex in the high density spent fuel storage racks. Peach Bottom was using a formula to calculate density that differed from the federal code’s formula, mixing existing and new methodologies within the system.

The finding could affect the functionality of the fuel barrier (cladding), the report said, but stated the condition was of very low safety significance.

Peach Bottom agreed to correct the problem by coming up with an evaluation method adequate for testing safety of the spent fuel pool storage racks in accordance with federal code.

2010

Sept. 22, 2010 – Plant officials notify NRC at 5:53 p.m. that a number of emergency sirens lost power during a thunderstorm that passed through York County and Harford County, Md. Plant said 21 emergency sirens lost power in York County and eight sirens lost power in Harford County. Because more than 25 percent of the sirens were unavailable, the following agencies were contacted: Pennsylvania and Maryland Emergency Management; Harford and Cecil counties in Maryland; and Lancaster, Chester and York counties in Pennsylvania.

Sept. 30, 2010- On Sept. 30, 2010, the NRC issued a report on an audit conducted on units 2 and 3 during Dec. 16-17, 2009. An audit is conducted every three years to determine whether licensee programs are consistent with industry guidance.

In the audit, the NRC said Peach Bottom implemented NRC commitments on a timely basis for licensing activities and has implemented an adequate program for managing NRC commitment changes. The NRC also found that there were some discrepancies regarding the implementation of some commitments.

The audit found that there was a non-implemented commitment relating to “fuel moving and core loading with secondary containment inoperable (plant shutdown)” at units 2 and

3. The NRC said the licensee did not implement the commitment it received in September 2008, and “did not process a commitment change to evaluate and document this decision.” The NRC said this discrepancy was entered into the licensee’s correction action program.

The audit also found issues relating to the use of Delta Mururoa BLU respiratory suits. . “The licensee indicated that the associated commitments had not been implemented since the suits have not been used” at Peach Bottom, the NRC report said. “However, the NRC staff noted that there was no indication in the commitment tracking system documenting that the site did not have to comply with the commitment until the suits were used.”

The audit found that Peach Bottom had not developed a lesson plan for training, and had partially implemented commitments with the manufacturer for reporting any defects of the suits, and the proper procedures in case the suits begin to lose air, condensation appears on the visor, or the wearer feels unusual warmth.

The audit also found there were complications regarding the use of two tracking systems and inadequacies in the assignment of commitments at the corporate level. ”Corporate and site personnel have access to both systems, but a manual interface is required to coordinate the two systems,” the NRC report said. “The NRC staff identified issues regarding the tracking of fleet wide commitments” at Peach Bottom, the report said. “One such commitment was to revise the placement of dosimetry in response to the use of new

weighting factors for the determination of the deep-dose equivalent for external exposures.’

According to the NRC report, the licensee “found that the commitment had not been routed to the plant site correctly, and therefore, did not appear in the licensee’s search.“ The discrepancy was entered into the plant’s corrective action program, the NRC said.

Oct. 22, 2010 – A helium leak was discovered in a cask that stores spent nuclear fuel. The cask was located within the Unit 3 containment building at the Peach Bottom Atomic Power Station.

According to the NRC, a preliminary review showed “that a leak exists at the weld plug that provides sealing of the drilled interseal passageway associated with the drain port penetration of the cask lid.” It added, “This leak effectively provides a bypass of the main lid outer confinement seal.

Plant officials said they were working with a vendor to repair the leak, and no radiation had been released.

Nov. 10, 2010- The NRC issued its findings from an integrated inspection conducted at Units 2 and 3 at the Peach Bottom plant for the third quarter ending Sept. 30.

Based on the inspection, the NRC said it identified one non-cited violation of very low safety significance. It was entered into the plant's corrective action program.

The finding involved the failure to adhere to technical specifications to make sure that adequate voltage was available to all safety-related components required to respond to a loss-of-coolant accident.

"The licensee must demonstrate that the existing degraded voltage trip setpoints… are adequate to protect and provide the required minimum voltage to all safety-related

equipment," the NRC said. "Since load tap chargers (which plant operator Exelon used in its calculations) are not safety-related and are subject to operational limitations and credible single failures, they cannot be relied on to establish degraded voltage relay setpoints and time delay input for design basis calculations."

The NRC said it informed Exelon that the voltage levels used in its calculations were not correct, and "to show safety-related equipment would be operable during design basis events, the technical specifications degraded grid relay

setpoints must be used." It added that Exelon performed electrical calculations using the most limiting voltage levels allowed by the specs, and "determined that multiple components would not have adequate voltage."

On another matter in the report, the NRC inspectors focused on a Nov. 12, 2009, non-cited violation when Exelon implemented a temporary configuration change without a review that would have likely required a license amendment before its implementation. In response to this incident, the NRC said, "The inspectors concluded that Exelon has identified and taken appropriate actions to resolve the issues …The inspectors reviewed the procedure revision and determined that the new changes were appropriate to address the program gaps that existed in the old revision."

The NRC report also noted there was an unresolved item dealing with potential procedural inadequacies during fuel handling incidents in the reactor core and spent fuel pool from Sept. 18 to Sept. 24, 2010.

"The events appear to be examples where inadequate procedures contributed to fuel handling issues," the NRC said. "This issue will remain unresolved pending completion of Peach Bottom's investigation and cause evaluation processes under the corrective action program."

May 13, 2011 – The NRC said there would be no significant environmental impact with the transfer of low-level radioactive waste from the Limerick Generating Station in southeastern Pennsylvania to a storage facility at the Peach Bottom plant.

Peach Bottom officials initially requested a license amendment to allow the transfer of the waste on Jan. 6, 2010. The waste does not include any transfer of spent nuclear fuel from Limerick.

Exelon operates both nuclear power plants.

The Limerick plant does not have the capacity to store all of the low-level radioactive waste it generates. The NRC noted that the Barnwell disposal facility in South Carolina is no longer available for Limerick, but Peach Bottom has the ability to store a large amount of low-level waste on an interim basis.

In its environmental analysis, the NRC noted that there would be two or three shipments a year from Limerick to Peach Bottom. "The distance between the plant sites is less than the distance that was previously traveled to the Barnwell disposal facility in South Carolina," the NRC noted.

"The staff concludes that the radiological impacts associated with the transportation, handling and storage of low-level radioactive waste at Peach Bottom will not result in a significant impact to plant workers and members of the public," the NRC said.

"The proposed action will not significantly increase the probability or consequences of accidents. No changes are being made in the types of effluents that may be released offsite. There is no significant increase in the amount of any effluent released offsite. There is no significant increase in occupational or public radiation exposure. Therefore, there are no significant radiological environmental impacts associated with the proposed action."

Sept. 18, 2011 – The York Daily Record reported that an injured Peach Bottom worker was transported to York Hospital while wearing a contaminated work glove. The glove was covered by a bag and handled by a radiation protection technician, but was not removed due to the worker's injuries, the newspaper reported. Once the ambulance arrived at the hospital, the glove was removed, tested and transported back to the plant.

No contamination was passed to surrounding areas, Peach Bottom spokesman David Tillman told the newspaper.

The incident occurred while the worker was fixing a valve at Unit 3 of the plant, which was in shutdown mode for maintenance and refueling. The paper said a valve the worker was examining closed on the fingertips of his left hand.

Nov. 10, 2011 – The NRC issued its inspection report for Units 2 and 3 completed for the third quarter ending Sept. 30, 2011.

No findings of significance were identified. However, a licensee-identified violation was determined to be of very low safety significance and was treated as a non-cited violation.

Nov. 17, 2011 – An NRC inspector conducted a routine safety inspection of Unit 1 at the Peach Bottom Atomic Power Station on Oct. 26-27, 2011. Unit 1 is a gas-pooled demonstration power reactor that operated from February 1966 through October 1974, and has been permanently shut down and in safe storage since then.

Based on the inspection, no issues of safety significance were identified, the NRC said in a letter.

Dec. 15, 2011 – The NRC issued a report on the inoperability associated with an offsite power circuit at Units 2 and 3. This situation was confirmed on Nov. 16, 20101, and is a violation of technical specifications

The NRC report said modifications performed in the mid-1990s failed to upgrade the reliability of offsite sources, essentially minimizing redundancies.

Technical specifications require that there be two qualified circuits between offsite transmission networks and Units 2 and 3, the NRC said. “With one offsite circuit inoperable, the inoperable circuit must be returned to an operable status within seven days or the unit must be brought to a hot shutdown condition within 12 hours,” the NRC report said. There were two occasions in 2010 (March and May) when this requirement was not met, the NRC said. There was another period in 2010 as well, but the violation did not exceed seven days.

“There were no actual safety consequences associated with this event,” the NRC said.

Feb. 10, 2012 – The NRC issued its report of the quarterly inspection of Units 2 and 3 for the period ending Dec. 31, 2012. The report said there were four findings, two identified by the licensee Exelon that were of very low safety significance.

One NRC finding involved a failure to establish and implement an adequate quality assurance program regarding effluent and environmental monitoring of Units 2 and 3. "The finding is more than minor because it is associated with the public radiation safety cornerstone attribute of programs and processes," the NRC report said. "The licensee reassessed the dose to members of the public from routine releases and determined that projected doses did not, nor were likely to, exceed applicable limits," the NRC added.

The violation related to the finding "is currently under review by the NRC," the report said.

NRC inspectors said it identified six examples where the effluent and environmental quality assurance program was ineffective. Among the examples: Exelon did not conduct an evaluation of its 2010 land use census results that show a need for additional monitoring stations; Exelon did not conduct an assessment of its long-term meteorological data to compare the 2010 results against long-term averages; Exelon's failure to evaluate its first, second and third quarter 2011 inter-laboratory samples to determine if sample analyses met applicable quality assurance requirements; and a failure to conduct its onsite biennial evaluation for liquid tritium analysis during its second quarter 2011 sample activity.

"The failure to establish, implement and maintain such a quality assurance program were reasonably within Exelon's ability to foresee and should have been prevented," the NRC said.

The NRC added, "There was no indication of a spill or release of radioactive material on the licensee's site or to the offsite environment that would impact public dose assessments and there was no substantial failure to implement the radioactive effluent release program. There was no effluent monitor calibration issue and the licensee had data by which to assess dose to a member of the public. Exelon plans to provide updated effluent release and dose reports, as necessary, to reflect revised analyses."

Another finding involved Exelon's failure to correct a safety related matter of a motor-operated valve. "Specifically, corrective actions to prevent recurrence of motor-operated valve program testing failures due to degraded stem lubrication in 2009 were not performed in a timely manner to prevent the inoperability of a safety related" valve, the NRC said. It noted that a valve did not develop sufficient thrust during diagnostic testing on Sept. 22, 2011, and "would not have been able to perform its safety function to close during the most limiting design condition."

The report observed that Peach Bottom officials determined that degraded motor-operated valve stem lubrication resulted in four safety-related program failures in March and April of 2009. It was found that the lubricant should be changed, noting that the vendor for the old lubricant canceled production in 2001. At the time, Peach Bottom began a transition to another lubricant for its motor-operated valves, a process that was to be completed by the end of 2014.

By the end of 2011, 128 of the 182 motor-operated valves had been transitioned to a different grease, the NRC report said. Based on a review, 14 motor-operated valves had their conversion dates moved up, and Peach Bottom said it decided to expedite its correction program to complete the transition process by the end of 2013, not 2014.

The NRC report also listed two licensee-identified violations that were of very low safety significance. One involved a failure to perform maintenance that affected an emergency diesel generator. "Specifically, Peach Bottom determined that a damaged lubrication oil drain line should have been identified and replaced during planned maintenance activities prior to the occurrence of leakage," the report said.

Peach Bottom also found that a particular pump was in inoperable during a period of time from April 27, 2010, to Oct. 2, 2011. Officials determined that a leaking relief valve body could have become detached from a residual heat removal suction piping, resulting in the pump's inoperability. Peach Bottom "determined the cause of the delay in identifying the inoperable condition was due to inadequate technical rigor when evaluating the operability of the relief valve on April 27, 2010," the NRC said. The leaking valve was replaced on Oct. 2, 2011.

The NRC also commented on an issue regarding the start time for a 15-minute classification period of a fire. (See previous reports dated Sept. 12, 2011, with both Peach Bottom and Three Mile Island.) The NRC had said the Peach Bottom policy decreased the effectiveness of the plant's emergency plan. The NRC said Exelon entered the matter into its corrective action program and implemented a revision. "The inspectors determined that Exelon's response and corrective actions were reasonable and appropriate to address the non-cited violation and finding and their underlying performance deficiency, " the NRC said. "The NRC considers the issue to be closed."

The NRC also observed that Peach Bottom was appropriately identifying and entering issues into its corrective action program. However, the inspectors did note some ominous trends, including issues of industrial safety and equipment reliability.

It noted that there were three Occupational Safety and Health Administration recordable injuries in September 2011, and there were 45 first aid events during the September/October 2011 Unit 3 refueling outage

The report also noted that Peach Bottom submitted five event reports related to degraded or failed equipment from June 1 to Dec. 31, 2011. "The inspectors verified that all of the equipment issues identified … have been entered' the plant's corrective action program, the NRC said.

NRC inspectors also evaluated the performance of an emergency drill on Dec. 5, 2011. No problems were identified.

March 12, 2012 -

July 23, 2012 – The NRC issued a letter to Peach Bottom officials informing them of some security inspection issues in January 2011.

Specifically, the NRC said its Office of Investigations determined that a security lead supervisor and a security officer “willfully falsified security post inspection documentation.” The incidents occurred on Jan 16 and Jan. 25 in 2011, the NRC said.

On these two dates, the NRC said, the lead supervisor did not physically access security posts to conduct inspections that are designed to make sure the security officer is attentive to duties and is free from any condition that would detract from workplace performance. On those two days, the NRC said, the lead supervisor contacted the security officer by phone, and then forged the security officer’s signature on a post inspection form with the security officer’s consent. “Additionally,” the NRC said, “the security officer forged the lead supervisor’s signature on the post activity log with an entry indicating the inspection had been conducted.”

The NRC said the violation was of very low safety significance because, “although the (lead supervisor) did not access the post locations on those occasions to monitor the environmental conditions and to monitor the assigned security officer for attentiveness and signs of fatigue, other (plant) security supervisors inspected those posts both before and after the (lead supervisor) failed to do so. Additionally, when the lead supervisor contacted the security officer by telephone, the security officer answered the telephone.”

The NRC said that corrective actions were take by the plant, including disciplinary action against the lead supervisor and the security officer, and training with security department personnel on the proper procedures for signing logs.

The OI completed its investigation on April 11, 2012.

September 12, 2012

About 50 workers at Peach Bottom nuclear plant exposed to
low levels of radiation
Peach Bottom Atomic Power Station in Peach Bottom
Township. (FILE)
York, PA -
Roughly 50 workers at Peach Bottom Atomic Power Station were exposed to low levels of radiation early Tuesday after a discharge of contaminated steam. At 1 a.m. that morning, workers were loosening a two-inch vent on top of the Unit 2 reactor vessel head when a "puff" of radioactive steam escaped from a flange, said Neil Sheehan, a spokesman for the U.S. Nuclear Regulatory Commission. Radiation monitoring alarms sounded as workers, dressed in bright yellow radiation-protection suits, hurried to close the vent. In total, the length of the release lasted about 2 minutes.

The reactor is offline for a planned refueling outage. About 2,000 contracted or outage workers at the plant will spend the next several weeks completing maintenance work and replacing nearly one-third of the reactor's fuel.

Initially, 51 of the 138 workers stationed in the area of the Unit 2 reactor vessel early Tuesday didn't clear the plant's radiation monitors, meaning that they still registered a higher dose of contamination, Sheehan said. After a change of clothes and a shower, seven of the 51 workers no longer triggered the monitors.

Of the remaining workers, 27 had been exposed to more than 10 millirems of radiation and 17 registered a dose of less than 10 millirems. A millirem is a measure of radiation exposure. One worker came back with a dose of 173 millirems- the highest level of exposure tied to the radioactive
steam, Sheehan said.
"For that employee, follow-up monitoring shows that contamination levels have fallen off and, today, are
almost at the level of being undetectable," said David Tillman, a Peach Bottom spokesman.
The occupational radiation exposure limit for nuclear industry workers is 5,000 millirems per year, Sheehan
said.
The average American citizen is exposed to 610 millirems each year from natural and manmade sources, he
said.
What happened?
On Tuesday morning, as workers disassembled the vent, a step in the process of refueling Unit 2, water
levels inside the reactor were higher than expected, Sheehan said.

Nov. 14, 2012 – The NRC issued its report on its inspection of Units 2 and 3 of the Peach Bottom Atomic Power Station for the third quarter ending Sept. 30.

In the report, the NRC identified one self-revealing finding of very low safety significance. In addition, the report listed one licensee-identified violation determined to be of very low safety significance.

The NRC finding involved the failure of the plant operator to avoid a situation during maintenance activities of the lower pressure coolant injection system at Unit 2.

The incident occurred on July 25, 2012, when electricians were performing an electrical cable pull “for the multiple spurious operations project into the Unit 2 energized low pressure coolant injection swing bus motor control cabinet.” During the pull, lubrication contacted one of the electrician’s gloved hands and caused the hand to suddenly slide up the cable and contact the edge of an adjacent interposing closing relay, the report said. The contact actuated the relay, the report added, resulting in an over current alarm in the control room

The NRC said the potential over-thrust event "called into question the qualification and operability of the valve."

The report added, "The inspectors noted that the workers performed a two-minute-drill to assess the hazards and safety concerns in the work area, but did not consider the possibility of lubrication contacting their work gloves and causing their hands to slip during the cable pull. The inspectors also noted that the operational risk of the cable pull was not communicated to the workers."

The report also mentions a Sept. 11, 2012, review of radiological issues due to the release of steam during the opening of the reactor vent line flange at Unit 2. "A total of 47 individuals received internal uptakes and were whole body counted," the report said. "There was no radioactive release from the rector building due to this event."

The licensee identified violation involved the failure to promptly correct defective welds in the E-3 emergency diesel generator lube oil piping that were identified in 1998. A leak was identified in the piping during surveillance testing on Sept. 3, 2012. Corrective action was taken.

Jan. 29, 2013 – The NRC issued a report of its fourth quarter inspection of the Peach Bottom Atomic Power Station Units 2 and 3. The NRC identified no findings, although it noted that the plant owner, Exelon , identified three matters that were viewed of very low safety significance The NRC said the licensee-identified violations were placed in the company's correction action program and were being treated as non-cited violations.

March 4, 2013 – In an annual assessment letter for 2012, the NRC said it determined that overall, Peach Bottom Units 2 and 3 "operated in a manner that preserved public health and safety and met all cornerstone objectives."

March 12, 2013 – The NRC issued a report on a two-week inspection competed Jan 31, 2013, relating to an application for an operating renewal license for Unit 2. No findings were identified during the inspection.

April 26, 2013 – The NRC submitted a letter to plant operator Exelon seeking additional information relating to a request to increase the maximum power level at Units 2 and 3 from 3,514 megawatts thermal to 3,951 megawatts thermal. The request, the NRC notes, represents an approximate 12.4 percent increase from the current licensed thermal power level.

Exelon submitted the licensee amendment request on Sept. 28, 2012, and supplemented it by letter on Dec. 18, 2012.

May 9, 2013 - The NRC issued its quarterly inspection report of Units 2 and 3 for the period Jan. 1, 2013, to the end of March.

In the report, the NRC identified one finding.stemming from a Feb. 24, 2013, incident when a determination of operability was not made in a timely manner. The issue stemmed from a monthly functional test of the power load unbalance (PLU) circuit. The NRC said the purpose of a PLU circuit is to prevent overspeed of a main turbine.

"Inspectors determined operators had sufficient information, as of 6:15 a.m. on Feb. 24, to make an immediate determination of PLU functionality and subsequent minimum critical power ratio thermal limit impact, and document the basis for their decision." Nonetheless, the NRC inspectors determined that the operators did not follow its procedures that state "operability should be determined immediately upon discovery of a degraded or nonconforming condition, and that the determination should be made without delay and in a controlled manner using the best information possible." The NRC added that the status of the problem was not documented in the conditions report. The issued continued until 10:30 a.m.

"This finding does not involve an enforcement action because no violation of a regulatory requirement was identified," the NRC report said. It added that Peach Bottom entered the matter into its corrective action program.

June 6, 2013 – The NRC issued a directive to 31 U.S. reactors to improve their systems for safely venting pressure from their containment building during potential accidents. Units 2 and 3 at Peach Bottom are affected by the directive.

June 20, 2013 – The NRC issued a special report of an investigation after a instrumentation and controls technician failed to follow posted high radiation area requirements when he crossed a boundary to manipulate a valve on June 28, 2012. During the investigation, the NRC found that the employee deliberately failed to comply with the posted boundary. The investigation was initiated at the behest of plant licensee Exelon.

The NRC said it concluded that the action should be classified as a severity level IV violation, and was treated as a non-cited violation for a variety of reasons. The NRC noted that the radiological conditions did not "actually constitute a high radiation area in accordance with the regulatory definition," but it decided to increase the significance of the violation to security level IV "since it was deliberate and the NRC's regulatory program is based, in part, on licensees and their contractors acting with integrity."

It treated the matter as a non-cited violation because Exelon placed the issue in its corrective action program; it identified the problem and immediately conducted an investigation; the violation was not repetitive; and the violation "did not involve a lack of management oversight and was the result of the isolated action of the employee."

June 25, 2013 – The NRC issued a report on its inspection of Units 2 and 3 relating to the safe operation of the plant.

“The inspectors concluded that Exelon (the plant licensee) was generally effective in identifying, evaluating and resolving problems,” the NRC report said. “Exelon personnel identified problems, entered them into their corrective action program at the low threshold, and in general, prioritized issues commensurate with their safety significance.

“The inspectors concluded that Exelon adequately identified, reviewed and applied relevant industry operating experience to Peach Bottom operations,” the report added.

In addition, the report said that “inspectors did not identify any indication that site personnel were unwilling to raise safety issues, not did they identify any condition that could have had a negative impact on the site’s safety conscious work environment.”

Feb. 4, 2014 – The NRC issued a report on its quarterly inspection at Units 2 and 3 at the Peach Bottom Atomic Power Station. The report covered the period from October through December 2013.

In the report, the NRC said no findings were identified. However, it added that there was one licensee-identified violation that was determined to be of very low safety significance and was being treated as a non-cited violation.

The licensee-identified violation involved setpoint deficiencies with four safety relief valves and one safety valve at Unit 3. Their setpoints were found to be outside the technical specification variance of plus or minus 1 percent. They were within the allowable range of plus or minus 3 percent. The NRC report said this issue was caused by “setpoint drift” and the valves were replaced.

March 4, 2014 – The NRC completed its annual assessment of Units 2 and 3 at the Peach Bottom Atomic Power Station and said the reactors were operated in a “manner that preserved public health and safety and met all cornerstone objectives.”

The NRC added that the two units were within the “Licensee Response Column” of the NRC’s oversight process because all inspection findings had a very low safety significance.

July 16, 2014- The Alpha Cooling Tower had to be shut down due to damaged (burned up) cable on the feed motor power supply. Exelon i currently trying to determine the details on why and how it happened. They have mobilized in house staff in response as well as having reached out to contractors and motor/pump specialist to determine the problem.

Aug.23, 2014 – Both trains for the Peach Bottom Atomic Power Station Emergency Service Water System were declared inoperable on Units 2 and 3 due to a pin-hole wall piping leak,

Oct. 21, 2014 – The NRC conducted an inspection of Unit 1 from Oct. 7-9, 2014. Unit 1 is a high temperature, gas-cooled demonstration power reactor that operated from

February 1966 to Oct. 31, 1974. In the report, the NRC said there were no findings of safety significance.

Nov. 3, 2014 – In a letter to officials of Exelon, the plant's owner, the NRC said it found an apparent violation identified during a security inspection of the Independent Spent Fuel Storage Installation at the Peach Bottom plant. Details were not disclosed.

The letter said the NRC characterized the violation as an escalated enforcement action. However, no civil penalties were imposed.

"Because your facility has not been the subject of escalated traditional enforcement action within either the last two years or the two most recent inspections, the NRC considered whether credit was warranted for corrective action," the NRC said. "The NRC considered that credit is warranted for Exelon's corrective actions taken to address the violation.

"Therefore, in recognition of the absence of previous escalated enforcement action, and to encourage prompt and comprehensive correction of violations," a civil penalty would not be imposed, the NRC said.

Nov. 7, 2014 – The NRC completed a three month inspection ending Sept. 30. In the quarterly report, the NRC listed three findings of very low safety significance that were treated as non-cited violations.

One finding said Exelon, the plant operator, "did not have the ability to implement all provisions of its approved Fire Protection Program." This stemmed from broken electrical wires in a safety-related breaker cubicle associated with the E-2 alternate shutdown panel. "This condition potentially existed for an extended period of time (greater than a year), but was not readily identified by established periodic testing and maintenance procedures," the NRC said. The finding was placed in Exelon's corrective action program.

A self-revealing finding involved a July 11, 2014, incident in which an "eyebolt installed on the end of the discharge check valve swing arm (was found) in contact with a scaffold mid-rail, preventing full closure of the valve." The NRC said, "Operators closed the check valve by pushing the swing arm past the scaffold pole. Operators then removed the eyebolt and verified that full range of motion … was restored. In addition, the scaffold was modified to remove the mid-rail that caused the interference." The NRC said this condition existed from Sept. 16, 2012, until its correction. "Although difficult for an operator performing rounds to visualize the scaffold obstructing the swing arm's path of travel, the inspectors determined that opportunities were missed to identify the event beforehand," the NRC said.

The other finding was that the plant "did not provide the evacuation time estimate to the responsible offsite response organizations by the required date." The NRC said it found Exelon's evacuation time estimates submitted on Dec. 12, 2012, and Sept. 5, 2013, were inadequate. The NRC cited the following examples: there was no allowance for weather factors in speed and capacity reduction; snow removal was not addressed; no bus routes or plans were included in the analysis; and there was no discussion of the means of evacuating ambulatory and non-ambulatory residents. “The inadequate (evacuation time estimates) had the potential to reduce the effectiveness of public protective actions implemented by the offsite response organizations,” the NRC said

March 2, 2015- Joseph Tolle awakened to see a refrigerator still plugged into the wall, swinging above his head. The refrigerator had been on a shelf situated 8 feet high in the security office in the watchtower. The former armed security officer described how that shelf and part of a wall collapsed, causing the refrigerator to fall on his head. "I woke up on the floor and was dizzy and had a headache. My back was hurting. I was knocked unconscious for a period of time," the 26-year-old from Lancaster testified during a Feb. 18 workers' compensation hearing in Lancaster. Tolle was working for Exelon Corp.'s Peach Bottom Atomic Power Station in southern York County when the October incident occurred<http://www.pennlive.com/midstate/index.ssf/2014/11/peach_bottom_security_guard_to.html>. The company had denied his initial claim and so Tolle is pursuing his claim before Judge Robert J. Goduto at a workers' compensation hearing. During the hearing, both parties presented Tolle testified about the incident, had his medical history combed through and explained his current condition. Tolle and Exelon can settle before the judge holds a final hearing in July.

The Occupational Safety and Health Administration, a branch of the U.S. Department of Labor, did not find any wrongdoing on the part of the nuclear plant related to the incident. The plant has been inspected twice in the past 3 years, October 2012 and November 2014. Exelon received a citation from OSHA in October of 2012, which was informally resolved and cost the company a $4,000 fine. No fines were levied following the November inspection. David Tillman, a spokesman for Exelon, said in an email that the company could not comment on the workers' compensation case until a judge has ruled on the case, adding that OSHA found no wrongdoing related to Tolle's case. "In this case, we inspected the officer's work area, put compensatory measures in place and cooperated fully with OSHA during an onsite review," Tillman said, noting that this investigation is completely separate from the workers' compensation case. Tolle described the room at the top of the watchtower as a 9-foot by 9-foot box, containing weapons, vests, radio equipment, a computer and desk. A microwave and refrigerator were sitting on shelves above the computer stand. He entered this room around 3:30 a.m. on Oct. 13 after relieving a co-worker from one of the watchtowers and checking weapons and gun ports, he testified. He started eating his lunch and was reading an article on the Fox News website about Ebola when the refrigerator fell. "I was reading the article, it's a little blurry, but I heard a snap ... I woke up and was scared," Tolle told Goduto. "I thought we might have been attacked. I looked around to see if anyone was in the tower. "He said he experienced pain in his left arm and back and his head hurt, adding that he was extremely dizzy. During the nearly 3½ hours he waited before being transported from the watchtower to Lancaster General Hospital, Tolle said he tried to pull himself up and turn on a light. The wall he used to brace himself collapsed. Since the incident, Tolle said doctors have treated him for traumatic, neurological and orthopedic injuries, but he cannot pay for any ongoing physical therapy to rehabilitate. Jerry Lehocky, Tolle's worker's compensation attorney, said he is working with doctors to get some of Tolle's treatment provided because his doctors say he isn't fit to work. "My balance is really bad. My memory is really bad." Tolle said. "Physically I can't do the job. I can't walk," Tolle testified, adding that he has anxiety and vertigo.

On cross-examination, Tolle told Exelon's attorney Robert Elias that he didn't have any contact with the wall before it or the shelves fell. He said that when he woke up after the refrigerator hit him, he tried to pull himself up to call for help. "I thought I was going to die, to be honest with you," Tolle said in response to Elias' questioning. Elias also questioned Tolle's health history and mental health issues prior to working at the nuclear plant. Tolle revealed that he had to leave the U.S. Air Force after having a heart disorder discovered, as well as having to be treated for anxiety after the military discharge. Tolle's medical records included car crashes in 2009, 2011 and 2013, suffering injuries in 2009, he said. Tolle, who worked at the power plant since June 2011, said he was subjected to physical, psychological, a written test, oral interviews and weapons training, passing them all before getting the job. Ron Calhoon, a workers' compensation attorney in Harrisburg at Calhoon and Associates, who has tried more than 1,000 such cases, said it can take up to a year for case to come to completion once a claim is filed. He noted that the process gives the plaintiff and defendant time to seek medical exams, depose union officials and doctors, among other background information on the case. "A year is not a long time compared to personal injury action in civil court, those can take multiple years," Calhoon said. In 2013, there were 46,630 petitions and remands assigned, with 46,032 judges decisions in workers' compensation claims filed in Pennsylvania, which is on the decline, but has a large impact on the state's workforce. Calhoon said that because workers' compensation insurance is capped at $951 a week no matter how much someone earns, but is generally 2/3 of what someone's wages, it keeps the costs lower and spread across each employee. "I do not think people understand that employees are covering the cost of workers' compensation," Calhoon said. "Most people think it's coming out of employer's pockets. That's the last place it's coming out of."

May 2015- EXECUTIVE SUMMARY
“Leak First, Fix Later” was first published in April 2010. Now nearly five years later, Beyond Nuclear has taken another look at the problem of aging and deteriorating piping systems carrying radioactive liquids that still run under every nuclear power plant. Nuclear power plants have an extensive network of buried piping systems and tanks which transport liquids that contain radioactive isotopes including tritium -- a radioactive form of hydrogen -- and long-lived strontium-90. These piping systems -- defined either as “buried” or “underground” --are not adequately inspected or maintained due to their inaccessibility.

The United States Nuclear Regulatory Commission (NRC) is the federal regulator charged by Congress with the oversight and enforcement of regulations and its licensing agreements governing these nuclear power plants. U.S. reactors continue to experience leaks and spills of radioactive material into groundwater the unmonitored pathways from unknown and unanticipated sources. To date, the nuclear industry and the federal

regulator have failed to focus action plans on how to control and monitor pathways carrying radioactive material to prevent these leaks from occurring.
Instead, despite broad uncertainties, the federal regulator and industry are using predictive and probabilistic models to estimate the remaining service life on uninspected and unmaintained pipes before leaks may be expected to occur.

As early as 1979, the NRC publicly identified the need for the nuclear industry to begin a proactive program of inspections and maintenance for the "Prevention of Unplanned Releases of Radioactivity" from reactors. Now, more than three decades later, the call for preventive action remains totally ignored by both the nuclear industry and its regulator. The only apparent gain is that leaks are being reported. But the nuclear industry is self-reporting these repeated uncontrolled radioactive leaks to groundwater under an industry-led "voluntary initiative" program. In our view, voluntary reporting is not a reliable or acceptable substitute for a comprehensive regulatory program aimed at protecting water resources. Now, five years after our initial 2010 report, Beyond Nuclear has determined that the NRC has failed to mandate any corrective action programs that focus on inspection and maintenance programs aimed at groundwater protection by preventing ongoing radioactive leaks and contamination of water resources.

Leak First, Fix Later: May 2015

Main Findings-The licensing agreement between the nuclear power plant operators and the NRC is determined by General Design Criteria including control of radioactivity including "Criterion 60—Control of releases of radioactive materials to the environment. The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Uninspected, unmaintained and aging buried piping systems at nuclear power plants continue to experience unanticipated and unpredicted radioactive leaks into groundwater. The number of these uncontrolled and unmonitored leaks is increasing. The NRC has failed to mandate any enforcement or corrective action programs that focus on inspection and maintenance programs aimed at groundwater protection by preventing ongoing radioactive leaks and contamination of water resources. The nuclear industry and the federal regulator have failed to focus action plans on how to prevent these leaks from occurring. Instead, the federal regulator and industry are using predictive and probabilistic models to estimate the remaining service life on uninspected and unmaintained pipes before leaks may be expected to occur. The industry "voluntary" actions remain focused on radioactive leak detection, fixing and mopping up after a leak to groundwater as opportunities occur. In fact, the initiative serves more to protect the industry from liability than to protect the water.

Main Recommendations
•Regulatory oversight, authority and enforcement must be restored and strengthened.
•Standardized NRC regulations should require that underground pipes and tanks be promptly replaced so that systems carrying radioactive effluent can be inspected, monitored, maintained and contained in the event of leaks.

•The nuclear industry must be held accountable for radioactive releases to air, water and soil.
•There must be more public transparency describing the source, cause and extent of radioactive releases from nuclear power plants.
•Radiation protection standards must be strengthened and applied consistently nationwide.

June 18, 2015- Radioactive material was detected in a monitoring well in April at an Exelon-owned nuclear power plant in Pennsylvania about 40 miles from Baltimore, according to nuclear regulators. Exelon, the parent company of Baltimore Gas and Electric Co. and the largest owner of nuclear power plants in the United States, notified the U.S. Nuclear Regulatory Commission that it found dangerous levels of tritium, a radioactive isotope of hydrogen, in a monitoring well at Peach Bottom Atomic Power Station on the Susquehanna River in Delta, Pa. The agency said the contamination posed no danger.

June 18, 2015- "I would say there's no cause for concern for people who work at the plant or members of the public," said Neil Sheehan, a spokesman for the NRC. "It's not used by members of the public. We're talking about low levels" of contamination. Exelon found tritium at 37,700 picocuries per liter, higher than the 20,000 picocuries per liter drinking water limit set by the U.S. Environmental Protection Agency

A groundwater monitoring well at the Peach Bottom nuclear power plant in Pennsylvania that tested positive in April 2015 for significant levels of tritium contamination is just the latest example of a decades-long pattern of leaking nuclear reactors and a weak regulatory system that fails to openly address and fix the problem as required in licensing agreements.

These were the conclusions of a Beyond Nuclear investigative report – Leak First, Fix Later: Uncontrolled and Unmonitored Radioactive Releases from Nuclear Power Plants – released today. The 2015 version of the report updates the findings of the first edition, published in 2010.

“Nuclear plant operators and their regulator consistently fail to address and enforce reactor performance requirements to protect the environment and public health,” said Paul Gunter, Director of Reactor Oversight at Beyond Nuclear and the author of the report. “Our research found that U.S. nuclear power plants continue to experience uncontrolled leaks and spills of radioactive water because the buried pipes and tanks that transport and store it remain inaccessible,” Gunter said.

July 29, 2015- 'Disoriented' man who drove up to Peach Bottom Atomic Power Station taken for mental health evaluation

Trooper Rob Hicks, a spokesman for the Pennsylvania State Police, said he does not expect charges to be filed

The "disoriented" man who drove up to a security checkpoint at the Peach Bottom Atomic Power Station on Friday was not arrested, but instead taken for a mental health evaluation by police.

Trooper Rob Hicks, a spokesman for the Pennsylvania State Police, said he does not expect charges to be filed against the man. Hicks said he did not have information including the man's age, or where he is from.

At about 6 p.m., the man drove up to the checkpoint and was displaying "unusual behavior," a spokeswoman for the Peach Bottom Atomic Power Station has said. He did not get past the outer layer of security, and the plant was not shut down.

The Nuclear Regulatory Commission has said the man did not pose any threat to the power plant or its employees.

April 19, 2018 - By letter dated April 19, 2018 (ADAMS Accession Nos. ML18109A116), Exelon Generation Company, LLC submitted five relief requests for Peach Bottom Atomic Power Station Units 2 and 3, that request relief from certain requirements related to reactor pressure vessel internals, containment, nozzles, and threads in flange that are included in the ASME Section XI Code, 2013 Edition.

The NRC staff has reviewed the requests for relief and concluded that they provide technical information in sufficient detail to enable the NRC staff to complete its detailed technical review and make an independent assessment regarding the acceptability of the relief requests in terms of protection of public health and safety and the environment.

Given the lesser scope and depth of the acceptance review as compared to the detailed technical review, there may be instances in which issues that impact the NRC staff's ability to complete the detailed technical review are identified despite completion of an adequate acceptance review.

Based on the information provided in the submittal, the NRC staff has estimated that these relief requests will take a total of approximately 500 hours to complete. The NRC staff expects to complete this review by April 19, 2019, as requested. If there are emergent complexities or challenges in the review that would cause changes to the initial forecasted completion date (greater than a month) or significant changes in the forecasted hours (greater than 25%), the reasons for the changes, along with the new estimates, will be communicated during the routine interactions with the assigned project manager. These estimates are based on the NRC staff's initial review of the application and they could change, due to several factors including requests for additional information, unanticipated addition of scope to the review, and review by NRC advisory committees or hearing-related activities. Additional delay may occur if the submittal is provided to the NRC in advance or in parallel with industry program initiatives or pilot applications.

May 9, 2018 – Letter dated May 9, 2018, the Nuclear Regulatory Commission issued a letter to Senior Vice President, Bryan Hanson of Exelon Generation Company with the subject of: Peach Bottom Atomic Power Station Units 2 and 3 – safety evaluation regarding implementation of mitigating strategies and reliable spent fuel pool instrumentation related to orders EA-12-049 and EA-12-051 (CAC NOS. MF0845, MF0846, MF0849 and MF0850; EPID NOS L-2013-JLD-0017 and L-2013-JLD-0018).

On March 12, 2012, the U.S. Nuclear Regulatory Commission (NRC) issued Order EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond Design-Basis External Events," and Order EA-12-051, "Order to Modify Licenses With Regard To Reliable Spent Fuel Pool Instrumentation," (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML12054A736 and ML12054A679, respectively). The orders require holders of operating reactor licenses and construction permits issued under Title 10 of the *Code of Federal Regulations* Part 50 to modify the plants to provide additional capabilities and defense in depth for responding to beyond-design-basis external events, and to submit for review Overall Integrated Plans {OIPs) that describe how compliance with the requirements of Attachment 2 of each order will be achieved.

By letter dated February 28, 2013 (ADAMS Accession No. ML13059A305), Exelon Generation Company, LLC (Exelon, the licensee) submitted its OIP for Peach Bottom Atomic Power Station, Units 2 and 3 (Peach Bottom), in response to Order EA-12-049. At six-month intervals following the submittal of the OIP, the licensee submitted reports on its progress in complying with Order EA-12-049. These reports were required by the order, and are listed in the attached safety evaluation. By letter dated August 28, 2013 (ADAMS Accession No. ML13234A503), the NRC notified all licensees and construction permit holders that the staff is conducting audits of their responses to Order EA-12-049 in accordance with NRC Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, "Regulatory Audits" (ADAMS Accession No. ML082900195). By letters dated November 22, 2013 (ADAMS Accession No. ML13220A105), and September 23, 2015 (ADAMS Accession No. ML15254A135), the NRC issued an Interim Staff Evaluation (ISE) and an audit report, respectively, on the licensee's progress. By letter dated January 6, 2017 (ADAMS Accession No. ML17006A167), Exelon reported that Peach Bottom, Unit 2, was in full compliance with Order EA-12-049. By letter dated January 5, 2018 (ADAMS Accession No. ML18005A701), Exelon reported that Peach Bottom, Unit 3 was in full compliance with Order EA-12-049, and submitted a Final Integrated Plan for Peach Bottom, Units 2 and 3.

By letter dated February 28, 2013 (ADAMS Accession No. ML13059A390), the licensee submitted its OIP for Peach Bottom, Units 2 and 3, in response to Order EA-12-051. At six- month intervals following the submittal of the OIP, the licensee submitted reports on its progress in complying with Order EA-12-051. These reports were required by the order, and are listed in the attached safety evaluation. By letters dated October 30, 2013 (ADAMS Accession No. ML13295A303), and September 23, 2015 (ADAMS Accession No. ML15254A135), the NRC staff issued an ISE and an audit report, respectively, on the licensee's progress. By letter dated March 26, 2014 (ADAMS Accession No. ML14083A620), the NRC notified all licensees and construction permit holders that the staff is conducting audits of their responses to Order EA-12-051 in accordance with NRC NRR Office Instruction LIC-111, similar to the process used for Order EA-12-049. By letter dated December 15, 2015 (ADAMS Accession No. ML15352A135), Exelon submitted a compliance letter in response to Order EA-12-051. The compliance letter stated that the licensee had achieved full compliance with Order EA-12-051 at Peach Bottom, Units 2 and 3.

The below conclusions provide the results of the NRC staffs review of Exelon's strategies for Peach Bottom, Units 2 and 3. The intent of the safety evaluation is to inform Exelon on whether or not its integrated plans, if implemented as described, appear to adequately address the requirements of Orders EA-12-049 and EA-12-051. The staff will evaluate implementation of the plans through inspection, using Temporary Instruction 2515-191, "Inspection of the Implementation of Mitigation Strategies and Spent Fuel Pool Instrumentation Orders and Emergency Preparedness Communication/Staffing/Multi-Unit Dose Assessment Plans" (ADAMS Accession No. ML15257A188). This inspection will be conducted in accordance with the NRC's inspection schedule for the plant.

Conclusions for Order EA-12-051

In its letter dated December 15, 2015 [Reference 38], the licensee stated that they would meet the requirements of Order EA-12-051 for each unit by following the guidelines of NEI 12-02, which has been endorsed, with clarifications and exceptions, by JLD-ISG-2012-03. In the evaluation above, the NRC staff finds that, if implemented appropriately, the licensee has conformed to the guidance in NEI 12-02, as endorsed by JLD-ISG-2012-03. In addition, the NRC staff concludes that if the SFP level instrumentation is installed at Peach Bottom according to the licensee's design, it should adequately address the requirements of Order EA-12-051.

CONCLUSION

In August 2013, the NRC staff started audits of the licensee's progress on Orders EA-12-049 and EA-12-051. The staff conducted an onsite audit at Peach Bottom in June 2015 [Reference 23]. The licensee reached its final compliance date on November 6, 2017, for Order EA-12-049, and October 21, 2015 for Order EA-12-051, and has declared that both of the reactors are in compliance with the orders. The purpose of this safety evaluation is to document the strategies and implementation features that the licensee has committed to. Based on the evaluations above, the NRC staff concludes that the licensee has developed guidance and designs that, if implemented appropriately, should adequately address the requirements of Orders EA-12-049 and EA-12-051. The NRC staff will conduct an onsite inspection to verify

that the licensee has implemented the strategies and equipment to demonstrate compliance with the orders

<u>May 23, 2018</u> - Letter dated May 23, 2018, the Nuclear Regulatory Commission issued a letter to Senior Vice President, Bryan Hanson of Exelon Generation Company with the subject of: Exelon Generation Company, LLC, Peach Bottom Atomic Power Station unit 1 – NRC Inspection Report No. 05000171/2018001.

On May 7-9, 2018, the U.S. Nuclear Regulatory Commission (NRC) conducted an inspection at the Peach Bottom Atomic Power Station Unit 1. The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and the conditions of your license. The inspection consisted of observations by the inspectors, interviews with personnel, and a review of procedures and records. The results of the inspection were discussed with Pat Navin, Site Vice President, and other members of your organization on May 9, 2018, at the conclusion of the inspection. The enclosed report presents the results of this inspection. No findings of safety significance were identified.

Current NRC regulations and guidance are included on the NRC's website at www.nrc.gov; select **Nuclear Materials; Med, Ind, & Academic Uses;** then **Regulations, Guidance and Communications.** The current Enforcement Policy is included on the NRC's website at www.nrc.gov; select **About NRC, Organizations & Functions; Office of Enforcement; Enforcement documents;** then **Enforcement Policy (Under 'Related Information').** You may also obtain these documents by contacting the Government Printing Office (GPO) toll-free at 1-866-512-1800. The GPO is open from 8:00 a.m. to 5:30 p.m. EST, Monday through Friday (except Federal holidays).

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure(s), and your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC document system (ADAMS), accessible from the NRC website at http://www.nrc.gov/reading-rm/adams.html.

Executive summary of inspection report:

An announced safety inspection was conducted on May 7-9, 2018, at Unit 1. The inspectors reviewed activities related to the safe storage of radioactive material, including site operations, engineering, maintenance, fire protection, plant support activities, management oversight, and corrective action program (CAP) implementation. The inspection consisted of observations by the inspectors, interviews with Exelon personnel, a review of procedures and records, and plant walk-downs. The NRC's program for overseeing the safe operation of a shut-down nuclear power reactor is described in Inspection Manual Chapter (IMC) 2561, "Decommissioning Power Reactor Inspection Program." Based on the results of this inspection, no findings of safety significance were identified.

September 23, 2018 – WGAL News 8 story: Peach Bottom Atomic Power Station Unit 3 Offline For Maintenance.

The Peach Bottom nuclear power plant is located in southern York County. Operators removed Peach Bottom Atomic Power Station Unit 3 from service around 5 p.m. Saturday, to address a steam leak in the dry well. Officials say that technicians will make repairs and conduct inspections before returning the unit to service. Peach Bottom's Unit 2 is not impacted and continues to operate.

Peach Bottom Atomic Power Station is a dual-unit nuclear power plant located on the west bank of the Conowingo Pond (Susquehanna River) in York County, Pa.
The station's two boiling water reactors are capable of powering more than 2.25 million homes and businesses. Both reactors began commercial operation in 1974.

November 15, 2018 - By letter dated November 15, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18150A387), Exelon Generation Company, LLC (EGC, the licensee) requested changes to the Technical Specifications (TSs) for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3, to allow continued operation with two Safety Relief Valves/Safety Valves (SRVs/SVs) out-of-service and to increase the Reactor Coolant System Pressure Safety Limit.

The Nuclear Regulatory Commission's (NRC) staff is reviewing the submittal and has determined that additional information is needed to complete its review. The specific request for additional information (RAI) is provided below. A clarification phone call was held November 15, 2018. As a result of the call, the draft RAIs have been clarified. A response to these RAIs is requested by December 10, 2018.

By application, dated May 30, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18150A387), Exelon Generating Company, LLC submitted a License Amendment Request (LAR) for Peach Bottom Atomic Power Station, Units 2 and 3 (PBAPS). The proposed LAR would revise PBAPS Technical Specifications to allow continued operation with two Safety Relief Valves/Safety Valves (SRVs/SVs) out-of-service and to increase the Reactor Coolant System Pressure Safety Limit (SL).

RAI-SRXB-1: ASME Overpressure Analysis with New Reactor Pressure Safety Limit

Draft GDCs 9, 33 and final GDC 31 require overpressure protection during power operation be provided by relief/safety valves (SRVs/SVs) and protection system. The LAR proposed to raise a new reactor coolant system pressure safety limit so that the impact of the ASME overpressure analysis with 2 SRVOOS can be accepted. To facilitate the staff review, provide the following information associated with the analysis as provided in the LAR:

1. Peach Bottom technical specification bases 2.1.2 indicates the RCS pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes. Please verify the locations for the peak vessel pressure as reported in Tables 1 and 2 of the LAR are consistent with the TS bases.
2. A verification of whether a TRACG statistical pressure adder had been applied to the peak vessel pressure as reported in the Tables 1 and 2 of LAR. Note that it is known that an adder will be applied to the peak steam dome pressure. However, it is not clear if an adder will also be applied to the peak vessel pressure to be reported. Provide justification if the TRACG statistical pressure adder is not applied,
3. Justify that if the steam dome pressure were to approach the proposed reactor steam dome limit of 1340 psig the corresponding peak vessel pressure will still be below the ASME limit of 1375 psig with margin.

November 21, 2018 - Letter dated November 21, 2018, the Nuclear Regulatory Commission issued a letter to Senior Vice President, Bryan Hanson of Exelon Generation Company with the subject of: Peach Bottom Atomic Power Station – Information request for the cyber-security inspection, notification to perform inspection 05000277/2019403 and 05000278/2019403.

On April 1, 2019, the U.S. Nuclear Regulatory Commission (NRC) will begin a team inspection in accordance with Inspection Procedure 71130.10P, "Cyber-Security," issued May 15, 2017, at your Peach Bottom Atomic Power Station (Peach Bottom), Units 2 and

3. The inspection will be performed to evaluate and verify your ability to meet full implementation requirements of the NRC's Cyber-Security Rule, Title 10 of the *Code of Federal Regulations* (CFR) Part 73, Section 54, "Protection of Digital Computer and Communication Systems and Networks." The onsite portion of the inspection will take place during the weeks of April 1, 2019, and April 15, 2019. Experience has shown that team inspections are extremely resource intensive, both for the NRC inspectors and the licensee staff. In order to minimize the inspection impact on the site and to ensure a productive inspection for both parties, we have enclosed a request for documents needed for the inspection. These documents have been divided into four groups.

The first group specifies information necessary to assist the inspection team in choosing the focus areas (i.e., "sample set") to be inspected by the cyber security inspection procedure. This information should be made available via compact disc and delivered to the regional office no later than January 4, 2019. The inspection team will review this information and, by February 1, 2019, will request the specific items that should be provided for review.

The second group of additional requested documents will assist the inspection team in the evaluation of the critical systems and critical digital assets (CSs/CDAs), defensive architecture, and the areas of your plant's Cyber Security Program selected for the cyber security inspection. This information will be requested for review in the regional office prior to the inspection by March 1, 2019

The third group of requested documents consists of those items that the inspection team will review, or need access to, during the inspection. Please have this information available by the first day of the onsite inspection, April 1, 2019.

The fourth group of information is necessary to aid the inspection team in tracking issues identified as a result of the inspection. It is requested that this information be provided to the lead inspector as the information is generated during the inspection. It is important that all of these documents are up to date and complete in order to minimize the number of additional documents requested during the preparation and/or the onsite portions of the inspection.

The lead inspector for this inspection is Eugene (Gene) DiPaolo. We understand that our regulatory contact for this inspection is Dan Dullum of your organization.

December 10, 2018 - Letter dated December 10, 2018, the Nuclear Regulatory Commission issued a letter to Senior Vice President, Bryan Hanson of Exelon Generation Company with the subject of: Peach Bottom Atomic Power Station, Units 2 and 3 – issuance of relief request re: use of ASME code case N-513-4 in lieu of specific ASME code requirements (EPID L-2018-LLR-0039).

By application dated March 26, 2018 (Agency wide Documents Access and Management System Accession No. ML180868110), Exelon Generation Company, LLC (the licensee) submitted a request to the U.S. Nuclear Regulatory Commission (NRC) for a proposed alternative, Relief Request 15R-07, to the requirements of Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, for the Peach Bottom Atomic Power Station (Peach Bottom), Units 2 and 3. The proposed alternative would allow the licensee to use ASME Code Case N-513-4, "Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 or 3 Piping Section XI, Division 1," in lieu of specified ASME Code requirements.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (1OCFR) Section 50.55a(z)(2), the licensee requested to use the alternative on the basis that complying with the specified requirement would result in hardship or unusual difficulty, without a compensating increase in the level of quality and safety.

The NRC staff has reviewed the subject request and finds that the proposed alternative provides a reasonable assurance of structural integrity of the moderate energy piping systems included in ASME Code Case N 513-4. The NRC staff finds that complying with the requirements of the ASME Code, Section XI, would result in hardship or unusual difficulty, without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes, as set forth in the enclosed safety evaluation, that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2). Therefore, the NRC authorizes the use of Relief Request 15R-07 to use ASME Code Case N 513-4 at Peach Bottom, Units 2 and 3, for the fifth 10-year inservice inspection interval, or until such time as the NRC approves ASME Code Case N-513-4 for general use through revision of Regulatory Guide 1.147, Revision 18, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1."

All other requirements of the ASME Code, Section XI, for which relief has not been specifically requested and authorized by NRC staff remain applicable, including a third party review by the Authorized Nuclear Inservice Inspector.

Conclusion of the safety evaluation:

As set forth above, the NRC staff finds that the proposed alternative provides a reasonable assurance of structural integrity of the subject components and that complying with IWC-3120, IWC-3130, IWD-3120, and IWD-3130 of the ASME Code, Section XI, would result in a hardship or unusual difficulty, without a compensating increase in the level of quality and safety. Accordingly, the staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2). ·

Therefore, the NRC authorizes the use of Relief Request ISR-07 to use Code Case N-513-4 at Peach Bottom, Units 2 and 3, for the fifth 10-year ISi interval, or until such time as the NRC approves Code Case N-513-4 for general use through revision of RG 1.147. If the proposed alternative is applied to a flaw near the end of the authorized 10-year ISi interval and the next refueling outage is in the subsequent interval, the licensee is authorized to continue to apply the proposed alternative to the flaw until the next refueling outage.

All other requirements of the ASME Code, Section XI, for which relief has not been specifically requested and authorized by NRC staff remain applicable, including a third-party review by the Authorized Nuclear Inservice Inspector.

December 13, 2018 - Letter dated December 13, 2018, the Nuclear Regulatory Commission issued a letter to Senior Vice President, Bryan Hanson of Exelon

Generation Company with the subject of: Errata for Peach Bottom Atomic Power Station – integrated inspection report 05000277/2018002 and 05000278/2018002 and independent spent fuel storage installation report 07200029/2018002.

The U.S. Nuclear Regulatory Commission (NRC) identified an omission in the original issuance of NRC Integrated Inspection Report 05000277/2018002 and 05000278/2018002 and Independent Spent Fuel Storage Installation Report 07200029/2018002, dated August 13, 2018 (ADAMS Accession No. ML18225A086). Specifically, the inspection report inadvertently omitted the completion of four samples in the Radiation Safety section pertaining to Inspection Procedure 71124.04, "Occupational Dose Assessment." As a result, the NRC is reissuing the report in its entirety to correct this omission. The necessary corrections are reflected in the enclosed revised report.

Inspection Report – Inspection dates April 1, 2018 to June 30, 2018

List of Findings and Violations:

1. Failure to identify and promptly correct a condition adverse to quality concerning battery charger 2B-003-1
    a. The NRC identified a Green non-cited violation (NCV) of 10 *Code of Federal Regulations* (CFR) Part 50, Appendix B, Criterion XVI, "Corrective Action," because Exelon did not identify and promptly correct a condition adverse to quality (CAQ) commensurate with its safety significance concerning the 2BD-003-1 safety-related battery charger. Specifically, Exelon did not appropriately prioritize repairs for a CAQ and, as a result, the 2BD-003-1 battery charger failed to operate when placed in service on June 5, 2018.
    b. Peach Bottom has two independent safety-related 125/250 VDC systems per unit. Each system is comprised of two 125 V batteries, each with its own charger panel consisting of two 100 percent chargers. The safety-related chargers are full wave, silicon controlled rectifiers, suitable for float charging the lead-calcium battery at 2.25 V per cell, and supplying an equalizing charge at 2.33 V per cell. The chargers operate from 480 V, 3 phase, 60 Hz sources supplied from separate 480 V motor control centers and are capable of carrying the normal DC system load and, at the same time, supplying charging current to keep the batteries in a fully charged condition.
    c. On March 5, 2018, IR 4111441 was initiated for Exelon to investigate and troubleshoot a fan failure alarm of the 2BD-003-1 battery charger under Work Order (WO) 4755435. The IR was placed on Exelon's priority work list (PWL) and operators swapped in-service battery chargers to the 2BD-003-2 charger in preparation to conduct troubleshooting on the 2BD-003-01 charger. During the troubleshooting for the fan failure alarm, Exelon's fix-it-now (FIN) department observed a separate condition; the battery fail alarm light was lit when the battery was placed in-service but unloaded.
    d. IR 4116697 was initiated and closed to WO 4755435 to investigate the new issue concerning the lit 2BD-003-01 fail light. Exelon installed a recorder to obtain data on the 2BD-003-1 while in service before swapping back to the 2BD-003-2 to remain in-service. The recorder data was reviewed for both unloaded and full load battery service. IR 4116697

documents that under full load service, the 2BD-003-1 showed no abnormalities in the recorder traces and that the battery fail light extinguished when load was placed on the charger. The IR recommended no additional actions and concluded that the condition was being worked under and could be closed to WO 4755435. Subsequently, the 2BD-003-1 issue was removed from the PWL.

e. However, after March 19, 2018, during review of the in-service unloaded traces identified during troubleshooting, FIN identified that the frequency reading on the silicon-controlled rectifier (SCR) bus was 180 Hz as opposed to the expected 360 Hz. FIN also observed that the gate pulses originating from the negative gate SCR driver board were approximately half the amplitude of the positive driver board, and consequently half the amplitude of what would be expected pulses from the negative board. Additionally, FIN observed that the fail light returned to being lit when the battery was unloaded. Following the troubleshooting, FIN concluded that the negative SCR gate driver board and/or the connectors on the harness of the driver board were degraded. FIN initiated a material request on April 3, 2018, to the station warehouse to obtain an in-stock negative gate SCR driver board for replacement. The inspectors identified that this new information that FIN had noted was not documented in a new IR, nor added to the existing IR 4116697, nor documented in the WO completion notes, but only kept on an unofficial record by the FIN lead technician. Therefore, Exelon missed the opportunity to place the issue back on their PWL, to evaluate the risk of a degraded negative SCR gate driver board, and to have work control assign a due date commensurate with Exelon's Procedure WC-AA-106, Attachment 1, Revision 18, "Priority Screening Matrix." Considering the part was in stock and work could be performed while 2BD-003-01 was not in- service, the inspectors determined it was reasonable for Exelon to have repaired the degraded condition before the condition worsened or the charger was placed back into service.

f. On June 5, 2018, Exelon attempted to place the 2BD-003-01 battery charger in service; however, voltage could not be maintained at 130 VDC. Exelon secured 2BD-003-01, entered Technical Specification (TS) 3.8.4, which required restoration of the Unit 2 DC electrical power subsystem within 2 hours and then to be in Mode 3 within 12 hours. Exelon subsequently placed the 2BD-003-02 battery charger in-service, and exited TS 3.8.4. IR 4144546 was then initiated and troubleshooting recommenced to determine why there was insufficient DC output on 2BD-003-01. Exelon determined that the negative SCR gate driver board had failed rendering 2BD-003-01 inoperable. The negative SCR gate driver board was replaced with the in-stock driver board, the battery charger was tested satisfactorily, and was returned to an operable status with no abnormalities being identified. Exelon subsequently captured the inspectors concerns regarding CAP documentation and prioritization in IR 4149360 written on June 21, 2018.

g. Corrective Actions: Exelon replaced the negative SCR gate driver board and restored the charger. Additionall, Exelon initiated IR 4149360 to address advocating an earlier repair window, communicating troubleshooting results in a formal manner to other departments (operations, work control, maintenance), and ensuring troubleshooting results are documented in a quality record.

h. Corrective Action Reference: IR 4149360

2. On July 13, 2018, the inspectors presented the quarterly resident inspector inspection results to Mr. Matthew Herr, Plant Manager, and other members of the Exelon staff.

December 21, 2018 - Letter dated December 21, 2018, the Nuclear Regulatory Commission issued a letter to Senior Vice President, Bryan Hanson of Exelon Generation Company with the subject of: Peach Bottom Atomic Power Station, units 2 and 3 – issuance of relief request RE: use of ASME code case N-513-3 in lieu of specific ASME code requirements (EPID L-2018-LLR-0040).

By application dated March 26, 2018 (Agencywide Documents Access and Management System Accession No. ML18086B110), Exelon Generation Company, LLC (the licensee) submitted two relief requests (I5R-07 and I5R-08) to the U.S. Nuclear Regulatory Commission (NRC) for proposed alternatives to the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, for the Peach Bottom Atomic Power Station (Peach Bottom), Units 2 and 3. Relief Request I5R-08 proposed an alternative to allow the licensee to use ASME Code Case N-513-3, "Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 or 3 Piping Section XI, Division 1," in lieu of specified ASME Code requirements. (By letter dated December 10, 2018 (ADAMS Accession No. ML18327A062), the NRC authorized the proposed alternative, Relief Request I5R-07.)

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(z)(2), the licensee requested to use the alternative on the basis that complying with the specified requirement would result in hardship or unusual difficulty, without a compensating increase in the level of quality and safety.

The NRC staff has reviewed the subject request and finds that the proposed alternative provides a reasonable assurance of structural integrity of the moderate energy piping systems included in ASME Code Case N-513-3. The NRC staff finds that complying with the requirements of the ASME Code, Section XI, would result in hardship or unusual difficulty, without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes, as set forth in the enclosed safety evaluation, that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2). Therefore, the NRC authorizes the use of Relief Request I5R-08 to use ASME Code Case N-513-3 at Peach Bottom, Units 2 and 3, for the fifth 10-year inservice inspection interval.

All other requirements of the ASME Code, Section XI, for which relief has not been specifically requested and authorized by the NRC staff remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Introduction of report:

By application dated March 26, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18086B110), Exelon Generation Company, LLC (the licensee) submitted a request to the U.S. Nuclear Regulatory Commission (NRC) for a proposed alternative, Relief Request I5R-08, to the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, for the Peach Bottom Atomic Power Station (Peach Bottom), Units 2 and 3. The proposed alternative would allow the licensee to use ASME Code Case N-513-3, "Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 or 3 Piping Section XI, Division 1," in lieu of specified ASME Code requirements.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(z)(2), the licensee requested to use the alternative ASME Code Case N-513-3 to temporarily accept degraded piping on the basis that complying with the specified ASME Code requirement to repair the degraded piping would result in hardship or unusual difficulty, without a compensating increase in the level of quality and safety.

Conclusion of report:

As set forth above, the NRC staff finds that the proposed alternative provides a reasonable assurance of structural integrity of the subject components, and that complying with IWD-3130 of the ASME Code, Section XI, would result in a hardship or unusual difficulty, without a compensating increase in the level of quality and safety. Accordingly, the staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in
10 CFR 50.55a(z)(2). Therefore, the NRC authorizes the use of Relief Request I5R-08 to use ASME Code Case N-513-3 at Peach Bottom, Units 2 and 3, for the fifth 10-year ISI interval.

All other requirements of the ASME Code, Section XI, for which relief has not been specifically requested and authorized by NRC staff remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

December 21, 2018 - Letter dated December 21, 2018, the Nuclear Regulatory Commission issued a letter to Senior Vice President, Bryan Hanson of Exelon Generation Company with the subject of: Peach Bottom Atomic Power Station, Units 2 and 3 – issuance of alternative requests related to the fifth inservice inspection interval (EPID L-2018-LLR-0055, EPID L-2018-LLR-0057, EPID L-2018-LLR-0058 and EPID L-2018-LLR-0059).

By letter dated April 19, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18109A116), as supplemented by letters dated July 31, 2018; September 6, 2018; and November 28, 2018 (ADAMS Accession Nos. ML18109A116, ML18250A068, and ML18337A196, respectively), Exelon Generation Company, LLC (Exelon, the licensee) submitted relief requests to the U.S. Nuclear Regulatory Commission (NRC). Exelon proposed alternatives to certain inservice inspection requirements of the American Society of Mechanical Engineers Boiler & Pressure Vessel Code (ASME Code) for the Peach Bottom Atomic Power Station (Peach Bottom), Units 2 and 3 pursuant to Title 10 of the *Code of Federal Regulations* Section 50.55a(z).

Exelon submitted the following relief requests:

1. I5R-02 − Examination of Inaccessible Surfaces
2. I5R-03 – Use of BWRVIP [Boiling Water Reactor Vessel and Internals Project]

Guidelines

3. I5R-04 − Alternative Nozzle-to-Vessel Weld and Inner Radii Examination
4. I5R-05 − Encoded Phases Array Ultrasonic Examination Techniques
5. I5R-06 − Examination Category B-G-1 Item No. B6.40 Threads in Flange

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(z)(1), the NRC staff concluded, in the enclosed safety evaluation, that Relief Requests I5R-04, I5R-05, and I5R-06 are authorized on the basis that the proposed alternatives provide an acceptable level of quality and safety. The subject relief requests are for the fifth 10-year interval of the inservice inspection program at Peach Bottom, Units 2 and 3, which begins on January 1, 2019, and is currently scheduled to end on December 31, 2028.

Pursuant to 10 CFR 50.55a(z)(2), the NRC staff concluded, in the enclosed safety evaluation, that Relief Request I5R-02 is authorized on the basis that the proposed alternative provides a reasonable assurance of an acceptable level of quality and safety for the subject welds and has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2). The NRC staff finds that, provided the requirements from which relief is requested in I5R-02 stay the same after the fifth inservice inspection interval (third containment inservice inspection) and for the remaining term of the Peach Bottom Renewed Facility Operating Licenses, compliance with such requirements will continue to be a hardship, and the performance of the integrated leak rate testing will continue to provide reasonable assurance of structural integrity and leaktightness for the primary containment drywell penetration N-3.

By letter dated July 18, 2018 (ADAMS Accession No. ML18179A394), NRC authorized the proposed alternative Relief Request I5R-03.

Safety Evaluation Introduction:

By letter dated April 19, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18109A116), as supplemented by letters dated July 31, 2018; September 6, 2018; and November 28, 2018 (ADAMS Accession Nos. ML18109A116, ML18250A068, and ML18337A196, respectively), Exelon Generation Company, LLC (Exelon, the licensee) submitted requests to the U.S. Nuclear Regulatory Commission (NRC). Exelon proposed alternatives to certain inservice inspection (ISI) requirements of the American Society of Mechanical Engineers Boiler & Pressure Vessel Code (ASME Code) for the Peach Bottom Atomic Power Station (Peach Bottom), Units 2 and 3.

Safety Evaluation Conclusion:

the NRC staff finds that the proposed alternative for I5R-04 provides a reasonable assurance of structural integrity of the subject welds and that complying with Code Cases N-702 and N-648-1 of the ASME Code, Section XI, provides an acceptable level of quality and safety. Additionally, the NRC staff concludes that the licensee’s proposed alternative I5R-05 to use UT in lieu of RT using encoded PAUT provides reasonable assurance of structural integrity and leaktightness of Class 1 and 2 ferritic piping welds. Thus, UT, using the procedure described in the submittal of the subject welds, would provide an acceptable level of quality and safety. Also, the NRC staff determines that proposed alternative I5R-06 provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1).

For I5R-02, the NRC staff has reviewed the proposed alternative, and concludes that the alternative proposed by the licensee in Relief Request I5R-02 to use ILRTs (Type A tests) in lieu of compliance with the IWE-1232(a) ASME Code requirements would result in a hardship or unusual difficulty, without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed the regulatory requirements set forth in 10 CFR 50.55a(z)(2). The NRC staff finds that there is reasonable assurance that the integrity of both containments and their respective penetration N-3 remains intact. The staff finds that, provided the requirements from which relief is requested in I5R-02 stay the same after the fifth ISI interval (third CISI) and for the remaining term of the Peach Bottom RFOLs, compliance with such requirements will continue to be a hardship, and the performance of the ILRTs will continue to provide reasonable assurance of structural integrity and leaktightness for the primary containment drywell penetration N-3.

Therefore, the NRC staff authorizes the use of Relief Requests I5R-02, I5R-04, I5R-05, and I5R-06 at Peach Bottom, Units 2 and 3, for the affected components. The fifth ISI interval for Peach Bottom, Units 2 and 3, is currently scheduled to begin on January 1, 2019, and end on December 31, 2028.

All other requirements of the ASME Code, Section XI, for which relief has not been specifically requested and authorized by the NRC staff remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

January 25, 2019 – In an email dated January 25, 2019 from Jennifer Tobin (Project Manager, NRR/DORL/LPL-1) to David Helker with the subject of: Peach Bottom Units 2 and 3 – Request for additional information – Secondary containment LAR (EPID L-2018-LLA-0264)

REQUEST FOR ADDITIONAL INFORMATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION
FOR A LICENSE AMENDMENT REQUEST TO REVISE THE APPLICABILITY OF FUNCTIONS 3 AND 4 IN TECHNICAL SPECIFICATIONS 3.3.6.2 EXELON GENERATION COMPANY LLC
PEACH BOTTOM ATOMIC POWER STATION UNITS 2 AND 3 DOCKET NUMBERS 50-277 AND 50-278
ENTERPRISE PROJECT IDENTIFIER L-2018-LLA-0264

By letter dated September 27, 2018 (Accession No. ML18271A009), Exelon Generation Company, LLC requested to change technical specifications for Peach Bottom Atomic

Power Station Units 2 and 3. The proposed change would modify TSs to help alleviate scheduling difficulties associated with reactor building and refueling floor ventilation system.

The Nuclear Regulatory Commission's (NRC) staff is reviewing your submittal and has determined that additional information is needed to complete its review. The specific request for additional information (RAI) questions are provided below. These questions are being sent to ensure that the questions are understandable, the regulatory basis for the questions is clear, and to determine if the information was previously docketed. A clarification phone call to discuss the draft RAIs was held January 24, 2019, and both RAIs were clarified as a result of the call. A 30-day response time was agreed upon so please provide your response to the RAIs by February 25, 2019.

By letter dated September 27, 2018, Exelon Generation Company LLC, the licensee, proposes to change technical specifications for Peach Bottom Atomic Power Station Units 2

and 3 (Peach Bottom). The proposed change would modify the applicability for technical specifications 3.3.6.2, functions 3 and 4. Specifically, function 3 (reactor building ventilation exhaust radiation - high) would be revised to only be required when function 4 (refueling floor ventilation exhaust radiation - high) is not maintained, and function 4 would be revised to only be required when function 3 is not maintained. Additionally, this change clarifies which standby gas treatment (SGT) subsystems are required to be put into operation or declared inoperable as described in TS 3.3.6.2 condition C for required actions C.2.1 and C.2.2.

During the Nuclear Regulatory Commission (NRC) staff's review of the license amendment request, the NRC staff determined that more information was needed to complete the review.

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.67, "Accident Source Term," allows licensees seeking to revise their current accident source term in design basis radiological consequence analyses to apply for a license amendment under § 50.90. The application shall contain an evaluation of the consequences of applicable design basis accidents previously analyzed in the safety analysis report. Section 50.67(b)(2) requires that the licensee's analysis demonstrates with reasonable assurance that:

1. (i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).
2. (ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) TEDE.
3. (iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident.

10 CFR 50.36, “Technical Specifications,” in part, requires that the technical specifications be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto and includes items in following categories: (1) safety limits, limiting safety systems settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; (5) administrative controls; (6) decommissioning; (7) initial notifications; and (8) written reports.

In the license amendment request the licensee determined that it is acceptable to revise TS 3.3.6.2 functions 3 and 4 applicability as described above in section 2.0 as long as the refuel floor hatch plug remains removed and no other physical obstruction seals the air flow path between the refuel floor and the reactor building. The licensee stated that the absence of the refuel floor hatch cover allows air flow in the spaces between the reactor building and the refuel floor and that for a design basis accident where excessive

radioactive material is released into secondary containment, airborne radioactivity will be drawn down and detected at either the refuel floor exhaust radiation monitors or the reactor building exhaust radiation monitors (whichever set is not isolated) to provide a valid secondary containment isolation signal. The NRC staff agrees that the reactor building and refuel floor air spaces allow air flow when the refuel floor hatch plug is removed. While in this condition, both the refuel floor ventilation exhaust radiation monitor and reactor building ventilation exhaust radiation monitor are able to generate a high radiation secondary containment isolation signal if there is a mechanism to ensure mixing between the reactor building and refuel floor. However, the license amendment request did not discuss if there is a mechanism available in all conditions i.e., normal operations and during a loss of offsite power, to ensure mixing between the reactor building and refuel floor air spaces.

The licensee proposed the addition of footnote (c) to the applicability for TS 3.3.6.2 function 3, *Reactor Building Ventilation Exhaust Radiation – High*, and footnote (d) to the applicability for TS 3.3.6.2 function 4, *Refueling Floor Ventilation Exhaust Radiation - High*. Footnote (c) would state:

Function is only applicable if Function 4 isolation capability is not maintained. Footnote (d) would state:

Function is only applicable if Function 3 isolation capability is not maintained.

The NRC staff reviewed the proposed TS wording to ensure that the allowance of reducing the required functions to either the reactor building ventilation exhaust radiation function or the refuel floor ventilation exhaust radiation function is only allowed when the refuel floor plug is removed. The proposed TS wording doesn’t appear to be limited to when the refuel floor plug is removed and seems to allow reducing the required functions even if the refuel floor plug is installed.

Additionally, the proposed footnotes are worded such that neither function is required by TS 3.3.6.2 during their applicable modes or other specified conditions if isolation capability is maintained. The proposed footnotes seem to essentially negate the requirements for the functions in modes 1, 2, 3 and during movement of recently irradiated fuel assemblies in secondary containment when isolation capability is maintained for both functions.

Because the proposed change to TS 3.3.6.2 is not limited to when the refuel floor plug is removed and essentially seems to negate the function 3 and 4 requirements when isolation capability is maintained for both functions, a revision to the proposed TS change is necessary or a technical evaluation discussing these aspects is needed.

RAI-1

Provide a discussion that explains if there is a mixing mechanism available during all conditions assumed in the licensing basis (i.e., normal operations, during a loss of offsite power, etc.), to ensure mixing between the reactor building and refuel floor air spaces.

In addition, discuss any impacts on the current licensing basis that may result from the mixing and transporting mechanisms with respect to the detector response times for the refuel floor ventilation exhaust and reactor building ventilation exhaust radiation detectors.

This clarification applies to response time delay when the radiation monitor in that affected area is not available and the radiation monitor in the other area is in operation.

RAI-2

Provide a revision to the proposed TS 3.3.6.2 footnotes such that the footnote is: (1) dependent on the refuel floor hatch plug being removed, and (2) identifies that at least one function must be operable during their applicable modes or other specified conditions, either the reactor building ventilation exhaust radiation instrumentation or the refueling floor ventilation exhaust radiation instrumentation, or,

Provide a technical basis for removing these functions from TS 3.3.6.2.

<u>February 13, 2019</u> - In a letter dated February 13, 2019 from Jennifer Tobin (Project Manager, NRR/DORL/LPL-1) to Bryan C. Hanson, SR. VP of Exelon Generation Company with the subject of: Transmittal of Final Peach Bottom Atomic Power Station, Unit 3 – Accident Sequence Precursor Report (Licensee Event Report 278-2018-001).

By letter dated June 21, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18172A260), Peach Bottom Atomic Power Station, Unit 3, submitted Licensee Event Report (LER) 278-2018-001, "Reactor Core Isolation Cooling System Pressure Switch Failure Results in Condition Prohibited by TS [Technical Specifications]," to the U.S. Nuclear Regulatory Commission (NRC) pursuant to Title 10 of the *Code of Federal Regulations* Section 50.73. As part of the Accident Sequence Precursor (ASP) Program, the NRC staff reviewed the event to identify potential precursors and to determine the probability of the event leading to a core damage state. The results of the analysis are provided in the enclosure to this letter.

The NRC does not request a formal analysis review in accordance with Regulatory Issue Summary 2006-24, "Revised Review and Transmittal Process for Accident Sequence Precursor Analyses" (ADAMS Accession No. ML060900007), because the analysis resulted in an increase in core damage probability (6CDP) of less than 1x10-4.

Final ASP Analysis Summary. A brief summary of the final ASP analysis, including the results, is provided below.

Reactor Core Isolation Cooling System Pressure Switch Failure Results in Condition Prohibited by Technical Specifications. This event is documented in LER 278-2018-001.

Executive Summary. On April 22, 2018, the reactor core isolation cooling (RCIC) pump turbine tripped approximately 28 seconds after startup during surveillance testing. The pump had failed to reach rated system pressure and flow. Concurrent with the RCIC pump trip, a turbine high exhaust pressure was received. Local exhaust pressure indicated a pressure of approximately 12 pounds per square inch gauge (psig), which is well below the trip setpoint of 50 psig. The RCIC system was declared inoperable, and TS 3.5.3, "RCIC System," Condition A, was entered, which requires RCIC to be restored within 14 days. Licensee troubleshooting determined one of the two pressure switches had failed, resulting in the RCIC turbine trip. Following replacement of the failed pressure switch and successful testing, the RCIC system was declared operable on April 23, 2018. The licensee determined that corrosion caused by water intrusion had failed the pressure switch sometime between the last successful surveillance test on January 16, 2018, and the RCIC pump failure on April 22, 2018 (96 days). Due to the uncertainty of when (during the 96-day period) the pressure switch failed, a 48-day (t/2) exposure period was used in the best estimate analysis for this event.

This ASP analysis reveals that the most likely core damage scenarios are transients that result in a loss of feedwater with RCIC unavailable and the postulated unavailability of the high-pressure coolant injection and failure of operators to depressurize the reactor. These accident sequences account for approximately 100 percent of the ~CDP for the event. The point estimate ~CDP for this event is 3x 10-6 (internal events), which is considered a precursor in the ASP Program. The seismic contribution for 48-day unavailability of RCIC is ~CDP of 3x10-a (approximately 1 percent of the internal events contribution).

To date, no performance deficiency associated with this event has been identified; therefore, an ASP analysis was performed since a Significance Determination Process evaluation was not performed.

Summary of Analysis Results. This operational event resulted in a best estimate ~CDP of 3x1Q-6. The detailed ASP analysis can be found in the enclosure to this letter.

Reactor Core Isolation Cooling System Pressure Switch Failure Results in Condition Prohibited by Technical Specifications. Event date 4-22-18.

Executive Summary:

On April 22, 2018, the reactor core isolation cooling (RCIC) pump turbine tripped approximately 28 seconds after startup during surveillance testing. The pump had failed to reach rated system pressure and flow. Concurrent with the RCIC pump trip, a turbine high exhaust pressure was received. Local exhaust pressure indicated a pressure of approximately 12 psig, which is well below the trip set point of 50 psig. The RCIC system was declared inoperable and Technical Specification (TS) 3.5.3 Condition A was entered, which requires RCIC to be restored within

14 days. Licensee troubleshooting determined one of the two pressure switches had failed, resulting in the RCIC turbine trip. Following replacement of the failed pressure switch and successful testing, the RCIC system was declared operable on April 23rd. The licensee determined that corrosion caused by water intrusion had failed the pressure switch sometime between the last successful surveillance test on January 16th and the RCIC pump failure on April 22nd (96 days). Due to the uncertainty of when (during the 96-day period) the pressure switch failed, a 48-day (t/2) exposure period was used in the best estimate analysis for this event.

This accident sequence precursor (ASP) analysis reveals that the most likely core damage scenarios are a transients that result in a loss of feedwater with RCIC unavailable and the postulated unavailability of the high-pressure coolant injection (HPCI} and failure of operators to depressurize the reactor. These accident sequences account for approximately 100 percent of the increase in core damage probability (~CDP) for the event. The point estimate ~CDP for this event is 3x 1o-s (internal events), which is considered a precursor under the ASP Program. The seismic contribution for 48-day unavailability of RCIC is ~CDP of 3x1Q-8 (approximately

one percent of the internal events contribution).

To date, no performance deficiency associated with this event has been identified and, therefore, an ASP analysis was performed since a Significance Determination Process (SDP) evaluation was not performed.

Event Details:

On April 22, 2018, the RCIC pump turbine tripped approximately 28 seconds after startup during surveillance testing. The pump had failed to reach rated system pressure and flow. Concurrent with the RCIC pump trip, a turbine high exhaust pressure was received. Local exhaust pressure indicated a pressure of approximately 12 psig, which is well below the trip set point of 50 psig. The RCIC system was declared inoperable and TS 3.5.3 Condition A was entered, which requires RCIC to be restored within 14 days. Licensee troubleshooting determined one of the two pressure switches had failed, resulting in the RCIC turbine trip. Following replacement of the failed pressure switch and successful testing, the RCIC system was declared operable on April 23rd. Additional information is provided in licensee event report (LER) 278-2018-001 (Ref. 1).

Cause:

Water intrusion within the switch enclosure resulted in corrosion and degradation of the switch internals, causing an electrical short of the pressure switch. A diaphragm normally isolates the switch from the instrument line that contains condensed steam from the RCIC turbine exhaust piping. However, a tear in the diaphragm resulted in a small amount of water entering the switch enclosure.

February 13, 2019 - In a letter dated February 13, 2019 from Jonathan Greives, Chief Reactor Projects Branch 4, Division of Reactor Projects to Bryan C. Hanson, SR. VP of Exelon Generation Company with the subject of: Peach Bottom Atomic Power Station – Integrated Inspection Report 05000277/2018004 and 05000278/2018004 and Exercise of Enforcement Discretion.

On December 31, 2018, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Peach Bottom Atomic Power Station (Peach Bottom), Units 2 and 3. On January 11, 2019, the NRC inspectors discussed the results of this inspection with Mr. Pat Navin, Peach Bottom Site Vice President; Mr. Matthew Herr, Plant Manager; and other members of your staff. The results of this inspection are documented in the enclosed report.

NRC inspectors documented one finding of very low safety significance (Green) in this report. Additionally, a violation of Exelon's site-specific licensing basis for tornado-generated missile protection was identified. Because this violation was identified during the discretion period covered by Enforcement Guidance Memorandum (EGM) 15-002, Revision 1, "Enforcement Discretion for Tornado Generated Missile Protection Non-Compliance," (ADAMS Accession No. ML16355A286) and because Exelon is implementing compensatory measures, the NRC is exercising enforcement discretion by not issuing an enforcement action and is allowing continued reactor operation.

If you disagree with a cross-cutting aspect assignment or a finding not associated with a regulatory requirement in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the U. S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I, and the NRC's Resident Inspector at Peach Bottom.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and the NRC's Public Document Room in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Part 2.390, "Public Inspections, Exemptions, Requests for Withholding."

U.S. NUCLEAR REGULATORY COMMISSION Inspection Report

Inspection Dates: October 1, 2018 to December 31, 2018

Summary:

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring Exelon's performance at Peach Bottom Atomic Power Station, Units 2 and 3, by conducting the baseline inspections described in this report in accordance with the Reactor Oversight Process. The Reactor Oversight Process is the NRC's program for overseeing the safe operation of commercial nuclear power reactors. Refer to https://www.nrc.gov/reactors/operating/oversight.html for more information. NRC-identified and self-revealed findings, violations, and additional items are summarized in the table below.

List of Findings and Violations:

Installation of Condensate Pump Cables not in Accordance with Standard

The inspectors identified a self-revealing Green finding because Exelon did not conduct cable replacement in accordance with E-1317, "Wire and Cable Notes and Details Power, Control, Instrument Cables," for the Unit 3 condensate pump transformers. Specifically, Exelon installed new power cables to the condensate pump transformers without proper waterproof protection which resulted in water being entrained in the cable conductors and caused premature cable failure on September 30, 2018. In addition, the condensate pump cable shielding was not grounded in accordance with E-1317, and resulted in a false fault indication which tripped a second condensate pump and resulted in a reactor SCRAM on September 30, 2018.

Additional Tracking Items:

Issue number: 05000277/2017- 001-01

- Emergency Diesel Generator (EDG) Exhaust Stacks Nonconforming Design for Tornado Missile Protection
- On November 1, 2018, it was determined that Peach Bottom's RCIC system and the RHR suppression pool cooling system did not conform with the licensing basis for protection against tornado-generated missiles. Power and instrumentation cabling for RCIC and RHR were identified in rooms adjacent to the Unit 2 and Unit 3 reactor buildings which were not tornado missile protected.
- As a result of the non-conforming condition, on November 1, 2018, the RCIC system and the RHR suppression pool cooling system were declared inoperable for both units. Compensatory measures were put in place and, in accordance with NRC guidance contained in Enforcement Guidance Memorandum (EGM) 15-002, the RCIC and RHR systems were returned to an operable but non-conforming status.
- Corrective Actions: Exelon took immediate compensatory measures which included verifying that procedures are in place, equipment was appropriately staged, and training is current for performing actions in response to a tornado to preserve RCIC and RHR operability.
- Status – Closed

Issue number: 05000278/2018- 003-00

- Automatic Reactor Scram Due to Loss of Two Condensate Pumps
- Status – Closed

Observations:

Inaccessible External Flood Seal Inspections

- In 2012, Exelon performed the required post-Fukushima walkdowns in accordance with Nuclear Energy Institute (NEI) 12-07, "Guidelines for Performing Verification Walkdowns of Plant Flood Protection Features," to confirm the condition of the external flood barrier system. Exelon evaluated the accessibility of the external flood seals using the definition and guidelines in NEI 12-07. As a result, Exelon determined that a population of 186 seals were inaccessible due to configuration or operational constraints and documented a technical justification for reasonable confidence that the seals existed and no inspections were required.
- In 2018, Exelon performed a review of the inaccessible seals and developed methods to access the seals and perform inspections. The project was planned to be performed one building at a time as funding allowed. On August 16, 2018, Exelon performed inspections of electrical conduit junction boxes located in the EDG building and identified an unsealed 4" electrical conduit penetration. Exelon's design basis external flood height is 132' and the unsealed penetration was at elevation 127' and communicated directly with the external flood water. This degraded condition could allow external flood water intrusion into the E-1 diesel bay. The degraded condition was entered into the CAP under IR 4164952 and the conduit was immediately filled with sealant material to restore the flood barrier. An extent of condition review was performed in each diesel bay and one 4" conduit in each bay was found unsealed. The degraded penetrations were immediately sealed. Exelon performed a cause evaluation and determined that the unsealed penetrations were a result of a modification in the 1990s that did not consider its impact on flood seals. Exelon's review did not identify any further extent of condition vulnerabilities related to this modification.
- The inspectors reviewed the degraded seal conditions, cause evaluation, and the immediate corrective actions. The inspectors validated that the sealant material applied was capable of withstanding the forces developed by the flood waters and would remain in-tact. In addition, the inspectors reviewed the licensee's original evaluation on the inaccessibility of the EDG room flood seals and determined that the seals were accessible and should have been inspected during the post-Fukushima walkdowns in 2012. Furthermore, it was identified that a total population of 108 inaccessible floods seals on site were incorrectly evaluated for accessibility and needed to be inspected. Exelon performed an expedited review of this population of seals and did not identify any required flood seals that were missing. The inspectors reviewed the extent of condition population and performed risk informed
- inspections of flood seal inspections. The inspectors did not identify any significant issues with the flood seal inspections that were performed.
- The inspectors reviewed the as-found unsealed penetration condition and the potential challenge to the operability and availability of the EDGs. The inspectors reviewed the site original design basis flood analysis along with the updated post-Fukushima flooding reanalysis to determine the impact on the EDGs. The inspectors determined that the sites original external flood design basis of 132' was conservative and the post-Fukushima flooding hazard reanalysis determined the actual stillwater flood height would remain below the penetration elevation. Exelon's external flood reanalysis was performed using analytical methods acceptable by the NRC and was qualified for use as an alternative analytical

method in support of an operability determination. The inspectors review determined that the reanalyzed flood height was below the height of any equipment that could impact the EDG operability or availability and it would remain operable despite the missing flood seals. Therefore, the inspectors did not identify any performance deficiencies more than minor.

Unit 2 Instrument Nitrogen Moisture Content

- The inspectors reviewed Exelon's corrective actions for an adverse trend in instrument nitrogen quality documented in IRs 04056044 and 04175504. Specifically, it was identified that the Unit 2 instrument nitrogen system repeatedly failed biennial testing acceptance criteria for moisture content. Upon each occurrence, corrective actions were taken to replace the desiccant and verify that the moisture content was left below the acceptance criteria. Exelon appropriately entered the identified trend into their CAP and developed actions to monitor and evaluate it. Upon evaluation, it was identified that the relevant industry standard, ANSI/ISA-7.0.01-1996, "*Quality Standard for Instrument Air*," does not specify moisture content as an element of instrument air quality for use in pneumatic instruments. Additionally, the performance history of instrumentation supplied by the instrument nitrogen system over the last 15 years was reviewed, and no evidence was discovered to suggest that the variable moisture content experienced during that time period contributed to adverse performance of the instrumentation. Therefore, Exelon determined that moisture content can be considered a best practice not required by the standard or station operating experience.
- Nonetheless, Exelon developed a tool for trending and potential incorporation into the instrument nitrogen system's Performance Monitoring Plan. Additionally, in light of potential extended operation under subsequent license renewal, Exelon planned to further evaluate the underlying issue of moisture in the instrument nitrogen system to determine if further corrective action, beyond replacing the desiccant when needed, is warranted. Extent of condition reviews have been performed and no similar trend has been observed on the other instrument nitrogen systems at the site. The inspectors walked down the system, observed its operation, and reviewed the industry standard and recent preventative maintenance test results. The inspectors determined that Exelon's completed and proposed actions were reasonable and no additional issues of concern were identified.

Semi-Annual Trend Review

- The inspectors evaluated a sample of issues and events that occurred over the course of the third and fourth quarters of 2018 to determine whether issues were appropriately considered as emerging or adverse trends. The inspectors verified that these issues were addressed within the scope of the CAP or through department review.
- Exelon identified an adverse trend in equipment reliability during the first two quarters of 2018 and the trend continued through the remainder of 2018. A relatively high number of equipment performance challenges had occurred at Peach Bottom associated with adjustable speed drives, E-3 diesel, Unit 3 RCIC, external flood seals, condensate pump cables, and MSIVs. An analysis of

common issues was performed to evaluate the cause of this adverse trend. Exelon identified that Peach Bottom has declined in the technical rigor applied to decision making which has directly impacted equipment performance issues. These results were documented in IR 4155200. The station developed performance improvement plans and focused briefings to site personnel to reinforce technical decision making standards. In addition, the station performed evaluations of risk significant equipment issues that are currently outstanding to confirm actions to mitigate and eliminate the issues. The inspectors reviewed the IRs and determined that Exelon had performed an adequate evaluation and the corrective actions were commensurate with the safety significance of the adverse trend. Furthermore, the station is performing a root cause evaluation in response to an NRC White finding (IR 4195110, NRC Inspection Report 05000277/2018003 and 05000278/2018003) that will result in additional corrective actions. Currently, the inspectors did not identify any issues of concern. However, additional inspection and assessment of the licensee's actions to address this trend will be reviewed in 2019.

- Generally, the station's implementation of the CAP has been effective in promptly identifying and correcting issues. In addition, the station is generally effective in identifying their own weaknesses and taking corrective actions to address the issues. Notwithstanding this, the inspectors have identified a recent trend in the effectiveness of the CAP in resolving equipment related issues in a timely and effective manner. The inspectors noted examples of conditions adverse to quality in the CAP not being addressed in a timely manner (containment atmosphere control/containment atmosphere dilution primary containment isolation valves, '2B' battery charger, E-3 EDG dashpot oil leak, spent fuel pool level indication). The station has recognized the adverse trend in CAP effectiveness and documented the concern in IR 4209875. The evaluation and corrective actions have not been completed and the residents will continue to monitor the licensee's performance closely in this area.
- No additional issues of concern were identified.

<u>Licensee Identified Non-Cited Violation Severity Level IV</u>

- This violation of very low safety significant was identified by the licensee and has been entered into the licensee's CAP and is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.
- Violation: Peach Bottom Atomic Power Station, Unit No. 2 Renewed License No. DPR-44, Condition 2.C.5.b.3 and Peach Bottom Atomic Power Station, Unit No. 3 Renewed License No. DPR-56, Condition 2.C.5.b.3 requires, in part, that no disbursements or payments from the [decommissioning] trust shall be made by the trustee until the trustee has first given the NRC 30 days' notice of the payment.
- Contrary to the above, on occasions between 2001 and 2015, disbursements from the Peach Bottom Atomic Power Station decommissioning trust were made by the trustee and the trustee had not first given the NRC 30 days' notice of the payment. Specifically, in 2001, 2012, and 2015, PSEG directed the Bank of New York Mellon (the trustee of the decommissioning trust for Peach Bottom Atomic Power Station) to disburse payments equaling $145,548.34 for Unit 2 and $145,548.34 for Unit 3 for decommissioning cost estimates. However, PSEG failed to notify the NRC of these disbursements until October 19, 2018.

- Significance/Severity: This issue is considered within the traditional enforcement process because the failure to inform the NRC prior to disbursing decommissioning funds impacts the ability of the NRC to perform its regulatory oversight function. As noted in Section 2.2.4 of the NRC Enforcement Policy, such violations are dispositioned using traditional enforcement.
- The inspectors evaluated the violation in accordance with the NRC Enforcement Policy and determined that it is appropriately characterized at Severity Level IV (SL IV) because it is similar to the SL IV example violation 6.9.d.7, describing a licensee's failure to provide or make a 15-day or 30-day written report or notification that does not impact the regulatory response by the NRC. For this Peach Bottom issue, the inspectors determined that the disbursements were made for acceptable decommissioning expenses and would not have necessitated further inquiry or caused the NRC to object to the payments.
- Corrective Action Reference: IR 4202344

February 26, 2019 – Letter dated February 26, 2019 from Jennifer Tobin, Project Manager Plant Licensing Branch I Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to Mr. Bryan C. Hanson Senior Vice President Exelon Generation Company, LLC President and Chief Nuclear Officer Exelon Nuclear with subject of PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 - ISSUANCE OF AMENDMENT NOS. 323 AND 326 TO REVISE TECHNICAL SPECIFICATIONS TO ALLOW CONTINUED OPERATION WITH TWO SAFETY RELIEF VALVES/SAFETY VALVES OUT OF SERVICE (EPID L-2018-LLA-0151)

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment Nos. 323 and 326 to Renewed Facility Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station (Peach Bottom), Units 2 and 3, respectively. These amendments are in response to your application dated May 30, 2018, as supplemented by letter dated December 6, 2018 (Agencywide Documents Access and Management System Accession Nos. ML18150A387 and ML18340A185, respectively).

The amendments revise the Peach Bottom, Unit 2 and 3, Technical Specifications to allow continued operation with two safety relief valves/safety valves out of service and to increase the reactor coolant system pressure safety limit. Specifically, the amendments revise Technical Specification Safety Limit 2.1.2 and Limiting Condition for Operation 3.4.3 for both Units 2 and 3.

A copy of the related safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

EXELON GENERATION COMPANY, LLC
PSEG NUCLEAR LLC
DOCKET NO. 50-277
PEACH BOTTOM ATOMIC POWER STATION, UNIT 2 AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 323 Renewed License No. DPR-44

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

1. The application for amendment by Exelon Generation Company, LLC (Exelon Generation Company) and PSEG Nuclear LLC (the licensees), dated May 30, 2018, as supplemented by letter dated December 6, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
2. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
3. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
4. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
5. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Renewed Facility Operating License No. DPR-44 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 323, are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

This license amendment is effective as of the date of its issuance and shall be implemented within 60 days.

CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

March 4, 2019 – Letter dated March 4, 2019 from Daniel S. Collins, Director Division of Reactor Projects to Bryan C. Hanson Senior Vice President, Exelon Generation Company, LLC President and Chief Nuclear Officer, Exelon Nuclear with a subject of ANNUAL ASSESSMENT LETTER FOR PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 (REPORTS 05000277/2018006 AND 05000278/2018006)

The U.S. Nuclear Regulatory Commission (NRC) has completed its end-of-cycle performance assessment of Peach Bottom Atomic Power Station, Units 2 and 3, including review of performance indicators, inspection results, and enforcement actions from January 1, 2018, through December 31, 2018. This letter informs you of the NRC’s assessment of your facility during this period and its plans for future inspections at your facility. The NRC concluded that overall performance at your facility preserved public health and safety.

The NRC determined the performance at Peach Bottom Atomic Power Station, Units 2 and 3 during the most recent quarter was within the Regulatory Response Column (Column 2) of the NRC's Reactor Oversight Process (ROP) Action Matrix in Inspection Manual Chapter 0305, "Operating Assessment Program." This conclusion was based on one finding of low-to- moderate safety significance (White) associated with inadequate corrective actions which resulted in the failure of the E-3 emergency diesel generator shared between both units. Units 2 and 3 entered Column 2 as of the third quarter of 2018. The Notice of Violation was issued on December 11, 2018 (ML18341A206[1]).

Therefore, in addition to ROP baseline inspections, the NRC plans to conduct a supplemental inspection in accordance with Inspection Procedure 95001, "Supplemental Inspection for One or Two White Inputs in a Strategic Performance Area." The objectives of this inspection are to assure that the root and contributing causes of degraded performance are understood; to independently assess and assure that the extents of condition and cause are identified; and to assure that appropriate corrective actions are taken to prevent recurrence in a prompt manner. This inspection will be scheduled after you notify the NRC of your readiness.

The enclosed inspection plan lists the inspections scheduled through December 31, 2020. This updated inspection plan now includes planned security inspections which were formerly transmitted under separate correspondence. The NRC provides the inspection plan to allow for the resolution of any scheduling conflicts and personnel availability issues. Routine inspections performed by resident inspectors are not included in the inspection plan. You should be aware that the agency is pursuing potential changes to the ROP, including changes to engineering inspections (SECY-18-0113, "Recommendations for Modifying the Reactor Oversight Process Engineering Inspections"). Should these changes to the ROP be implemented, the engineering and other region-based inspections are subject to change in scope, as well as schedule, beginning in January 2020. Furthermore, all the inspections listed during the last twelve months of the inspection plan are tentative and may be revised. The NRC will contact you as soon as possible to discuss changes to the inspection plan should circumstances warrant any changes.

In addition to baseline inspections, the NRC will also conduct Inspection Procedure 81311, "Physical Security Requirements for Independent Spent Fuel Storage Installations," in February 2020.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390 of the NRC's "Rules of Practice," a copy of this letter will be available electronically for public inspection in the NRC's Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC's Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

IP 22 Inspection Activity Plan Report

Unit	Start	End	Activity	CAC	Title	Staff Count
EQ PROGRAM 4						
2, 3	01/14/2019	01/18/2019	IP 71111.21N	000714	Design Bases Assurance Inspection (Programs)	
2, 3	01/28/2019	02/01/2019	IP 71111.21N	000714	Design Bases Assurance Inspection (Programs)	
PEACH BOTTOM INITIAL OL EXAM 5						
2, 3	01/27/2019	02/01/2019	OV	000956	VALIDATION OF INITIAL LICENSE EXAMINATION (OV)	
2, 3	02/24/2019	03/08/2019	EXAD	000500	LICENSE EXAM ADMINISTRATION (EXAD)	
Annual Sample - HPCI Steam Sensing Line Leak 1						
3	02/24/2019	03/02/2019	IP 71152	000748	Problem Identification and Resolution	
EP PROGRAM INSPECTION 2						
2, 3	03/04/2019	03/08/2019	IP 71114.02	000717	Alert and Notification System Testing	
2, 3	03/04/2019	03/08/2019	IP 71114.03	000718	Emergency Response Organization Staffing and Augmentation System	
2, 3	03/04/2019	03/08/2019	IP 71114.05	000720	Maintenance of Emergency Preparedness	
2, 3	03/04/2019	03/08/2019	IP 71151	001397	Performance Indicator Verification	
PB REQUAL INSP WITH P/F RESULTS 2						
2, 3	03/10/2019	03/15/2019	IP 71111.11A	000703	Licensed Operator Requalification Program and Licensed Operator Performance (Annual)	
2, 3	03/10/2019	03/15/2019	IP 71111.11B	000704	Licensed Operator Requalification Program and Licensed Operator Performance (Biennial)	
ACCESS CONTROL, PROTECTIVE STRATEGY, TSR 4						
2, 3	03/25/2019	03/29/2019	IP 71130.02	000734	Access Control	
2, 3	03/25/2019	03/29/2019	IP 71130.05	000737	Protective Strategy Evaluation	
2, 3	03/25/2019	03/29/2019	IP 71130.14	000743	Review of Power Reactor Target Sets	
2, 3	03/25/2019	03/29/2019	IP 71151	001338	Performance Indicator Verification	
CYBER FULL IMP 4						
2, 3	04/01/2019	04/05/2019	IP 71130.10P	000741	Cyber Security	
2, 3	04/15/2019	04/19/2019	IP 71130.10P	000741	Cyber Security	
RAD EFFLUENTS Mod 06 1						
2, 3	04/15/2019	04/19/2019	IP 71124.06	000730	Radioactive Gaseous and Liquid Effluent Treatment	

This report does not include INPO and OUTAGE activities.
This report shows only on-site and announced inspection procedures.

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Enclosure

Peach Bottom

01/01/2019 - 12/31/2020

IP 22 Inspection Activity Plan Report

Unit	Start	End	Activity	CAC	Title	Staff Count
Triennial Heat Sink 1						
2, 3 05/13/2019 05/17/2019 IP 71111.07T 000700 Heat Sink Performance -Triennial						
RAD HAZARD & ALARA Mod 01 1						
2, 3 05/13/2019 05/17/2019 IP 71124.01 000725 Radiological Hazard Assessment and Exposure Controls						
PI&R BIENNIAL 4						
2, 3 06/10/2019 06/14/2019 IP 71152B 000747 Problem Identification and Resolution						
2, 3 06/24/2019 06/28/2019 IP 71152B 000747 Problem Identification and Resolution						
Radwaste 1						
2, 3 07/29/2019 08/02/2019 IP 71124.08 000732 Radioactive Solid Waste Processing and Radioactive Material Handling, Storage, and Transportation						
Material Control and Accountability 1						
2, 3 08/11/2019 08/17/2019 IP 71130.11 000742 Material Control and Accounting (MC&A)						
50.59 Process 3						
2, 3 09/22/2019 09/28/2019 IP 71111.17T 000709 Evaluations of Changes, Tests, and Experiments						
INSERVICE INSPECTION - UNIT 3 1						
3 10/27/2019 11/02/2019 IP 71111.08G 000701 Inservice Inspection Activities (BWR)						
RAD HAZARD, ALARA, AIR ACT CNTRL Mod 01 1						
2, 3 10/28/2019 11/01/2019 IP 71124.01 000725 Radiological Hazard Assessment and Exposure Controls						
RAD SAFETY Mods 01, PI (OR01, PR01) 1						
2, 3 12/02/2019 12/06/2019 IP 71124.01 000725 Radiological Hazard Assessment and Exposure Controls						
2, 3 12/02/2019 12/06/2019 IP 71151 000746 Performance Indicator Verification						
Access Authorization and Fitness for Duty 2						
2, 3 01/06/2020 01/10/2020 IP 71130.01 000733 Access Authorization						
2, 3 01/06/2020 01/10/2020 IP 71130.08 000740 Fitness For Duty Program						
2, 3 02/03/2020 02/07/2020 IP 71130.01 000733 Access Authorization						
2, 3 02/03/2020 02/07/2020 IP 71130.02 000734 Access Control						
2, 3 02/03/2020 02/07/2020 IP 71130.08 000740 Fitness For Duty Program						
2, 3 02/03/2020 02/07/2020 IP 71151 001338 Performance Indicator Verification						

This report does not include INPO and OUTAGE activities.
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Peach Bottom

01/01/2019 - 12/31/2020

IP 22 Inspection Activity Plan Report

Unit	Start	End	Activity	CAC	Title	Staff Count
Access Authorization and Fitness for Duty 2						
2, 3 02/03/2020 02/07/2020 IP 81311 000831 Physical Security Requirements for Independent Spent Fuel Storage Installations						

RAD MONITORING INSTRUMENT Mod 05 1						
2, 3	03/23/2020	03/27/2020	IP 71124.05	000729	Radiation Monitoring Instrumentation	
FY2020 Peach Bottom Initial Exam 3						
2, 3	04/19/2020	04/24/2020	OV	000956	VALIDATION OF INITIAL LICENSE EXAMINATION (OV)	
2, 3	05/24/2020	05/29/2020	EXAD	000500	LICENSE EXAM ADMINISTRATION (EXAD)	
PEACH BOTTOM EP EXERCISE INSPECTION 5						
2, 3	04/20/2020	04/24/2020	IP 71114.01	000716	Exercise Evaluation	
2, 3	04/20/2020	04/24/2020	IP 71151	001397	Performance Indicator Verification	
AIR ACTIVITY CNTRL, OCC DOSE ASSESS; Mod 03, 04 1						
2, 3	04/27/2020	05/01/2020	IP 71124.03	000727	In-Plant Airborne Radioactivity Control and Mitigation	
2, 3	04/27/2020	05/01/2020	IP 71124.04	000728	Occupational Dose Assessment	
Design Basis Assurance Inspection - Team 4						
2, 3	07/13/2020	07/19/2020	IP 71111.21M	000713	Design Bases Assurance Inspection (Teams)	
2, 3	07/27/2020	08/02/2020	IP 71111.21M	000713	Design Bases Assurance Inspection (Teams)	
FORCE-ON-FORCE PLANNING AND EXERCISE WEEKS 6						
2, 3	07/27/2020	07/31/2020	IP 71130.03	000735	Contingency Response - Force-On-Force Testing	
2, 3	08/17/2020	08/21/2020	IP 71130.03	000735	Contingency Response - Force-On-Force Testing	
REMP Mod 07 1						
2, 3	09/21/2020	09/25/2020	IP 71124.07	000731	Radiological Environmental Monitoring Program	
ISI - UNIT 2 1						
2	10/26/2020	10/30/2020	IP 71111.08G	000701	Inservice Inspection Activities (BWR)	
RAD HAZARD, ALARA, AIR ACT CNTRL Mods 01, 02, 03 1						
2, 3	10/26/2020	10/30/2020	IP 71124.01	000725	Radiological Hazard Assessment and Exposure Controls	
2, 3	10/26/2020	10/30/2020	IP 71124.02	000726	Occupational ALARA Planning and Controls	
2, 3	10/26/2020	10/30/2020	IP 71124.03	000727	In-Plant Airborne Radioactivity Control and Mitigation	

This report does not include INPO and OUTAGE activities.
This report shows only on-site and announced inspection procedures.

Peach Bottom

01/01/2019 - 12/31/2020

IP 22 Inspection Activity Plan Report

Unit	Start	End	Activity	CAC	Title	Staff Count
RAD HAZARD, ALARA, AIR ACT CNTRL Mods 01, 02, 03, PI 1						
2, 3	12/07/2020	12/11/2020	IP 71124.01	000725	Radiological Hazard Assessment and Exposure Controls	
2, 3	12/07/2020	12/11/2020	IP 71124.02	000726	Occupational ALARA Planning and Controls	
2, 3	12/07/2020	12/11/2020	IP 71124.03	000727	In-Plant Airborne Radioactivity Control and Mitigation	
2, 3	12/07/2020	12/11/2020	IP 71151	000746	Performance Indicator Verification	

This report does not include INPO and OUTAGE activities.
This report shows only on-site and announced inspection procedures.

March 11, 2019 – Letter dated March 11, 2019 from Brett Titus, Acting Chief Beyond-Design-Basis Engineering Branch Division of Licensing Projects Office of Nuclear Reactor Regulation to Bryan C. Hanson Senior Vice President Exelon Generation Company, LLC President and Chief Nuclear Officer Exelon Nuclear with a subject of

PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3- SAFETY EVALUATION REGARDING IMPLEMENTATION OF HARDENED CONTAINMENT VENTS CAPABLE OF OPERATION UNDER SEVERE **ACCIDENT CONDITIONS RELATED TO ORDER EA-13-109 (CAC NOS.** MF4416 AND MF4417; EPID NO. L-2014-JLD-0053)

On June 6, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13143A334), the U.S. Nuclear Regulatory Commission **(NRC)** issued Order EA-13-109, "Order to Modify Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions," to all Boiling Water Reactor licensees with Mark I and Mark II primary containments. The order requirements are provided in Attachment 2 to the order and are divided into two parts to allow for a phased approach to implementation. The order required each licensee to submit an Overall Integrated Plan (OIP) for review that describes how compliance with the requirements for both phases of Order EA- 13-109 would be achieved.

By letter dated June 30, 2014 (ADAMS Accession No. ML14181A301), Exelon Generation Company, LLC (the licensee) submitted its Phase 1 OIP for Peach Bottom Atomic Power Station, Units 2 and 3 (Peach Bottom) in response to Order EA-13-109. At 6-month intervals following the submittal of the Phase 1 OIP, the licensee submitted status reports on its progress in complying with Order EA-13-109 at Peach Bottom, including the combined Phase 1 and Phase 2 OIP in its letter dated December 15, 2015 (ADAMS Accession No. ML15364A015). These status reports were required by the order, and are listed in the enclosed safety evaluation. By letters dated May 27, 2014 (ADAMS Accession No. ML14126A545), and August 10, 2017 (ADAMS Accession No. ML17220A328), the NRC notified all Boiling Water Reactor Mark I and Mark II licensees that the staff will be conducting audits of their implementation of Order EA-13-109 in accordance with NRC Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, "Regulatory Audits" (ADAMS Accession No. ML082900195). By letters dated February 12, 2015 (Phase 1) (ADAMS Accession No. ML15026A469), August 2, 2016 (Phase 2) (ADAMS Accession No. ML16099A272), and November 30, 2017 (ADAMS Accession No. ML17328A163), the NRC issued Interim Staff Evaluations and an audit report, respectively, on the licensee's progress. By letter dated September 28, 2018 (ADAMS Accession No. ML18271A008), the licensee reported that Peach Bottom is in full compliance with the requirements of Order EA-13-109, and submitted a Final Integrated Plan for Peach Bottom.

The enclosed safety evaluation provides the results of the NRC staffs review of Peach Bottom's hardened containment vent design and water management strategy for Peach

Bottom. The intent of the safety evaluation is to inform Peach Bottom on whether or not its integrated plans, if implemented as described, appear to adequately address the requirements of Order EA-13-109. The staff will evaluate implementation of the plans through inspection, using Temporary Instruction 2515-193, "Inspection of the Implementation of EA-13-109: Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions" (ADAMS Accession No. ML17249A105). This inspection will be conducted in accordance with the NRC's inspection schedule for the plant.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO ORDER EA-13-109
EXELON GENERATION COMPANY, LLC
PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 DOCKET NOS. 50-277 AND 50-278

INTRODUCTION

The earthquake and tsunami at the Fukushima Dai-ichi nuclear power plant in March 2011 highlighted the possibility that extreme natural phenomena could challenge the prevention, mitigation and emergency preparedness defense-in-depth layers already in place in nuclear power plants in the United States. At Fukushima, limitations in time and unpredictable conditions associated with the accident significantly challenged attempts by the responders to preclude core damage and containment failure. During the events at Fukushima, the challenges faced by the operators were beyond any faced previously at a commercial nuclear reactor and beyond the anticipated design basis of the plants. The U.S. Nuclear Regulatory Commission (NRC) determined that additional requirements needed to be imposed at U.S. commercial power reactors to mitigate such beyond-design-basis external events (BDBEEs) during applicable severe accident conditions.

On June 6, 2013 [Reference 1], the NRC issued Order EA-13-109, "Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation under Severe Accident Conditions". This order requires licensees to implement its requirements in two phases. In Phase 1, licensees of boiling-water reactors (BWRs) with Mark I and Mark II containments shall design and install a venting system that provides venting capability from the wetwell during severe accident conditions. In Phase 2, licensees of BWRs with Mark I and Mark II containments shall design and install a venting system that provides venting capability from the drywall under severe accident conditions, or, alternatively, those licensees shall develop and implement a reliable containment venting strategy that makes it unlikely that a licensee would need to vent from the containment drywall during severe accident conditions.

By letter dated June 30, 2014 [Reference 2], Exelon Generation Company, LLC (the licensee) submitted a Phase 1 Overall Integrated Plan (OIP) for Peach Bottom Atomic Power Station, Units 2 and 3 (PBAPS, Peach Bottom) in response to Order EA-13-109. By letters dated December 19, 2014 [Reference 3], June 30, 2015 [Reference 4], December 15, 2015 (which included the combined Phase 1 and Phase 2 OIP) [Reference 5], June 30, 2016 [Reference 6], December 15, 2016 [Reference 7], June 30, 2017 [Reference 8], December 15, 2017 [Reference 9], and June 29, 2018 [Reference 10], the licensee submitted 6-month updates to its OIP. By letters dated May 27, 2014 [Reference 11], and August 10, 2017 [Reference 12], the NRC notified all BWR Mark I and Mark II licensees that the staff will be conducting audits of their implementation of Order EA-13-109 in accordance with NRC Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, "Regulatory Audits" [Reference 13]. By letters dated February 12, 2015 {Phase 1) [Reference 14], August 2, 2016 (Phase 2) [Reference 15], and November 30, 2017 [Reference 16], the NRC issued Interim Staff Evaluations (ISEs) and an audit report, respectively, on the licensee's progress. By letter dated September 28, 2018 [Reference 17], the licensee reported that full compliance with the requirements of Order EA-13- 109 was achieved and submitted its Final Integrated Plan (FIP).

CONCLUSION

In June 2014, the NRC staff started audits of the licensee's progress in complying with Order EA-13-109. The staff issued an ISE for implementation of Phase 1 requirements on February 12, 2015 [Reference 14], an ISE for implementation of Phase 2 requirements on August 2, 2016 [Reference 15], and an audit report on the licensee's responses to the ISE open items on November 30, 2017 [Reference 16]. The licensee reached its final compliance date on September 28, 2018, and has declared in letter dated September 28, 2018 [Reference 17], that Peach Bottom Atomic Power Station, Units 2 and 3, is in compliance with the order.

Based on the evaluations above, the NRC staff concludes that the licensee has developed guidance that includes the safe operation of the HCVS design and a water management strategy that, if implemented appropriately, should adequately address the requirements of Order EA-13-109.

March 14, 2019 – letter dated March 14, 2019 from Glenn T. Dentel, Chief Engineering Branch 2 to Bryan Hanson Senior Vice President, Exelon Generation Company, LLC President and Chief Nuclear Officer, Exelon Nuclear with a subject of PEACH BOTTOM ATOMIC POWER STATION UNITS 2 AND 3 – DESIGN BASES ASSURANCE INSPECTION (PROGRAMS) REPORT 05000277/2019011 AND 05000278/2019011

On February 1, 2019, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Peach Bottom Atomic Power Station Units 2 and 3 and discussed the results of this inspection with Mr. Pat Navin, Peach Bottom Site Vice President and other members of your staff. The results of this inspection are documented in the enclosed report.

NRC inspectors documented one finding of very low safety significance (Green) in this report. This finding involved a violation of NRC requirements. The inspectors also documented a licensee-identified violation which was determined to be of very low safety significance in this report. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2.a of the Enforcement Policy.

If you contest the violation or significance or severity of the violation documented in this inspection report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory

Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement; and the NRC resident inspector at Peach Bottom.

If you disagree with a cross-cutting aspect assignment in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; and the NRC resident inspector at Peach Bottom.

Inspection Report Summary

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring Exelon's performance by conducting a Design Bases Assurance Inspection of the Environmental Qualification Program implementation at Peach Bottom Units 2 and 3 in accordance with the Reactor Oversight Process. The Reactor Oversight Process is the NRC's program for overseeing the safe operation of commercial nuclear power reactors. Refer to https://www.nrc.gov/reactors/operating/oversight.html for more information. Findings and violations being considered in the NRC's assessment are summarized in the table below. Licensee-identified non-cited violations are documented in report sections: 71111.21N.

List of Findings and Violations

Drywell Local Temperature Exceeds Analyzed Environmental Qualification (EQ) Value, Shortening Qualified Life for Several EQ Components			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000278,05000277/2019011-01 Open/Closed	[H.7] - Documentation	71111.21N
The inspectors identified a finding of very low safety significance (Green) and associated non-cited violation (NCV) of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.49(j) and 10 CFR Part 50.49(d), Environmental Qualification (EQ) of Electric Equipment Important to Safety for Nuclear Power Plants, because Exelon did not ensure EQ files included accurate and bounding normal service temperature values for EQ components located in drywell Zone 2. Therefore, the supporting analysis, including evaluation of equipment thermal aging, was inaccurate and did not verify EQ components located in drywell Zone 2 were qualified for the normal service temperature at the location where the equipment must perform their specified performance requirements up to the end of their qualified life.			

Inspection results

Drywell Local Temperature Exceeds Analyzed Environmental Qualification (EQ) Value, Shortening Qualified Life for Several EQ Components

Cornerstone	Significance	Cross-cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000278,05000277/2019011-01 Open/Closed	[H.7] - Documentation	71111.21N

The inspectors identified a finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50.49(j) and 10 CFR Part 50.49(d), Environmental Qualification (EQ) of Electric Equipment Important to Safety for Nuclear Power Plants, because Exelon did not ensure EQ files included accurate and bounding normal service temperature values for EQ components located in drywell Zone 2. Consequently, the supporting analysis, including evaluation of equipment thermal aging, was inaccurate and did not verify EQ components located in drywell Zone 2 were qualified for the normal service temperature at the location where the equipment must perform their specified performance requirements up to the end of their qualified life.

Description: Normal service temperature is a factor in thermal aging and life qualification of components within the environmental qualification process. Exelon established a normal service temperature of 150F for all main steam line safety relief valves (SRVs) and 145F for all remaining EQ components inside the Unit 2 and Unit 3 primary containment/drywells.

On January 15, 2019, engineers informed the inspectors that drywell bulk average temperature was used to verify the normal service temperature of all EQ components located inside the drywell. The drywell bulk average temperature is calculated monthly per RT-O-40C-530-2(3), “Drywell Temperature Monitoring,” to verify satisfactory operating temperature conditions as described in Technical Specifications section 3.6.1.4, “Drywell Air Temperature.” This specification requires the drywell average temperature shall be < 145 degrees Fahrenheit (F). The drywell bulk average temperature is obtained from 17 separate thermowells and associated temperature elements located at five different Zones inside the drywell. The inspectors noted Exelon had not considered potential effects of possible localized hot spots/areas within the drywell.

The inspectors requested additional data to determine whether there were EQ components within drywell zones where actual temperatures were above the qualified service life temperatures (150F for SRVs, 145F for all other EQ components). To address the inspectors’ concerns, engineers reviewed applicable drywell temperature data for the last

4 years and determined that actual ambient temperature for portions of drywell Zone 2 consistently exceeded 145F (Unit 2 4-year average was 155.6F, Unit 3 was 153.7F). The inspectors independently verified the highest actual Unit 2 drywell Zone 2 temperature element was 166F and increasing, and Unit 3 was 156F and increasing at the conclusion of this inspection. Environmentally qualified components located in drywell Zone 2

included: 11 SRV solenoid pilot valves located on the main steam lines within the drywell which provide a safety function to prevent nuclear system over-pressurization and to depressurize the system to support core cooling; reactor water cleanup inboard containment isolation valve (MO-2(3)-12-15); and the main steam inboard (AO-2(3)-02-316) and recirculation inboard (AO-2(3)-02-039) sample valves which provide a containment isolation safety function. The associated limiting sub-component and impact on qualified life is

described below:

- All 11 SRVs on each unit were affected. The limiting sub-component was a viton gasket on the AVCO pilot solenoid valve. Qualified life was reduced from 18.9 years to 7.1 years.
- The limiting sub-component on MO-2-12-15 was the Class RH insulation on the limitorque motor operator. Qualified life was reduced from 60 years to 33.6 years.
- The limiting sub-component on MO-3-12-15 was the nordel o-ring on the EGS quick disconnect connector on the limitorque motor operator. Qualified life was reduced from 14.8 years to 8.5 years.
- The limiting sub-component on AO-2-02-039 was the EPDM o-rings on the NAMCO EA740 series position limit switch. Qualified life was reduced from 23 years to 11.8 years.
- The limiting sub-component on AO-3-02-039 and AO-3-02-316 was the viton elastomer seat on the ASCO model NP8300142ERF solenoid valve. Qualified life was reduced from 23 years to 10.2 years.
- The limiting sub-component on AO-2-02-316 was the viton elastomer seat on the ASCO model NP8300142ERF solenoid valve. Qualified life was reduced from 23 years to 7.8 years.

The inspectors reviewed maintenance records and determined Exelon has replaced all SRV solenoid pilot valves on a 6-year periodicity to align this work activity with the 6-year American Society of Mechanical Engineers (ASME) Code requirement for periodic SRV pressure testing. Therefore, the SRVs were replaced more frequently than required by the revised EQ analysis (7.1 year qualified life) and remained qualified.

The inspectors also noted that Exelon had not performed required reviews of station ambient temperature data for all EQ zones inside the drywell. Procedure CC-MA-203-1001, section 3.4, requires the station EQ engineer to perform annual reviews of station ambient temperature conditions and revise qualification data to incorporate changing ambient temperature conditions as required. The EQ engineer will document this annual review in an engineering technical evaluation. However, the inspectors identified that although EQ zones temperatures within the reactor building were verified quarterly, no procedure existed to perform the required verification of EQ zones within the primary containment/drywell. Exelon's assumption that drywell bulk average temperature bounded the highest normal service temperature for all EQ components in the drywell was incorrect. As a result, drywell Zone 2 temperature exceeded the analyzed normal temperature of 145F (150F for SRVs), resulting in a shorter qualified life for several EQ components as stated above.

Corrective Actions: Exelon staff entered the issue into their corrective action program and performed a technical evaluation to determine a more accurate average ambient temperature for drywell Zone 2 and to requalify the affected components. Exelon determined none of the components were currently beyond their revised qualified life and all remained qualified. The inspectors reviewed the evaluations and determined they were technically

sound. Additionally, Exelon initiated action to assess the programmatic impact of this issue, develop procedure revisions to properly monitor the local temperature of all EQ zones, and schedule drywell Zone 2 EQ component replacement activities consistent with their respective revised analyzed qualified life.

Corrective Action References: Issue Reports 04211923 and 04212231

Performance Assessment:

Performance Deficiency:
Exelon did not ensure EQ files included accurate and bounding normal service temperature values for EQ components located in drywell zone 2 as required by 10 CFR 50.49, "Environmental Qualification." Consequently, the supporting analysis, including equipment thermal aging, was inaccurate and did not verify the EQ components located in drywell
zone 2 were qualified for the normal service temperature at the location where the equipment must perform their specified performance requirements up to the end of their qualified life.

Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the Design Control attribute of the Mitigating Systems cornerstone. The objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage) was adversely impacted. Specifically, failure to verify and analyze actual drywell Zone 2 normal service temperature, resulted in several EQ components having significantly shorter qualified lives than previously analyzed and supported by the existing EQ preventive maintenance replacement schedule.

Significance: The inspectors assessed the significance of the finding using Appendix A, "Significance Determination of Reactor Inspection Findings for At - Power Situations." The performance deficiency affected the qualification of Mitigating Systems cornerstone components. Because these components maintained their functionality, the deficiency screened to green; very low safety significance.

Cross-cutting Aspect: H.7 - Documentation: The organization creates and maintains complete, accurate and up-to-date documentation. The finding had a cross-cutting aspect in the area of Human Performance, Documentation, because Exelon did not create and maintain complete and accurate procedures for verifying drywell EQ zones' normal service temperatures. Procedure CC-MA-203-1001, required the station EQ engineer to perform annual reviews of station ambient temperature conditions and revise qualification data to incorporate changing ambient temperature conditions as required. However, although
EQ zones temperatures within the reactor building were verified quarterly (per procedure

RT-O-40C-530-2(3), no procedure existed to perform the required normal service temperature verification of zones within the primary containment/drywell, the radwaste building, and the turbine building. [H.7]
Enforcement: Violation: Title 10 CFR Part 50.49(d), Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants, states in part, the licensee shall prepare a qualification file which shall include the environmental conditions at the location where the equipment must perform as specified. Additionally, 10 CFR Part 50.49(j), states in part, the record of qualification must permit verification that the component is qualified for its application and meets its specified performance requirements up to the end of its qualified life. Contrary to the above, since February 22, 1983, environmental qualification files for components inside Zone 2 of the primary containment/drywell, EQ-PB-019A (AVCO Pilot Solenoid Valve for all MSL Safety Relief Valves), EQ-PB-42B (NAMCO Limit Switch EA740 Series), and EQ-PB-46A (Limitorque Valve Operators with AC Motors Class RH Insulation), did not include the correct environmental conditions (temperature) at the location where the equipment must perform as specified. Therefore the record of qualification, including analysis of equipment thermal aging, was inaccurate and did not verify the EQ components located in

drywell Zone 2 were qualified for their application and would meet specified performance requirements up to the end of their qualified life.

Enforcement Action: This violation is being treated as a Non-Cited Violation, consistent with Section 2.3.2 of the Enforcement Policy.

Licensee-Identified Non-Cited Violation	71111.21N
This violation of very low safety significance was identified by the licensee and has been entered into the licensee corrective action program and is being treated as a Non-Cited Violation, consistent with Section 2.3.2 of the Enforcement Policy.	
Violation: Title 10 CFR Part 50.49(e)(5), Environmental Qualification (EQ) of Electric Equipment Important to Safety for Nuclear power Plants, requires in part, that equipment must be replaced or refurbished at the end of this designated life unless ongoing qualification demonstrates that the item has additional life. Contrary to the above, since June 2008, Exelon did not replace equipment or demonstrate additional qualification prior to the end of designated qualified life. Specifically, in 2015, Exelon identified that eight reactor pressure high scram relays and	

two Rosemount high pressure trip units exceeded their designated life without prior evaluation demonstrating additional qualified life. In 2017 and 2019, Exelon identified additional EQ components that exceeded their qualified life prior to their required replacement including a drywell temperature element, four pressure switch relays, eight drywell torus connectors, and four high pressure service water cross-tie transfer switches.

Significance: Green.
The inspectors determined the performance deficiency was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the reliability and capability of systems that response to initiating events to prevent undesirable consequences (i.e., core damage).

The inspectors assessed the significance of the finding using IMC 0609.04, “Initial Characterization of Findings,” and IMC 0609, Appendix A, Exhibit 2, “Mitigating Systems Screening Questions.” The inspectors determined that this finding was a deficiency affecting the design or qualification of mitigating structures, systems, or components, where the structures, systems, or components maintained its operability or functionality. Therefore, the inspectors determined the finding to be of very low safety significance (Green). Specifically, for 26 of the 27 abovementioned components, operability and qualification was subsequently demonstrated through a technical evaluation which extended the EQ life. Additionally, the remaining component (TE-3105-36A) provided backup indication of drywell temperature. The primary drywell temperature indications remained unaffected. The inspectors determined the loss of environmental qualification for TE-2105-36A did not adversely affect operators’ ability to assess or mitigate consequences of an accident.

Corrective Action References: Issue Reports 2480628, 2538737, 4005664, 4017436, 4026616, and 4179677

June 6, 2019 – Letter dated June 6, 2019 from Bennett Brady, Senior Project Manager License Renewal Projects Branch Division of Materials and License Renewal Office of Nuclear Reactor Regulation to Mr. Michael Gallagher Vice President, License Renewal and Decommissioning Exelon Generation Company with a subject of PEACH BOTTOM ATOMIC POWER STATION UNITS 2 AND 3 – REVISED REPORT FOR THE OPERATING EXPERIENCE REVIEW AUDIT REGARDING THE SUBSEQUENT LICENSE RENEWAL APPLICATION REVIEW (EPID NO. L-2018-RNW-0012)

By letter dated July 10, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18193A689), the Exelon Generation Company, LLC, (Exelon) submitted to the U.S. Nuclear Regulatory Commission (NRC or staff) an application to renew the Renewed Facility Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station, Units 2 and 3 (Peach Bottom), respectively. Exelon submitted the application pursuant to Title 10 of the Code of Federal Regulations Part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants,” for subsequent license renewal. The NRC staff completed its operating experience review audit at the Excel Services Corporation offices in Rockville, Maryland, from September 17 through September 27, 2018, in accordance with the operating experience review audit plan (ADAMS Accession No. ML18249A280). The audit report is enclosed.

Audit Introduction

The U.S. Nuclear Regulatory Commission (NRC or the staff) conducted an audit of Exelon Generation Company, LLC (Exelon) Peach Bottom Atomic Power Station (PBAPS) Units 2 and 3 (PB or the applicant's) plant-specific operating experience (OpE), as part of the staff's review of the Peach Bottom subsequent license renewal application (SLRA) at the EXCEL Services Corporation located in Rockville, Maryland, from September 17 through 27, 2018. The purpose of the audit was for the NRC staff to perform an independent review of plant specific OpE to identify examples of age-related degradation, as documented in the applicant's corrective action program database. The regulatory bases for the audit was Title 10 of the Code of Federal Regulations, Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," (10 CFR Part 54). The staff also considered the guidance contained in NUREG-2192, Rev. 0, "Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants" (SRP-SLR), dated July 2017, and NUREG-2191, Rev. 0, "Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report," dated July 2017.

The identified OpE examples will be further evaluated during the staff's subsequent technical review and auditing of aging management programs (AMPs), time-limited aging analyses (TLAAs) and aging management review (AMR) line items. The staff's identification and evaluation of pertinent OpE and additional related documentation, provides a basis for the staff's conclusions on the ability of the applicant's proposed AMPs and TLAAs to manage the effects of aging in the subsequent period of extended operation.

May 24, 2019 – Letter dated May 24, 2019 from Blake A. Purnell, Project Manager Plant Licensing Branch Ill Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to Bryan C. Hanson Senior Vice President Exelon Generation Company, LLC President and Chief Nuclear Officer Exelon Nuclear with a subject of LIMERICK GENERATING STATION, UNITS 1 AND 2, AND PEACH BOTTOM ATOMIC POWER STATION, UNITS 1, 2, AND 3 - ISSUANCE OF AMENDMENTS TO REVISE THE EMERGENCY RESPONSE ORGANIZATION STAFFING REQUIREMENTS (EPID L-2018-LLA-0150)

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the following enclosed amendments in response to the Exelon Generation Company, LLC application dated May 10, 2018 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML18149A290), as supplemented by letters dated November 1, 2018, and November 29, 2018 (ADAMS Accession Nos. ML183056270 and ML18337A004, respectively):

1. Amendment No. 235 to Renewed Facility Operating License No. NPF-39 and Amendment No. 198 to Renewed Facility Operating License No. NPF-85 for the Limerick Generating Station, Units 1 and 2, respectively; and

2. Amendment No. 14 to Facility Operating License No. DPR-12, Amendment No. 325 to Renewed Facility Operating License No. DPR-44, and Amendment No. 328 to Renewed Facility Operating License No. DPR-56 for the Peach Bottom Atomic Power Station, Units 1, 2, and 3, respectively.

The amendments revise the emergency plans by changing the emergency response organization staffing requirements for each of these facilities.

A copy of the NRC staff's Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

EXELON GENERATION COMPANY, LLC
PSEG NUCLEAR LLC
DOCKET NO. 50-171
PEACH BOTTOM ATOMIC POWER STATION, UNIT 1 AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 14 License No. DPR-12

1. The U.S. Nuclear Regulatory Commission (the Commission or the NRC) has found that:
    1. The application for amendment by Exelon Generation Company, LLC (Exelon, the licensee) dated May 10, 2018, as supplemented by letters dated November 1, 2018, and November 29, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Chapter I;
    2. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
    3. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
    4. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
    5. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, Facility Operating License No. DPR-12 is hereby amended by revision to the emergency plan as set forth in the licensee's application dated May 10, 2018, as supplemented by letters dated November 1, 2018, and November 29, 2018, and evaluated in the NRC staff's safety evaluation for this amendment.
3. This license amendment is effective as of the date of its issuance and shall be implemented on or before December 31, 2019.

<u>May 30, 2019</u> – Letter dated May 30, 2019 from Raymond Powell, Chief Decommissioning, ISFSI, and Reactor HP Branch Division of Nuclear Materials Safety to

Bryan Hanson Senior Vice President, Exelon Generation, LLC President and Chief Nuclear Officer, Exelon Nuclear with a subject of NRC INSPECTION REPORT NO. 05000171/2019001, EXELON GENERATION COMPANY, LLC, PEACH BOTTOM ATOMIC POWER STATION UNIT 1, DELTA, PENNSYLVANIA

On May 1-2, 2019, the U.S. Nuclear Regulatory Commission (NRC) conducted an inspection at the Peach Bottom Atomic Power Station Unit 1. The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and the conditions of your license. The inspection consisted of observations by the inspectors, interviews with personnel, and a review of procedures and records. The results of the inspection were discussed with Pat Navin, Site Vice President, and other members of your organization on May 2, 2019, at the conclusion of the inspection. The enclosed report presents the results of this inspection. No findings of safety significance were identified.

Current NRC regulations and guidance are included on the NRC's website at www.nrc.gov; select **Nuclear Materials; Med, Ind, & Academic Uses;** then **Regulations, Guidance and Communications.** The current Enforcement Policy is included on the NRC's website at www.nrc.gov; select **About NRC, Organizations & Functions; Office of Enforcement; Enforcement documents;** then **Enforcement Policy (Under 'Related Information').** You may also obtain these documents by contacting the Government Printing Office (GPO) toll-free at 1-866-512-1800. The GPO is open from 8:00 a.m. to 5:30 p.m. EST, Monday through Friday (except Federal holidays).

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure(s), and your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC document system (ADAMS), accessible from the NRC website at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction.

No reply to this letter is required.

Inspection Report Summary

An announced safety inspection was conducted on May 1-2, 2019, at Unit 1. The inspectors reviewed activities related to the decommissioning performance and status, management oversight, corrective action program (CAP), and site radiological programs. The inspection consisted of interviews with Exelon personnel, a review of procedures and records, and plant walk-downs. The NRC's program for overseeing the safe operation of a shut-down nuclear power reactor is described in Inspection Manual Chapter (IMC) 2561, "Decommissioning Power Reactor Inspection Program." Based on the results of this inspection, no findings of safety significance were identified.

Observations and Findings

The inspectors verified that management oversight was adequate for the SAFSTOR phase of decommissioning and that no significant changes had been made to the Unit 1

SAFSTOR organization since the previous inspection. The inspectors confirmed that no design changes or plant modifications were made since the previous inspection.

The inspectors confirmed surveillances were performed as required by the TS. The inspectors determined site radiological programs were effective in limiting exposure and intrusion water was transferred to Units 2 and 3 in accordance with plant procedures. The inspectors verified that the annual radiological effluent report demonstrated that calculated doses were below regulatory dose criteria of 10 CFR 50, Appendix I.

The inspectors determined that issues were being identified and entered into the CAP in a timely manner and the issues were effectively screened, prioritized and evaluated commensurate with their safety significance.

Conclusion

Based on the results of this inspection, no findings of safety significance were identified

June 27, 2019 – Letter dated June 27, 2019 from Jonathan E. Greives, Chief Reactor Projects Branch 4 Division of Reactor Projects to Bryan C. Hanson Senior Vice President, Exelon Generation Company, LLC President and Chief Nuclear Officer, Exelon Nuclear with subject of PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 – SUPPLEMENTAL INSPECTION REPORT 05000277/2019040 AND 05000278/2019040 AND ASSESSMENT FOLLOW-UP LETTER

On May 16, 2019, the U.S. Nuclear Regulatory Commission (NRC) completed a supplemental inspection at Peach Bottom Atomic Power Station (Peach Bottom), Units 2 and 3 using Inspection Procedure 95001, “Supplemental Inspection Response to Action Matrix Column 2 Inputs,” and discussed the results of this inspection and the implementation of your corrective actions with Mr. Pat Navin, Site Vice President, and other members of your staff. The results of this inspection are documented in the enclosed report.

The NRC performed this inspection to review your station’s actions in response to a White finding in the Mitigating Systems cornerstone, which was documented and finalized in NRC Inspection Reports 05000277/2018013 and 05000278/2018013. The finding involved a failure by Exelon Generation Company, LLC (Exelon) staff at Peach Bottom to establish measures to assure that conditions adverse to quality associated with the E-3 emergency diesel generator (EDG) scavenging air check valve were promptly identified and corrected, which resulted in a failure of the E-3 EDG on June 13, 2018. This finding involved an apparent violation of Title 10 of the *Code of Federal Regulations* Part 50, Appendix B, Criterion XVI, “Corrective Action.” Additionally, as a consequence of the failed E-3 EDG, Exelon also violated Peach Bottom Units 2 and 3 Technical Specification 3.8.1, “Electrical Power Systems - AC Sources – Operating,” since the E-3 EDG was determined to be inoperable for a period greater than the technical specification allowed outage time.

This supplemental inspection was conducted to provide assurance that Exelon adequately identified the root and contributing causes of the event resulting in the E-3 EDG’s failure on June 13, 2018. In addition the inspectors verified that the extent of condition and extent of cause of any performance issues were identified, and the corrective actions for any performance issues were sufficient to address the causes in addition to preventing recurrence.

The NRC determined your staffs' evaluation appropriately identified the root and contributing causes of the White finding. The first root cause was determined to be inadequate work instructions that resulted in an inadequate repair being performed on the scavenging air check valve on April 1, 2017, during an E-3 EDG preventative maintenance window. A second root cause was determined to be inadequate use of operating experience to assist in development of the repair plan. Your staff reviewed the extent of condition for all Peach Bottom EDGs, and no additional degraded conditions were identified. Your staff determined the extent of cause was inadequate preventive maintenance work instructions, which resulted in a review of other station maintenance procedures used for major maintenance, with a focus on check valve and butterfly valve maintenance. Peach Bottom's extent of cause evaluation also identified opportunities for other vendor documents and operating experience to be incorporated into station programs.

The corrective actions to prevent recurrence included revising the EDG maintenance procedure to provide additional detail to identify and correct scavenging air check valve degradation. Additionally, Exelon incorporated the operating experience directly into the diesel engine maintenance procedure to enhance awareness should Exelon personnel identify a degraded inlet air check valve in the future.

The NRC determined that completed and/or planned corrective actions were sufficient to address the performance issues that led to the White finding. Therefore, the performance issue will not be considered as an Action Matrix input after the end of the second calendar quarter of 2019. Based on the results of this inspection and our Action Matrix assessment, the NRC has determined that Peach Bottom, Units 2 and 3 will be transitioned to the Licensee Response Column (Column 1) on July 1, 2019, in accordance with the guidance provided in NRC Inspection Manual Chapter 0305, "Operating Reactor Assessment Program."

Inspection Report Summary

The U.S. Nuclear Regulatory Commission (NRC) reviewed the licensee's planned and completed corrective actions to address a White finding by performing a supplemental inspection using Inspection Procedure 95001, "Supplemental Inspection Response to Action Matrix Column 2 Inputs," at Peach Bottom, Units 2 and 3 in accordance with the Reactor Oversight Process. The Reactor Oversight Process is the NRC's program for overseeing the safe operation of commercial nuclear power reactors. Refer to https://www.nrc.gov/reactors/operating/oversight.html for more information.

The inspectors determined that Peach Bottom appropriately evaluated and understood the root and contributing causes of the significant performance issue. The inspectors also determined that completed or planned corrective actions were sufficient to address

Additional Tracking Items

1. Type – LER
    a. Issue Number 05000277/2018-002-01

   b. Title Emergency Diesel Generator Air Inlet Check Valve Failure Results in a Condition Prohibited by Technical Specifications
   c. Report Section – 71153
   d. Status – Closed
2. Type – NOV
   a. Issue Number 05000277/2018013-01 and 05000278/2018013- 01 EA-18-107
   b. Title Inadequate Corrective Actions result in the failure of the E-3 emergency diesel generator
   c. Report section N/A
   d. Status – closed

July 25, 2019 – email dated July 25, 2019 from Blake Purnell to Thomas Loomis (GenCo-Nuc) cc Lisa Regner, James Barstow (GenCo-Nuc) with a subject of Exelon Generation Company, LLC - Acceptance of Fleet License Amendment Request to Adopt TSTF-427 (EPID L-2019-LLA-0132)

By application dated June 27, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19178A291), Exelon Generation Company, LLC (the licensee) submitted a license amendment request for Braidwood Station, Units 1 and 2; Byron Station, Unit Nos. 1 and 2; Clinton Power Station, Unit No. 1; Dresden Nuclear Power Station, Units 2 and 3; James A. FitzPatrick Nuclear Power Plant; LaSalle County Station, Units 1 and 2; Limerick Generating Station, Units 1 and 2; Nine Mile Point Nuclear Station Unit No. 1; Peach Bottom Atomic Power Station, Units 2 and 3; Quad Cities Nuclear Power Station, Units 1 and 2; and R. E. Ginna Nuclear Power Plant. The proposed amendments would revise the technical specifications based on Technical Specification Task Force (TSTF) traveler TSTF-427, Revision 2, "Allowance for Non Technical Specification Barrier Degradation on Supported System OPERABILITY" (ADAMS Accession No. ML061240055).

The purpose of this e-mail is to provide the results of the NRC staff's acceptance review of this amendment request. The acceptance review was performed to determine if there is sufficient technical information in scope and depth to allow the NRC staff to complete its detailed technical review. The acceptance review is also intended to identify whether the application has any readily apparent information insufficiencies in its characterization of the regulatory requirements or the licensing basis of the plant.

Consistent with Section 50.90 of Title 10 of the Code of Federal Regulations (10 CFR), an amendment to the license must fully describe the changes requested, and following as far as applicable, the form prescribed for original applications. Section 50.34 of 10 CFR addresses the content of technical information required. This section stipulates that the submittal address the design and operating characteristics, unusual or novel design features, and principal safety considerations.

The NRC staff has reviewed your application and concluded that it does provide technical information in sufficient detail to enable the NRC staff to complete its detailed technical review and make an independent assessment regarding the acceptability of the proposed amendment in terms of regulatory requirements and the protection of public health and safety and the environment. Given the lesser scope and depth of the acceptance review as compared to the detailed technical review, there may be instances in which issues that impact the NRC staff's ability to complete the detailed technical review are identified despite completion of an adequate acceptance review. You will be advised of any further information needed to support the NRC staff's detailed technical review by separate correspondence.

Based on the information provided in your submittal, the NRC staff has estimated that the review of this licensing request will take approximately 150 hours to complete. The NRC staff expects to complete this review by July 31, 2020. If there are emergent complexities or challenges in our review that would cause changes to the initial forecasted completion date or significant changes in the forecasted hours, the reasons for the changes, along with the new estimates, will be communicated during the routine interactions with the assigned project manager.

These estimates are based on the NRC staff's initial review of the application and they could change, due to several factors including requests for additional information, unanticipated addition of scope to the review, and review by NRC advisory committees or hearing-related activities. Additional delay may occur if the submittal is provided to the NRC in advance or in parallel with industry program initiatives or pilot applications.

August 9, 2019 – Letter dated August 9, 2019 from Jonathan E. Greives, Chief Reactor Projects Branch 4 Division of Reactor Projects to Brad Berryman President and Chief Nuclear Officer Susquehanna Nuclear with a subject of Peach Bottom Atomic Power Station Units 2 and 3 – Integrated inspection report 05000277/2019002 and 05000278/2019002

On June 30, 2019, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Peach Bottom Atomic Power Station, Units 2 and 3. On July 16, 2019, the NRC inspectors discussed the results of this inspection with Mr. Pat Navin, Site Vice President, and other members of your staff. The results of this inspection are documented in the enclosed report.

One severity level IV violation, without an associated finding, is documented in this report. We are treating this violation as a non-cited violation (NCV) consistent with Section 2.3.2.a of the Enforcement Policy.

If you contest the violation or significance or severity of the violation documented in this inspection report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement; and the NRC Resident Inspector at Peach Bottom.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with 10 CFR 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Inspection Report Summary

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting an integrated inspection at Peach Bottom Atomic Power Station, Units 2 and 3 in accordance with the Reactor Oversight Process. The Reactor Oversight Process is the NRC's program for overseeing the safe operation of commercial nuclear power reactors. Refer to https://www.nrc.gov/reactors/operating/oversight.html for more information.

List of Findings and Violations

Failure to Satisfy 10 CFR 50.72 Reporting Requirements for Loss of Unit 3 Core Spray (CS) Safety Function			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Not Applicable	Severity Level IV NCV 05000277,05000278/2019002-01 Open/Closed	Not Applicable	71153
The inspectors identified a Severity Level IV non-cited violation (NCV) of 10 *Code of Federal Regulations* (CFR) 50.72(b)(3)(v)(D) for not reporting an event or condition to the NRC within eight hours that, at the time of the discovery, could have prevented the fulfillment of a safety function. Specifically, on February 12, 2019, Exelon did not recognize and, therefore, did not report that both 'A' and 'B' trains of the Unit 3 CS systems were inoperable which resulted in a loss of safety function (LOSF).			

Additional Tracking Items

Type	Issue Number	Title	Report Section	Status
LER	05000277,05000278/ 2019-001-00	LER 2019-001-00 for Peach Bottom Atomic Power Station, Unit 2, Regarding Emergency Bus Breaker Relay Failure Results in Loss of Safety Function	71153	Closed

INSPECTION RESULTS

Observation: U-3 RCIC Pressure Switch Failure	71152
The inspectors reviewed Condition Report CR 4129583 that documents Exelon's evaluation, extent of condition reviews, and corrective actions associated with failure of RCIC system pressure switch PS-3-13-72B on April 22, 2018, during a surveillance test. The inspectors focused on Exelon's planned and/or implemented corrective actions to ensure they were commensurate with the significance of the problem. The enforcement aspects of this equipment issue were previously addressed in NRC Inspection Report 05000277/2018003 and 05000278/2018003 (ML18317A003). Exelon's evaluation determined the pressure switch failed due to a flaw in the sensing diaphragm or O-Ring that allowed water to leak into the body of the pressure switch. Exelon's evaluation documented two contributing causes involving their preventive	

maintenance strategy focused on condition monitoring and a single point vulnerability that a failure of the pressure switch could result in a RCIC or HPCI turbine trip.

In review of the extent of condition, Exelon staff identified eleven static O-Ring pressure switches in both units (eight of these switches are associated with HPCI and RCIC) and 36 other switches classified in the Exelon PM program as “critical components” that had been periodically tested but not replaced since original installation. The inspectors determined Exelon staff conducted an appropriate review of the issue, including an adequate extent of condition review and a safety system vulnerability review. Regarding corrective actions, the inspectors determined Exelon staff replaced the failed RCIC pump exhaust pressure switch; performed a visual examination of the seven remaining “critical” static O-Ring pressure switches (RCIC, HPCI) to verify no indications of water intrusion; developed a new replacement PM template for safety components identified as not having a replacement schedule; and developed two separate activities that would either remove the trip function for “critical” instruments to provide alarm only (plant modification) or replace the switches in the next four years. In review of the modification or replacement options, the inspectors noted that replacement of the seven remaining HPCI/RCIC switches is currently scheduled for 2023 and questioned whether this timeframe was commensurate with the potential significance of the issue as these were original components. The inspectors also noted the modification had not yet been developed. In consideration to the inspector’s questions, Exelon staff initiated a corrective action under CR 4129583, to create a new PM to open and inspect the HPCI and RCIC static O-Ring pressure switches on a six-month frequency to verify no water intrusion has occurred, until the switches are replaced or modified. The inspectors concluded this interim corrective action appeared commensurate with the safety significance of the potential water intrusion problem with these original HPCI and RCIC pressure switches.

Observation: Semi-Annual Trend Review	71152

The inspectors evaluated a sample of issues and events that occurred over the course of the first and second quarters of 2019 to determine whether issues were appropriately considered as emerging or adverse trends. The inspectors verified that these issues were addressed within the scope of the CAP or through department review.

Previously, Exelon had identified an adverse trend in equipment reliability in 2018 related to a relatively high number of equipment performance challenges and documented the condition in the CAP under issue report (IR) 4155200. Recently, the frequency of equipment performance challenges has decreased. However, equipment performance remains one of

the top three focus areas for the stations improvement. The inspectors have noted some improvement in the technical rigor involved in station decision making, which was one of the causes that lead to the equipment performance challenges. The inspectors will continue to focus on the station’s performance in this area and implementation of corrective actions from IR 4155200.

Exelon performed a CAP self-assessment in the spring of 2019, and determined that weaknesses existed in the closure quality of corrective action assignments associated with equipment performance. The potential trend was documented in IRs 4217536 and 4239374. The station performed an evaluation and detailed extent of condition review across all major site departments and identified numerous examples of CAP closure deficiencies. The station determined that individuals lacked an adequate questioning attitude and accountability, which directly led to inadequate assignment closure. A CAP get-well plan was developed and implemented to realign the standards of the station on the CAP requirements and establish measures to provide additional CAP oversight. Furthermore, a site supervisor and above stand down was held to review the potential trend and address the issues with the site leadership team. The inspectors reviewed the IRs and determined that Exelon had performed an adequate evaluation and the corrective actions were commensurate with the safety significance of the issue. No additional issues of concern were identified.

Failure to Satisfy 10 CFR 50.72 Reporting Requirements for Loss of Unit 3 CS Safety Function			
Cornerstone	Severity	Cross-Cutting Aspect	Report Section
Not Applicable	Severity Level IV NCV 05000277,05000278/2019002-01 Open/Closed	Not Applicable	71153
The inspectors identified a Severity Level IV non-cited violation (NCV) of 10 *Code of Federal Regulations* (CFR) 50.72(b)(3)(v)(D) for not reporting an event or condition to the NRC within eight hours that, at the time of the discovery, could have prevented the fulfillment of a safety function. Specifically, on February 12, 2019, Exelon did not recognize and, therefore, did not report that both 'A' and 'B' trains of the Unit 3 CS systems were inoperable which resulted in a loss of safety function (LOSF).			
Description: On February 11, 2019, at 2232 hours, an off-site power source (220-08 line) was lost due to a malfunction of a lightning arrestor located at an off-site substation. Peach Bottom's Unit 3 'A' CS train was already inoperable due to planned maintenance at the time of event. Per Peach Bottom's design, six of the site's eight commonly-shared 4kV emergency buses transferred power to their alternate off-site source. During this automatic transfer, a breaker that supplies power to the 480 volt load center (E-434) fed from the E43 emergency 4kV electrical bus failed to automatically re-close due to a failed relay. The E434 load center provides, in part, 480 volt supply power to Unit 3 'B' CS train equipment. Specifically, it provides power to the Unit 3 'D' CS minimum flow and torus suction valves. TS 3.8.7 Condition 'C' was entered for Unit 3, which states, in part, "One Unit 3 AC electrical power distribution subsystem inoperable, restore to operable status within eight hours." At 2250 hours, the E434 breaker was manually closed from the main control room to re-energize the emergency load center and TS 3.8.7 was exited. On February 12, 2019, at 0430 hours, Exelon recognized that TS Surveillance Requirement 3.8.1.11.c.2 and Surveillance Requirement 3.8.1.19.c.2, which require the E434 breaker to			

have the capability to automatically close in order for the E-4 EDG to remain operable, were not met. Exelon subsequently entered TS 3.8.1 Condition E, which requires the off-site source or the EDG to be restored to operable within 12 hours. The E-4 EDG was returned to operable on February 12 at 1559 hours when a replacement relay was

installed and tested in E434. Following the event, the inspectors engaged Exelon staff and challenged that both the ‘A’ and ‘B’ CS trains were inoperable during the event and, thus, a LOSF occurred. Specifically, between 2232 and 2250 hours with the E434 electrical bus de-energized, both CS trains would not have met their specified safety function. In addition, a LOSF occurred when both the E-4 EDG and the ‘A’ CS loop were determined to be inoperable. Furthermore, TS 3.0.6 states, in part, “If a LOSF is determined to exist by this program, the appropriate conditions and required actions of the limiting condition for operation (LCO) in which the LOSF exists are required to be entered.” The inspectors concluded that TS 3.0.3 should have been entered for the CS system LOSF during the event. Since Exelon did not recognize that a LOSF had occurred, Exelon therefore did not report the LOSF condition within eight hours to the NRC in accordance with 10 CFR 50.72(b)(3)(v)(D). Corrective Actions: Exelon reported the LOSF in a subsequent LER 05000277, 05000278/2019-001-00 under 10 CFR 50.73(a)(2)(v)(D) within the required sixty days. Exelon also entered the event into the CAP and conducted an evaluation of the event to address the underlying causes of the missed eight-hour report to the NRC. Exelon conducted training to improve operations crew on-shift proficiency in operability and reportability evaluations. Corrective Action References: IR 4246432
Performance Assessment: The inspectors determined this violation was associated with a minor performance deficiency. The inspectors determined that not recognizing that both Unit 3 CS loops were inoperable and, thus, not reporting the LOSF event within eight hours to the NRC under 10 CFR 50.72(b)(3)(v)(D) was reasonably within Exelon's ability to foresee and correct and should have been prevented and, therefore, was a performance deficiency. Screening: The inspectors reviewed this issue in accordance with IMC 0612 and determined that no more-than-minor ROP finding was identified. Specifically, inspectors determined that the failure to recognize that an LOSF occurred resulted in Exelon not tracking the appropriate TS actions statements. However, inspectors determined that Exelon's actions to restore the system to an operable status were commensurate with the safety significance and entering the appropriate action statement would not have required any additional actions to reduce power or alter plant mode. As such, inspectors determined that the performance deficiency did not adversely affect the mitigating systems cornerstone objective.
Enforcement: The ROP’s significance determination process does not specifically consider the regulatory process impact in its assessment of licensee performance. Therefore, it is necessary to address this violation which impedes the NRC’s ability to regulate using traditional enforcement to adequately deter non-compliance. Severity: The inspectors reviewed Section 6.9.d.9 of the NRC Enforcement Policy and determined this violation was a Severity Level IV violation because the licensee’s failure to make the report, as required by 10 CFR 50.72, did not cause the NRC to reconsider a regulatory position or undertake substantial further inquiry. Specifically, this violation is

similar to Example 9 in the Enforcement Manual, "The licensee fails to make a report required by 10 CFR 50.72 or 10 CFR 50.73."

Violation: 10 CFR 50.72(b)(3)(v)(D) requires, in part, that the licensee shall notify the NRC Operations Center via the Emergency Notification System of any event or condition that at the time of discovery could have prevented the fulfillment of the safety function of structures or systems that are needed to mitigate the consequences of an accident. Contrary to the above, on February 12, 2019, Exelon did not notify the NRC Operations Center via the Emergency Notification System within eight hours of the occurrence of a condition that could have prevented the fulfillment of the safety function of the Unit 3 CS systems that are needed to mitigate the consequences of an accident as required by 10 CFR 50.72(b)(3)(v)(D).

The disposition of this violation closes LER 05000277, 05000278/2019-001-00.

Enforcement Action: This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy.

August 20, 2019 – Email dated August 20, 2019 from Jennifer Tobin to David Helker (GenCo-Nuc) cc Richard Gropp (Exelon Nuclear), Francis Mascitelli (Exelon Nuclear) James Danna with a subject of Peach Bottom Units 2 and 3 - Request for Additional Information - TSTF-500 Implementation LAR (EPID L-2018-LLA-0265)

By application dated June 7, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19158A312), Exelon Generation Company, LLC (Exelon, the licensee) requested an amendment to the Renewed Facility Operating License Nos. DPR-44 and DPR-56 for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3. The license amendment request (LAR) would revise the Technical Specifications (TS) 3.8.4, "DC Sources - Operating," to add a condition for the opposite unit consistent with Nuclear Regulatory Commission (NRC)-approved Technical Specifications Task Force (TSTF)-500, Revision 2, "DC [direct current] Electrical Rewrite – Update to TSTF-360." Specifically, the proposed condition would allow a 72-hour-CT for an opposite unit battery charger that is required for particular plant configurations.

The Nuclear Regulatory Commission's (NRC) staff is reviewing your submittal and has determined that additional information is needed to complete its review. The specific request for additional information (RAI) question is provided below. A clarification phone call was held on August 20, 2019. No changes were made

to the draft RAI (as shown below) as a result of the call. Your response is requested by September 20th in order to allow sufficient review time to meet your expedited review request for this license amendment (December 31, 2019).

PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3

REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATIONS 3.8.4 CONSISTENT WITH TECHNICAL SPECIFICATION TASK FORCE (TSTF) -500, REVISION 2, "DC ELECTRICAL REWRITE – UPDATE TO TSTF-360"
NRC DOCKET NOS. 50-277 AND 50-278

By application dated June 7, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19158A312), Exelon Generation Company, LLC (Exelon, the licensee) requested an amendment to the Renewed Facility Operating License Nos. DPR-44 and DPR-56 for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3. The license amendment request (LAR) would revise the Technical Specifications (TS) 3.8.4, "DC Sources - Operating," to add a condition for the opposite unit consistent with Nuclear Regulatory Commission (NRC)-approved Technical Specifications Task Force (TSTF)-500, Revision 2, "DC [direct current] Electrical Rewrite – Update to TSTF-360," (ADAMS Accession No. ML092670242). Specifically, the proposed condition would allow a 72-hour-CT for an opposite unit battery charger that is required for particular plant configurations.

Title 10 of the *Code of Federal Regulations*, Part 50 (10 CFR 50), Section 36, "Technical Specifications," requires, in part, that the operating license of a nuclear production facility include TS. 10 CFR 50.36 (c)(2) requires that the TS include limiting conditions for operation (LCOs) which are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

RAI -1

The licensee proposed a new TS 3.8.4 Condition B with associated Required Actions and Completion Times (CT) for the required opposite unit battery charger. The licensee stated that the word "required" denotes that only specific batteries from the opposite unit are required to support operation of the unit for particular plant configurations.

The NRC staff has identified the following discrepancy:

Both the 12 hour-CT for Required Action B.1 and the initial 12-hour CT for Required Action B.2 start when Condition B is entered. If the battery terminal voltage was restored to greater than or equal to the minimum established float voltage within 12 hours (Required Action B.1), the battery would be on the exponential charging current portion of its recharging cycle at the end of the 12 hours. It appears that there would be no time remaining for the battery charging current to decrease to less than or equal to 2 amperes (amps) within the same 12 hours (i.e., initial 12-hour CT for Required Action B.2).

The staff requests the following information to address this discrepancy:

Provide a discussion to demonstrate that the required battery can be fully recharged with a charging current of less than 2 amps within the initial 12 hours from entry into Condition B (Required Action B.2) after the required battery terminal voltage is restored to greater than or equal to the minimum established float voltage at the end of 12 hours from entry into Condition B (Required Action B.1).

<u>August 26, 2019</u> – Letter dated August 26, 2019 from Jonathan Greives, Chief Reactor Projects Branch 4 Division of Reactor Projects to Bryan Hanson, Senior Vice President, Exelon generation Company, President and Chief Nuclear Officer, Exelon Nuclear with the subject of Peach Bottom Atomic Power Station, Units 2 and 3 – Biennial Problem Identification and Resolution Inspection Report 05000277/2019010 and 05000278/2019010

On July 12, 2019, the U.S. Nuclear Regulatory Commission (NRC) completed a problem identification and resolution inspection at Peach Bottom Atomic Power Station, Units 2 and 3 and discussed the results of this inspection with Patrick Navin, Site Vice President and other members of your staff. The results of this inspection are documented in the enclosed report.

The NRC inspection team reviewed the station's corrective action program and the station's implementation of the program to evaluate its effectiveness in identifying, prioritizing, evaluating, and correcting problems, and to confirm that the station was complying with NRC regulations and licensee standards for corrective action programs. Based on the samples reviewed, the team determined that your staff's performance in each of these areas adequately supported nuclear safety.

The team also evaluated the station's processes for use of industry and NRC operating experience information and the effectiveness of the station's audits and self-assessments. Based on the samples reviewed, the team determined that your staff's performance in each of these areas adequately supported nuclear safety.

Finally the team reviewed the station's programs to establish and maintain a safety conscious work environment, and interviewed station personnel to evaluate the effectiveness of these programs. Based on the team's observations and the results of these interviews the team found no evidence of challenges to your organization's safety conscious work environment. Your employees appeared willing to raise nuclear safety concerns through at least one of the several means available.

The NRC inspectors did not identify any finding or violation of more than minor significance.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with 10 CFR 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Inspection Report Summary

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting a biennial problem identification and resolution inspection at Peach Bottom Atomic Power Station, Units 2 and 3 in accordance with the Reactor Oversight Process. The Reactor Oversight Process is the NRC's program for overseeing the safe operation of commercial nuclear power reactors. Refer to https://www.nrc.gov/reactors/operating/oversight.html for more information.

List of Findings and Violations

No findings or violations of more than minor significance were identified.

August 26, 2019 – Letter dated August 26, 2019 from Jonathan Greives, Chief Reactor Projects Branch 4 Division of Reactor Projects to Bryan Hanson, Senior Vice President, Exelon generation Company, President and Chief Nuclear Officer, Exelon Nuclear with the subject of Peach Bottom Atomic Power Station, Units 2 and 3 – security biennial problem identification and resolution inspection report 05000277/2019411 and 05000278/2019411

On July 12, 2019, the U.S. Nuclear Regulatory Commission (NRC) completed a problem identification and resolution inspection at Peach Bottom Atomic Power Station, Units 2 and 3 and discussed the results of this inspection with Pat Navin, Site Vice President and other members of your staff. The results of this inspection are documented in the enclosed report. The overall results of the biennial problem identification and resolution inspection are documented in report 05000277/2019010 and 05000288/2019010 (ML19232A305).

The NRC inspectors did not identify any finding or violation of more than minor significance.

This letter will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and the NRC Public Document Room in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Part 2.390, "Public Inspections, Exemptions, Requests for Withholding."

However, the enclosed report contains Security-Related Information, so the enclosed report will not be made publically available in accordance with 10 CFR 2.390(d)(1). If you choose to provide a response that contains Security-Related Information, please mark your entire response "Security-Related Information–Withhold from Public Disclosure under 10 CFR 2.390" in accordance with 10 CFR 2.390(d)(1) and follow the instructions for withholding in 10 CFR 2.390(b)(1). The NRC is waiving the affidavit requirements for your response in accordance with 10 CFR 2.390(b)(1)(ii).

September 18, 2019 – email dated September 18, 2019 from Blake Purnell to Linda Torres Cruz cc Patric Simpson (GenCo-Nuc) and Lisa Regner with subject of Exelon Generation Company – Acceptance of License Amendment Request to Delete Decommissioning Trust License Conditions (EPID L-2019-LLA-0185)

By application dated August 28, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19240B609), Exelon Generation Company, LLC (the licensee) submitted a license amendment request for Braidwood Station, Units 1 and 2; Byron Station, Unit Nos. 1 and 2; Clinton Power Station, Unit No. 1; Dresden Nuclear Power Station, Units 1, 2, and 3; James A. FitzPatrick Nuclear Power Plant; LaSalle County Station, Units 1 and 2; Limerick Generating Station, Units 1 and 2; Nine Mile Point Nuclear Station, Units 1 and 2; Peach Bottom Atomic Power Station, Units 1, 2, and 3; Quad Cities Nuclear Power Station, Units 1 and 2; and R. E. Ginna Nuclear Power Plant. The proposed amendments would delete certain license conditions that specify requirements for decommissioning trust agreements for these facilities. The amendments would also delete some obsolete license conditions associated with completed license transfers for these facilities. The decommissioning trust fund requirements in 10 CFR 50.75(h) would become applicable to these facilities if the amendments are approved.

The purpose of this e-mail is to provide the results of the NRC staff's acceptance review of this amendment request. The acceptance review was performed to determine if there is sufficient technical information in scope and depth to allow the NRC staff to complete its detailed technical review. The acceptance review is also intended to identify whether the application has any readily apparent information insufficiencies in its characterization of the regulatory requirements or the licensing basis of the plant.

Consistent with Section 50.90 of Title 10 of the Code of Federal Regulations (10 CFR), an amendment to the license must fully describe the changes requested, and following as far as applicable, the form prescribed for original applications. Section 50.34 of 10 CFR addresses the content of technical information required. This section stipulates that the submittal address the design and operating characteristics, unusual or novel design features, and principal safety considerations.

The NRC staff has reviewed your application and concluded that it does provide technical information in sufficient detail to enable the NRC staff to complete its detailed technical review and make an independent assessment regarding the acceptability of the proposed amendment in terms of regulatory requirements and the protection of public health and safety and the environment. Given the lesser scope and depth of the acceptance review as compared to the detailed technical review, there may be instances in which issues that impact the NRC staff's ability to complete the detailed technical review are identified despite completion of an adequate acceptance review. You will be advised of any further information needed to support the NRC staff's detailed technical review by separate correspondence.

Based on the information provided in your submittal, the NRC staff has estimated that the review of this licensing request will take approximately 175 hours to complete. The NRC staff expects to complete this review by September 30, 2020. If there are emergent complexities or challenges in our review that would cause changes to the initial forecasted completion date or significant changes in the forecasted hours, the reasons for the changes, along with the new estimates, will be communicated during the routine interactions with the assigned project manager.

These estimates are based on the NRC staff's initial review of the application and they could change, due to several factors including requests for additional information, unanticipated addition of scope to the review, and review by NRC advisory committees or hearing-related activities. Additional delay may occur if the submittal is provided to the NRC in advance or in parallel with industry program initiatives or pilot applications.

September 24, 2019 – Letter dated September 24, 2019 from Bennett Brady, Senior Project Manager, License Renewal Projects Branch, Division of Materials and License Renewal Office of Nuclear Reactor Regulation to Michael Gallagher, Vice President, License Renewal and Decommissioning Exelon Generation Company with the subject of Peach Bottom Atomic Power Station, units 2 and 3 – report for the in-office regulatorty audit regarding the subsequent license renewal application review (EPID No. L-2018-RNW-0012)

By letter dated July 10, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18193A689), Exelon Generation Company, LLC, (Exelon) submitted to the U.S. Nuclear Regulatory Commission (NRC or staff) an application to renew the Renewed Facility Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station, Units 2 and 3 (Peach Bottom), respectively. Exelon submitted the application pursuant to Title 10 of the Code of Federal Regulations Part

54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants,” for subsequent license renewal.

The NRC staff completed its in-office regulatory audit from November 13, 2018, to April 29, 2019, in accordance with the in-office regulatory audit plan (ADAMS Accession No. ML18282A029).

In-Office Regulatory Audit Regarding the Peach Bottom AtomicPower Station, Units 2 and 3,
Subsequent License Renewal Application November 13, 2018 – April 29, 2019

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION, DIVISION OF LICENSE RENEWAL

Docket Nos: License No: Licensee: Facility: Location: Dates: Reviewers:

50-277 and 50-278
DPR-44 and DPR-56
Exelon Generation Company, LLC
Peach Bottom Atomic Power Station, Units 2 and 3 Rockville, Maryland
November 13, 2018 – April 29, 2019

Allik B., Materials Engineer, Division of Materials and License Renewal (DMLR)
Brimfield T., Reactor Systems Engineer, Division of Safety Systems (DSS)

Buford A., Structural Engineer, Division of Engineering (DE) Chereskin A., Chemical Engineer, DMLR
Cheruvenki G., Materials Engineer, DMLR
Cuadrado de Jesus S., Structural Engineer, DE

Fitzpatrick R., Electrical Engineer, DE
Fu B., Materials Engineer, DMLR
Gardner W., Physical Scientist, DMLR
Gavula J., Mechanical Engineer, DMLR
Heida B., Reactor Systems Engineer, DSS Hoang D., Structural Engineer, DE
Hoffman K., Materials Engineer, DMLR
Holston W., Senior Mechanical Engineer, DMLR Huynh A., Materials Engineer, DMLR

Jenkins J., Materials, Engineer, DMLR
Medoff J., Senior Materials Engineer, DMLR Johnson A., Materials Engineer, DMLR
Jones S., Senior Reactor Systems Engineer, DE Khan N., Electrical Engineer, DE
Lehman B., Structural Engineer, DE
Lopez J., Structural Engineer, DE
Min S., Materials Engineer, DMLR

Nguyen D., Electrical Engineer, DE
Nold D., Reactor Systems Engineer, DSS
Patel A., Reactor Engineer, DSS
Prinaris A., Structural Engineer, DE
Rezai A., Materials Engineer, DMLR
Rogers B., Senior Reactor Engineer, DMLR Sadollah M., Electrical Engineer, DE
Thomas G., Senior Structural Engineer, DE

Approved By:

David Alley, Chief
Vessels & Internals Branch
Division of Materials and License Renewal

Steve Bloom, Chief
Chemical, Corrosion, & Steam Generator Branch Division of Materials and License Renewal

Eric Oesterle, Chief
License Renewal Projects Branch
Division of Materials and License Renewal

Angela Buford, Acting Chief
Piping & Head Penetration Branch Division of Materials and License Renewal

Steve Jones, Acting Chief Balance of Plant Branch Division of Safety Systems

Jennifer Whitman, Chief Reactor Systems Branch Division of Safety Systems

Tania Martinez-Navedo, Chief
Electrical Engineering, New Reactors, & License Renewal Branch
Division of Engineering

Joseph Colaccino, Chief Structural Engineering Branch Division of Engineering

1. Introduction

Report for the In-Office Regulatory Audit Peach Bottom Atomic Power Station, Units 2 and 3 Subsequent License Renewal Application

The U.S. Nuclear Regulatory Commission (NRC) staff conducted an in-office audit of Exelon Generation Company, LLC (Exelon, the applicant) Peach Bottom Atomic Power Station (PBAPS) Units 2 and 3 (1) methodology to identify the systems, structures, and components (SSCs) to be included within the scope of subsequent license renewal (SLR) and subject to an aging management review (AMR) (Scoping and Screening

Portion); and (2) aging management programs (AMPs), AMR items, time-limited aging analyses (TLAAs) and associated bases and documentation as applicable (AMP and TLAA Portion) for the subsequent license renewal of Renewed Facility Operating License Nos. DPR-44 and DPR-56 for the Exelon PBAPS, Units 2 and 3.

The purpose of the scoping and screening portion of the audit is to evaluate the scoping and screening process as documented in the license renewal application, implementing procedures, reports, and drawings, such that the NRC staff:

- obtains an understanding of the process used to identify the SSCs within the scope of license renewal and to identify the structures and components subject to an aging management review; and
- has sufficient docketed information to allow the staff to reach a conclusion on the adequacy of the scoping and screening methodology as documented and applied.

The purpose of the AMP and TLAA portion of the audit is to:

- examine Exelon's AMPs, AMR items, and TLAAs
- verify the applicant's claims of consistency with the corresponding NUREG-2191, Rev. 0, "Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report," issued in July 2017, AMPs, and AMR items; and
- assess the adequacy of the TLAAs.

Enhancements and exceptions will be evaluated on a case-by-case basis. The NRC staff's review of enhancements and exceptions will be documented in the safety evaluation report (SER).

Guidance document NUREG-2192, Rev. 0, "Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants" (SRP-SLR), issued in July 2017, provides staff guidance for reviewing an SLR application (SLRA). The SRP-SLR allows an applicant to reference in its license renewal application the AMPs described in the GALL-SLR Report. By referencing the GALL-SLR Report AMPs, the applicant concludes that its AMPs correspond to those AMPs reviewed and approved in the GALL-SLR Report and that no further staff review is required. If an applicant credits an AMP for being consistent with a GALL-SLR Report program, it is incumbent on the applicant to ensure that the plant program contains all of

the elements of the referenced GALL-SLR Report program. The applicant should document this determination in an auditable form and maintain the documentation onsite.

2. Audit Activities

A regulatory audit is a planned, license-related activity that includes the examination and evaluation of primarily non-docketed information. A regulatory audit is conducted with the intent to gain greater understanding of an application, to verify information, and/or to identify information that will require docketing to support the staff's conclusions that form the basis of the licensing or regulatory decision.

Licensing conclusions or staff findings should not be made in the audit reports since licensing and regulatory decisions cannot be made solely based on an audit. Therefore, items identified but not resolved within the scope of the audit will be followed using other NRC processes, such as requests for additional information (RAIs), requests for confirmation of information, and conducting public meetings. Licensing conclusions, staff findings, and resolution of audit items will be documented in the staff's SER.

The following sections discuss the subsequent license renewal application (SLRA) areas reviewed by the staff.

2.1 Aging Management Programs (AMPs)

SLRA AMP B2.1.1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD"

Summary of Information in the Application. The SLRA states that AMP B.2.1.1, "ASME [American Society of Mechanical Engineers] Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," is an existing program that is consistent with the program elements in GALL- SLR Report AMP XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD." To verify this claim of consistency, the staff audited the SLRA AMP.

Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-XI.M1	Program Basis Document ASME Section VI Inservice Inspection, Subsections IWB, IWC, and IWD	Revision 2 4/24/2018
PBT05.G03	PBAPS Units 2 and 3 ISI Program Plan. 4th 10- Year Inspection Interval	Revision 4 Sept. 16, 2014
Passport IR 2677063	Pen and Ink Change to ISI Program Interval Dates	June 1, 2016
ER-AA-330	Conduct of Inservice Inspection Activities	Revision 13

Document	Title	Revision / Date
ER-AA-330- 001	Section XI Pressure Testing	Revision 14
ER-AA-330- 002	Inservice Inspection of Section XI Welds and	Revision 14

	Components	
ER-AA-330- 009	ASME Section XI Repair/Replacement Program	Revision 13
Passport IR 2685419	ISI Feedwater Nozzle Inspection Frequency Change	8/27/2016
AR 04034949	SLR. RX Internals Inspection Documentation Inconsistencies	7/24/2017
AR 04003429	SLR. ISI Database Discrepancies	4/24/2017
AR 02433243	ISI Program Limited Exams P2R20	1/5/2015
AR 04086591	SLR. Clarify Documentation for ISI Exams of MC Supports	12/21/2017
AR 00823657	Inaccurate Weld Category for ISI Exam	9/29/2008
AR 00811174	P2R17 3 ISI weld Inspections not Scheduled	8/26/2008

During the audit, the staff verified Exelon's claim that the "scope of program," "preventive actions," "parameters monitored or inspected," "detection of aging effects," "monitoring and trending," "acceptance criteria," and "corrective actions" program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

During the audit of the "operating experience" program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects.

The staff also audited the description of the SLRA AMP provided in the Updated Final Safety Analyses Report (UFSAR) supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.2, "Water Chemistry"

Summary of Information in the Application. The SLRA states that AMP XI.M2, "Water Chemistry," is an existing program with an exception that will be consistent with the program elements in GALL-SLR Report AMP XI.M2, "Water Chemistry." To verify this claim of consistency, the staff audited the SLRA AMP. Issues identified but not resolved in this report will be addressed in the SER. During the audit, the staff reviewed the exception associated with this AMP. The staff will document its review of the exception to the GALL-SLR Report in the SER.

Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	

		Revision / Date
N/A	Peach Bottom Atomic Power Station Units 2 and 3 Updated Safety Analysis Report (UFSAR)	Revision 26
PB-PBD-AMP-XI.M2	Program Basis Document – Water Chemistry	Revision 2
CY-AB-120-1000	BWR Strategic Water Chemistry Plan	Revision 13
CY-AB-120-0001	Chemistry Action Level Impact Assessments, Engineering Evaluations and Cleanup Projections	Revision 2
ASME ISBN-0-7918-1204-9	Consensus on Operating Practices for the Control of Feedwater and Boiler Water Chemistry in Modern Industrial Boilers	1994 Version
CH-10	Chemistry Goals	Revision 19
CY-AB-120-100	Reactor Water Chemistry	Revision 18
CY-AB-120-110	Condensate and Feedwater Chemistry	Revision 24
CY-AB-120-120	BWR Startup Chemistry	Revision 10
CY-AB-120-130	BWR Shutdown Chemistry	Revision 12
CY-AB-120-200	Storage Tanks Chemistry	Revision 12
CY-AB-120-300	Spent Fuel Pool	Revision 17
CY-AB-120-310	Suppression Pool/Torus Chemistry	Revision 10
CY-AB-120-320	Control Rod Drive Water Chemistry	Revision 8
CY-AA-120-420	Auxiliary Boiler Chemistry	Revision 13
CY-AB-120-1100	Reactor Water Hydrogen Water Chemistry, Noble Chem and Zinc Injection	Revision 13
CY-AA-110-200	Sampling	Revision 13

The staff also verified that aspects of the “scope of program,” “parameters monitored or inspected,” “detection of aging effects,” “monitoring and trending,” “acceptance criteria,” and “corrective actions” program elements not associated with the exception identified in the SLRA

or by the staff during the audit are consistent with the corresponding program elements in the GALL-SLR Report AMP.

In addition, the staff found that for the “preventive actions,” program element, sufficient information was not available to determine whether it was consistent with the corresponding program elements of the GALL-SLR Report AMP. The staff will consider issuing RAIs in order to obtain the information necessary to verify whether this program element is consistent with the corresponding program elements of the GALL-SLR Report AMP. The staff will document its evaluation of the potential RAI in the SER.

During the audit, the staff made the following observations:

- The staff reviewed AR 1511681 and noted that “flags” were created in the plant chemistry database to alert plant personnel to adverse water chemistry trends.
- The staff reviewed UFSAR Table 11.3.1, “Main Condenser,” and noted that the main condenser has titanium tubes. The staff also noted that AMR Item 3.4.1-111 for titanium heat exchanger tubes exposed to treated water is listed as “NA.” This discrepancy was discussed during the audit and will be documented.
- The staff reviewed the water chemistry parameters in procedure CY-AA-120-420 for the Auxiliary Boiler System and noted the parameters are based on the ASME ISBN-0-7918-1204-9 standard and are not included in the Electric Power Research Institute (EPRI) Guidelines referenced by the GALL-SLR.
- The staff reviewed procedure CY-AB-120-1100 and noted that when reactor power is greater than 10 percent there is a monitoring parameter to maintain measured reactor coolant excess dissolved hydrogen >20 ppb.

During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will document its evaluation of the identified plant-specific operating experience in the SER.

The staff also audited the description of the SLRA Water Chemistry program provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.3, “Reactor Head Closure Stud Bolting”

Summary of Information in the Application. The SLRA states that AMP B.2.1.3, “Reactor Head Closure Stud Bolting,” is an existing program that will be consistent, with an exception and an enhancement, with the program elements in GALL-SLR Report AMP XI.M3, “Reactor Head Closure Stud Bolting.” To verify this claim of consistency, the staff audited the SLRA AMP.

Audit Activities. During its audit, the staff discussed with the applicant’s staff and reviewed onsite documentation provided by the applicant.

The table below lists documents that were reviewed by the staff and were found relevant to the audit.

Relevant Documents Reviewed

Document	Title	Revision / Date
PI-AA-115-1003	Processing OE Evaluations	Rev. 4
M-004-400	Reactor Pressure Vessel Reassembly	Rev. 43
M-004-400	Reactor Pressure Vessel Disassembly	Rev. 38
P2R18-168976-HE2-ISI	In-Service Inspection Report for Peach Bottom Power Station	10/2010
P3R18-3Q11-NDE-2LO14H-ISI	In-Service Inspection Report for Peach Bottom Power Station	09/2011
H5814	Reactor Head Spare Stud CMTR	Rev. 0 11/08/1971
003N9506	Peach Bottom Units 2 and 3, Materials Properties and Test Results for Closure Studs, Nuts, Washers and Bushing	Rev. 0 12/2016
AR 00834915	Stuck Stud #80, Lessons Learned for Refuel Floor	09/15/2008

During the audit, the staff verified that the “scope of program,” “parameters monitored or inspected,” “detection of aging effects,” “monitoring and trending,” and “acceptance criteria” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP. The staff also verified that aspects of the “preventive actions” program element not associated with the exception are consistent with the corresponding element of the GALL-SLR Report AMP. The staff’s evaluation of the exception and enhancement to the AMP is documented in the SER.

During the audit of the “operating experience” program element, the staff’s independent database search did not identify any operating experience that would indicate that the AMP may not be adequate to manage the associated aging effects.

The staff also audited the description of the SLRA AMP provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SRP Report.

SLRA AMP B.2.1.4, “BWR Vessel ID Attachment Welds”

Summary of Information in the Application. The SLRA states that AMP B.2.1.4, “BWR Vessel ID Attachment Welds,” is an existing program that is consistent with the program elements in GALL-SLR Report AMP XI.M4, “BWR Vessel ID Attachment Welds.” To verify this claim of consistency, the staff audited the SLRA AMP.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-XI.M4	GALL-SLR PROGRAM XI.M4 – BWR Vessel ID Attachment Welds	Rev. 1 5/6/2018
AR 2735052	2R21 Exam of Steam Dryer Support Brackets	10/31/2016

During the audit, the staff verified Exelon's claim that the "scope of program," "preventive actions," "parameters monitored or inspected," "detection of aging effects," "monitoring and trending," "acceptance criteria," and "corrective actions" program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

During the audit, the staff made the following observations:

- The staff reviewed AR02735052 and noted that new wear was found on the top surface for all support brackets, but the licensee did not identify any linear indications. The licensee did not identify any indications that would impact the integrity of the attachment weld or the reactor vessel.

During the audit of the "operating experience" program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will document its evaluation of the identified plant-specific operating experience in the SER.

The staff also audited the description of the SLRA BWR Vessel ID Attachment Welds AMP provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.5, "BWR Stress Corrosion Cracking Program"

Summary of Information in the Application. The SLRA states that AMP B.2.1.5, "BWR Stress Corrosion Cracking" is an existing program that is consistent with the program elements in "GALL-SLR Report AMP XI.M7," "BWR Stress Corrosion Cracking." To verify this claim of consistency, the staff audited the SLRA AMP.

Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

	Title	Revision /

Document		Date
Exelon Program Basis Document No. PB-PDB-AMP- XI.7	Program Basis Document, BWR Stress Corrosion Cracking	Revision 1

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Document	Title	Revision / Date
AMEC AES Report No. PBT05.G03	ISI Program Plan, Fourth Ten-Year Inspection Interval, Peach Bottom Atomic Power Station, Units 2 and 3	Revision 4, August 6, 2014
General Electric- Hitachi Report No. PB-ISI-14-183901	Fall 2014, ISI/CISI Final Report (ISI Report for P2R20)	Fall 2014
General Electric- Hitachi Report No. 7480-1-24432V- HE3-ISI	Peach Bottom Atomic Power Station (P3R17), In- Service Inspection (ISI) Final Report Summary, 2009 Fall Outage	September 2009
Exelon Confidential/Proprie tary Procedure No. LS-AA-117-1004	10 CFR 50.55a Relief Requests	Revision 7

During the audit, the staff verified that the “scope of program,” “preventive actions,” “parameters monitored or inspected,” “detection of aging effects,” “monitoring and trending,” “acceptance criteria,” and “corrective actions” program elements of the SLRA AMP are consistent with the corresponding elements in the GALL-SLR AMP.

During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER.

The staff also audited the description of the SLRA’s BWR Penetrations Program provided in the UFSAR supplement. The staff verified that this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.6, “BWR Penetrations”

Summary of Information in the Application. The SLRA states that AMP B.2.1.6, “BWR Penetrations” is an existing program that is consistent with the program elements in GALL-SLR Report AMP XI.M8, “BWR Penetrations.” To verify this claim of consistency, the staff audited the SLRA AMP.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
Exelon Program Basis Document No. PB-PDB-AMP- XI.MB	Program Basis Document, BWR Penetrations	Revision 2
General Electric- Hitachi Report No. P2R18-168976- HE2-ISI	Inservice Inspection Report (ISI) for Peach Bottom Atomic Power Station, Refuel Outage 2R18, Fall 2010	October 2010
EPRI Report No. 1007279	BWRVIP-27-A: BWR Vessel and Internals Project, BWR Standby Liquid Control and Core Plate ΔP Inspection and Flaw Evaluation Guidelines	August 2003
EPRI Report No. 1006602	BWRVIP-49-A: BWR Vessel and Internals Project, Instrumentation Penetration Inspection and Flaw Evaluation Guidelines	March 2002
UFSAR Section 3.8	Standby Liquid Control System	Revision 26, April 2017

During the audit, the staff verified that the “scope of program,” “preventive actions,” “parameters monitored or inspected,” “detection of aging effects,” “monitoring and trending,” “acceptance criteria,” and “corrective actions” program elements of the SLRA AMP are consistent with the corresponding elements in the GALL-SLR AMP.

During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER.
The staff also audited the description of the SLRA’s BWR Penetrations Program provided in the UFSAR supplement. The staff verified that this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.7, “BWR Vessel Internals”

Summary of Information in the Application. The SLRA states that AMP B2.1.7, “BWR Vessel Internals,” is an existing program with three enhancements and one exception that is consistent with the program elements in GALL-SLR Report AMP XI.M9, “BWR Vessel Internals.” To verify this claim of consistency, the staff audited the SLRA AMP.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP- XI.M9	GALL-SLR Program XI.M9 – BWR Vessel Internals	Rev. 1 (undated)
AR 02734507	2R21 IVVI Replacement Steam Dryer	10/30/2016
AR 04069252	P3R21 IVVI – Repl. Steam Dryer Lifting Rods Indications	10/31/2017
EC# 621912	Technical Evaluation for P3R21 Replacement Steam Dryer Lifting Rod to Ring Weld Indications	11/03/2017
AR 02570717	3R20 Core Shroud UT Exam	10/14/2015
IR 2573102-03	Technical Evaluation for P320 Core Shroud Weld Examinations Rev. 1	12/01/2015
Structural Integrity Associates (SIA) Calc. Pkg. 1400870.301	Flaw Evaluation for PBAPS U3 Core Shroud Circumferential Welds H1 through H7 and Vertical Welds V3 through V6	Revision 1 11/25/2015
SIA Calc. Pkg. 1400870.302	Core Shroud Off-Axis Flaw Evaluation	10/18/2015
IR 1404300-01	P2R19 Core Shroud R2-SIA Plant Specific Eval	3/15/2013

During the audit, the staff verified Exelon's claim that for the program elements that Exelon declared were consistent, the "scope of program," "preventive actions," "parameters monitored or inspected," "detection of aging effects," "monitoring and trending," "acceptance criteria," and "corrective actions" program elements of the SLRA AMP are consistent with the corresponding

During the audit, the staff made the following observations:

 The staff reviewed AR02734507, AR04069252, and EC#621912 and verified that, in accordance with the exception and enhancements to the AMP, inspection of the replacement steam dryers has been performed consistent with requirements of WCAP-17635-P. The staff noted that cracks were found in some of the non-structural welds which maintained position of the hold down rods and lifting rods during construction. But staff verified that positioning of these rods is guaranteed by the threaded portions and structural welds of these components.

 The staff reviewed AR 02570717 and IR 2573102-03 and noted that numerous indications (one of which extended through-wall) had been documented during the inspection of the Unit 3 core barrel H4 weld. The staff also reviewed SIA Calculation Packages 1400870.301 and 1400870.302. elements of the GALL-SLR Report AMP.

During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER.

The staff also audited the description of the SLRA BWR Vessel Internals provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.8, “Thermal Aging Embrittlement of Cast Austenitic Stainless Steel” (CASS)

Summary of Information in the Application. The SLRA states that AMP B.2.1.8, “Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS),” is a new program that will be consistent with the program elements in GALL-SLR Report AMP XI.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)." To verify this claim of consistency, the staff audited the SLRA AMP.

At the time of the audit, Exelon had not yet fully developed the documents necessary to implement this new program, and the staff’s audit addressed only the program elements described in the applicant’s basis document. The staff will address issues identified but not resolved in this report in the SER.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP- XI.M12	Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)	Revision 1
AR 2671601-07	PBAPS CASS Delta Ferrite Screening Technical Evaluation	2/4/2018
GEH 004N3349	Exelon Nuclear LLC, Peach Bottom Atomic Power Stations Units 2 and 3 Material Properties and Test Results for Recirculation Pump Casing and Cover	Revision 0
ER-AA-330-013	Thermal Aging Embrittlement of Cast Aging Management Program	Revision 2
ER-AA-330-009	ASME Section XI Repair/Replacement Program	Revision 13
AR 2455499	PBAPS CASS Delta Ferrite Screening Technical Evaluation	2/18/2015

During the audit, the staff verified that the “scope of program,” “preventive actions,” “parameters monitored or inspected,” “monitoring and trending,” “acceptance criteria,” and “corrective actions” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

In addition, the staff found that for the “detection of aging effects” program element, sufficient information was not available to determine whether it was consistent with the corresponding program element of the GALL-SLR Report AMP. The staff will consider issuing an RAI in order to obtain the information necessary to verify whether this program element is consistent with the corresponding program element of the GALL-SLR Report AMP. The staff will document its evaluation of this potential RAI in the SER.

During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER.

The staff also audited the description of the thermal aging embrittlement of cast austenitic stainless steel (CASS) provided in the SLRA UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.9, “Flow-Accelerated Corrosion”

Summary of Information in the Application. The SLRA states that AMP B.2.1.9, “Flow-Accelerated Corrosion,” is an existing program with an enhancement that will be consistent with the program elements in GALL-SLR Report AMP XI.M17, “Flow-Accelerated Corrosion.” To verify this claim of consistency, the staff audited the SLRA AMP. Issues identified but not resolved in this report will be addressed in the SER. During the audit, the staff reviewed the enhancement associated with this AMP and will document its review in the SER.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The table below lists the documents that were reviewed by the staff and were found to be relevant to the audit.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-XI.M17	Program Basis Document -Flow-Accelerated Corrosion	Revision 1
ER-AA-430	Conduct of Flow-Accelerated Corrosion Activities	Revision 8
ER-AA-430-1001	Guidelines for Flow-Accelerated Corrosion Activities	Revision 12
ER-AA-430-1004	Erosion in Piping and Components Guide	Revision 2
6200.100-02	Peach Bottom Atomic Power Station Unit 2, FAC	Revision 0

	Susceptibility Non-Modeled Evaluation (SNM)	
6200-100-05	Peach Bottom Atomic Power Station Unit 3, FAC Susceptibility Non-Modeled Evaluation (SNM)	Revision 0

Document	Title	Revision / Date
EP-2016-0079-01- TR	Unit 2 Erosion Susceptibility Evaluation (ESE)	Revision 0
EP-2016-0079-01- TR	Unit 3 Erosion Susceptibility Evaluation (ESE)	Revision 0

During the audit, the staff verified that the “scope of program,” “preventive actions,” “parameters monitored or inspected,” “monitoring and trending,” “acceptance criteria,” and “corrective actions” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP. However, for the “detection of aging effects” program element, sufficient information was not available for the staff to determine whether it was consistent with the corresponding program element of the GALL-SLR Report AMP. The staff will consider issuing RAIs to obtain the information necessary to verify whether this program element is consistent with the corresponding program element of the GALL-SLR Report AMP. The staff will document its evaluation of the potential RAIs in the SER.

During the audit of the “operating experience” program element, the staff conducted an independent search of the plant-specific operating experience database as discussed in the operating experience audit report. The staff will document its evaluation of the identified plant- specific operating experience in the SER.

The staff also audited the description of the Flow-Accelerated Corrosion program provided in SLRA Section A.2.1.9. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.10, “Bolting Integrity”

Summary of Information in the Application. The SLRA states that AMP B.2.1.10, “Bolting Integrity,” is an existing program with enhancements and an exception that will be consistent with the program elements in GALL-SLR Report AMP XI.M18, “Bolting Integrity.” To verify this consistency, the staff audited the SLRA AMP. Issues identified but not resolved in this report will be addressed in the SER. During the audit, the staff reviewed the exception and enhancements associated with this AMP. The staff will document its review of the exception and enhancements in the SER.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-XI.M18	Program Basis Document Bolting Integrity	Revision 1
ER-AA-2030	Conduct of Plant Engineering Manual	Revision 18
MA-AA-410	Bolting Integrity Aging Management Program	Revision 1
ER-AA-335-017	VT-3 Visual Examination of Pump and Valve Internals	Revision 8
MA-AA-736-600	Torquing and Tightening of Bolted Connections	Revision 5
ER-AA-330-001	Section XI Pressure Testing	Revision 14
MA-PB-716-1000	Control of Bolting/Torquing/Tensioning	Revision 0
M-032-001	High Pressure Service Water (HPSW) Pump Maintenance	Revision 6
M-033-001	Emergency Service Water Pump Maintenance	Revision 3
M-037-002	Diesel Driven Fire Pump Maintenance	Revision 2
M-037-004	Motor Driven Fire Pump Maintenance	Revision 2
PMID 00223102-01	Diver Inspection Intake Structure (Unit 2)	N/A
PMID 00222819-01	Diver Inspection Intake Structure (Unit 3)	N/A
PMID 00201232-01	00P186: Diver Inspection, Mud Sample & Depth	N/A

During the audit, the staff verified that for the program elements that Exelon declared were consistent, the “parameters monitored or inspected,” “monitoring and trending,” “detection of aging effects” and “acceptance criteria” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP. The staff also verified Exelon’s claim that aspects of the “scope of program” program element not associated with the exception identified in the SLRA are consistent with the corresponding program elements in the GALL- SLR Report AMP. In addition, the staff found that for the “preventive actions” program element, sufficient information was not available to verify whether this program element is consistent with the corresponding program element of the GALL-SLR Report AMP. The staff will document its evaluation of this potential RAI in the SER.

During the audit, the staff made the following observations:

 The staff reviewed the applicant’s program basis document PB-PBD-AMP-XI.M18 and noted that it states “Aging Management Reviews have determined that high strength bolting material with actual yield strength of 150 ksi or greater (high strength) bolting are [sic] used for closure bolting, with 2 inches or less diameters, on pressure-retaining components within the scope of license renewal. Corporate level procedures require engineering approval to use high strength bolting material in system components within the scope of license renewal. Existing site procedures will be revised to minimize the use of high strength closure bolting material in portions of systems within the scope of license renewal." The staff notes that this is not consistent with the "preventive actions" program element of GALL- SLR Report AMP XI.M18 which recommends that preventive measures include using bolting material that has an actual measured yield strength less than 150 kilo-pounds per square inch (ksi) or 1,034 mega pascals (MPa).

During the audit of the "operating experience" program element, the staff independently searched for plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER.

The staff also audited the description of the SLRA AMP provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.11, "Open-Cycle Cooling Water System"

Summary of Information in the Application. The SLRA states that the AMP B.2.1.11, "Open- Cycle Cooling Water System" is an existing program with an enhancement that will be consistent with the program elements in GALL-SLR Report AMP XI.M20, "Open-Cycle Cooling Water System." To verify this claim of consistency, the staff audited the SLRA AMP. During the audit, the staff reviewed the enhancement associated with this AMP. The staff will document its review of the enhancement in the SER.

Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-XI.M20	Program Basis Document, Open-Cycle Cooling Water System	Revision 1
ER-AA-340	Generic Letter 89-13 Implementing Procedure	Revision 8
ER-AA-340-1001	Generic Letter 89-13 Program Implementation Instructional Guide	Revision 10
ER-AA-340-1002	Service Water Heat Exchanger Inspection Guide	Revision 6
ER-AA-2001	2016 Raw Water Integrity Update	09/21/2016
ER-AA-340-2000	Balance-of-Plant Heat Exchanger Inspection, Testing and Maintenance Guide	Revision 8

Document	Title	Revision / Date
ER-AA-5400-1001	Raw Water Piping Integrity Management Guide	Revision 11
CY-AA-120-410	Circulating/Service Water Chemistry	Revision 6
CY-AA-120-4110	Raw Water Chemistry Strategic Plan	Revision 10
CY-AA-120-4110- F-08	Peach Bottom Raw Water Treatment and Control	Revision 2
CY-PB-120-707	High Pressure Service Water System Monitoring	Revision 0
CY-PB-190-9003	Cooling Water Chemistry Monitoring Program	Revision 1
AO 33.5.A	Residual Heat Removal, Core Spray, High Pressure Coolant Injection, Reactor Core Isolation Cooling Flush	Revision 2
NA	Generic Letter 89-13 Program Basis Document	10/03/2016
NA	Peach Bottom Atomic Power Station, Units 2 and 3, Response to Generic Letter 89-13, “Service Water System Problems Affecting Safety-Related Equipment”	01/29/1990
NA	Peach Bottom Atomic Power Station, Units 2 and 3, Generic Letter 89-13, “Service Water System Problems Affecting Safety-Related Equipment” Implementation of Actions	06/01/1992
IR 01541900-02	Technical Evaluation for Peach Bottom Atomic Power Station Raw Water Corrosion Rate and Remaining Life Basis	03/26/2014
02734068-04	Technical Evaluation for Peach Bottom Atomic Power Station High Pressure Service Water Non- Destructive Examination and Integrity Basis	01/18/2017
PVP2014-28781	Piping Corrosion Rate and Remaining Life Basis: Commercializing Conservatism in First Time Inspections	07/20/2014
M-010-002	Residual Heat Removal Heat Exchanger Maintenance	Revision 17
RT-I-033-631-2 RT-I-033-631-3	Residual Heat Removal Room Cooler Emergency Service Water Heat Transfer Test	Revision 12 Revision 11

Document	Title	Revision / Date
RT-I-033-632-2 RT-I-033-632-3	Core Spray Room Cooler Emergency Service Water Heat Transfer Test	Revision 11 Revision 12
RT-M-033-675-2 RT-M-033-675-3	Pump Intake Structure Inspection and Cleaning	Revision 5 Revision 6

RT-O-010-660-2 RT-O-010-660-2	Residual Heat Removal Heat Exchanger Performance Test	Revision 15 Revision 14
RT-O-095-827-2	Chlorination of Circulating and Service Water	Revision 12

During the audit, the staff verified that the “scope of program,” “preventive actions,” “parameters monitored or inspected,” “detection of aging effects,” “monitoring and trending,” “acceptance criteria,” and “corrective actions” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects.

The staff also audited the description of the SLRA Open-Cycle Cooling Water System program provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.12, “Closed Treated Water Systems”

Summary of Information in the Application. The SLRA states that AMP B.2.1.12, “Closed Treated Water Systems,” is an existing program with an enhancement that, other than a stated exception, will be consistent with the program elements in GALL-SLR Report AMP XI.M21A, “Closed Treated Water Systems.” To verify this claim of consistency, the staff audited the SLRA AMP. During the audit, the staff reviewed the exception to the GALL-SLR Report AMP and the enhancement associated with this AMP. The staff will document its reviews of the exception and the enhancement in the SER.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The table below lists the documents that were reviewed by the staff and were found to be relevant to the audit.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-XI.M21A	Program Basis Document – Closed Treated Water Systems	Revision 1
CH-10	Chemistry Goals	Revision 20

Document	Title	

		Revision / Date
CY-AA-120-400	Closed Treated Water Chemistry	Revision 19
CY-AA-120-4000	Closed Treated Water Chemistry Strategic Plan	Revision 8
ER-AA-700-NEW	Inspection of Components Within the Scope of the Closed Treated Water Systems Aging Management Program	Revision 0
EPRI 3002000590	Closed Cooling Water Chemistry Guideline	Revision 2

During the audit, the staff verified that the “scope of program,” “preventive actions,” “parameters monitored or inspected,” “detection of aging effects,” “monitoring and trending,” “acceptance criteria,” and “corrective actions” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP. The staff also verified the portions of the “parameters monitored or inspected” program element that are not associated with the exception identified in the SLRA are consistent with the corresponding program element in the GALL-SLR Report AMP.

During the audit of the “operating experience” program element, the staff conducted an independent search of the plant-specific operating experience database as discussed in the operating experience audit report. The staff will evaluate the identified plant-specific operating experience in the SER.

The staff also audited the description of the Closed Treated Water Systems program provided in SLRA Section A.2.1.12. The staff verified it is consistent with the description provided in the GALL-SLR Report Table XI-01.

SLRA AMP B.2.3.13, “Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems”

Summary of Information in the Application. The SLRA states that AMP B.2.3.13, “Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems,” is an existing program with enhancements that will be consistent with the program elements in GALL-SLR Report AMP XI.M23, “Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems.” To verify this claim of consistency, the staff audited the SLRA AMP. Issues identified but not resolved in this report will be addressed in the SER. During the audit, the staff reviewed enhancements associated with this AMP. The staff will document its review of the enhancements in the SER.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-XI.M23	Program Basis Document Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems	Revision 1
MA-PB-716-021- 1000	Guideline for Rigging and Handling Heavy Loads	Revision 10
MA-AA-716-021	Rigging and Lifting Program	Revision 27
M-017-001	Periodic Inspection of Reactor Building Crane	Revision 3
M-C-700-327	Periodic Inspection of Electric an Air Operated Hoisting Devices	Revision 7
M-017-007	Periodic Inspection of the Turbine Building Cranes	Revision 5
M-C-797-008	Fuel Preparation Machine Maintenance	Revision 11
M-C-797-014	Refueling Platform Main Hoist Mechanical and Electrical Inspection and Maintenance	Revision 10
M-C-797-015	Refueling Platform Auxiliary Hoists Mechanical and Electrical Inspection and Maintenance	Revision 4
M-C-797-017	Refueling Platform Bridge Drive and Components Mechanical and Electrical Maintenance/Inspections	Revision 6
M-C-797-018	Refueling Platform Trolley Mechanical and Electrical Maintenance/Inspections	Revision 4
WC-AA-120-F02	System 17 New PM for PB-SLR Crane Inspections	Revision 0

During the audit, the staff verified that the “scope of program,” “preventive actions,” “parameters monitored or inspected,” “detection of aging effects,” “monitoring and trending,” and “acceptance criteria” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

During the audit, the staff made the following observations:

- The staff reviewed the SRP-SLR and respective SLRA AMR item 3.5.1-100 and noted that this item addresses cracking due to stress corrosion cracking (SCC) of stainless steel bolting components/connections. In addition to item 3.5.1-100, the staff also noted that for item 3.3.1-199, Table 3.3.2-15, “Fuel Handling System,” AMR item VII.B.A-730, high strength low alloy steel bolting with yield strength of 150 ksi, the SLRA states that the Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems (AMP B.2.3.13) will be used to manage cranes’ structural bolting for the aging effect of cracking due to SCC. The staff notes that the GALL-SLR Report recommends managing the aging effects of cracking due to SCC, but those recommendations are under other AMPs, such as GALL-SLR Report AMP XI.S3, “ASME Section XI, Subsection IWF” and GALL-SLR Report AMP XI.S6, “Structures Monitoring.” The GALL-

SLR Report AMPs XI.S3 and XI.S6 recommendation regarding the detection of cracking due to SCC is that structural bolting with actual measured yield strength greater

than or equal to 150 ksi in sizes greater than 1 inch nominal diameter be subject to volumetric examination comparable to that of ASME Code Section XI, Table IWB-2500-1, Examination Category B-G-1. For the aging effects of cracking due to SCC for bolts with actual measured yield strength greater than or equal to 150 ksi and a diameter greater than 1 inch, the staff notes that the GALL–SLR Report AMP XI.M23 does not include surface or volumetric examination recommendations for the applicant's AMP B.2.3.13 to address this aging effect. Therefore, the staff may submit an RAI to ascertain how the applicant's AMP B.2.3.13 is adequate to manage this aging effect for the subject bolted connections.

During the audit of the "operating experience" program element, the staff independently searched for plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER.

The staff also audited the description of the SLRA AMP provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.14, "Compressed Air Monitoring"

Summary of Information in the Application. The SLRA states that AMP B.2.1.14, "Compressed Air Monitoring," is an existing program with enhancements that will be consistent with the program elements in GALL-SLR Report AMP XI.M24, "Compressed Air Monitoring." To verify this claim of consistency, the staff audited the SLRA AMP.

Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-XI.M24	Program Basis Document- Compressed Air Monitoring	Revision 01
TQ-AA-161	Maintenance Training Program	Revision 08
TQ-AA-161-J010	Maintenance Initial Training Matrix Job Aid	Revision 02
ACAD 92-008	Guidelines for the Training and Qualification of Maintenance Personnel	09/1992
SLR-PB-M-333	License Renewal Drawings, Instrument Nitrogen	Revision 0
SLR-PB-M-320	License Renewal Drawings, Compressed Air System	Revision 0
SLR-PB-M-351	License Renewal Drawings, Nuclear Boiler	Revision 0

Document	Title	Revision / Date
SLR-PB-M-367	License Renewal Drawings, Containment Atmospheric Control	Revision 0
SLR-PB-M-372	License Renewal Drawings, Containment Atmospheric Dilution	Revision 0

During the audit, the staff verified that the “scope of program,” “preventive actions,” “parameters monitored or inspected,” “detection of aging effects,” “monitoring and trending,” “acceptance criteria,” and “corrective actions” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

During the audit, the staff made the following observation:

- The staff reviewed TQ-AA-161 and confirmed that the applicant is using qualified inspectors to inspect components that are associated with the Compressed Air Monitoring program.

During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER.

The staff also audited the description of the SLRA AMP provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.15, “BWR Reactor Water Cleanup System”

Summary of Information in the Application. The SLRA states that AMP B.2.1.15, “BWR Reactor Water Cleanup System,” is an existing program that is consistent with the program elements in GALL-SLR Report AMP XI.M25, “BWR Reactor Water Cleanup System.” To verify this claim of consistency, the staff audited the SLRA AMP.

Audit Activities. During its audit, the staff reviewed onsite documentation provided by Exelon. The table below lists the documents that were reviewed by the staff and were found relevant to the audit.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-XI.M25	“Program Basic Document - BWR Reactor Water Cleanup System	Revision 2
ML090930466	NRC letter dated September 15, 1995, “Reactor Water Cleanup (RWCU) System Weld Inspections at Peach Bottom Atomic Power Station, Units 2 and 3 (TAC Nos. M92442 and M92443)”	September 15, 1995

Document	Title	Revision / Date
Plant OE - XI.M25	Plant OE - XI.M25 BWR Reactor Water Cleanup System Aging Management Program	
Implementing Documents - XI.M25	Implementing Documents - XI.M25 BWR Reactor Water Cleanup System Aging Management Program	
Plant Operating Experience – X1.M2	Water Chemistry – Plant Operating Experience – XI.M2 Water Chemistry Program	

During the audit, the staff verified that the “scope of program,” “preventive actions,” “parameters monitored or inspected,” “detection of aging effects,” “monitoring and trending,” “acceptance criteria,” and “corrective actions” program elements of SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

During the audit, the staff made the following observations:

- The staff reviewed the program basis document and noted that the document provides supporting evidence that the augmented ISI performed as part of this AMP has been effective in both detecting cracks in piping welds susceptible to the intergranular stress corrosion cracking (IGSCC) and managing aging effects of the reactor water cleanup (RWCU) system piping. As an example, the operating experience reviews revealed detection of the IGSCC cracks in the Peach Bottom RWCU piping welds in 2017 and 1996 by the volumetric examinations. The corrective actions and sample expansions were taken and reported to the NRC.
- The staff verified from review of NRC letter dated September 15, 1995, the program basis document, and NRC GL 88-01 that the volumetric examinations have been performed on the RWCU system piping welds identified in SLRA in accordance with the NRC approved alternative. The NRC-approved alternative includes 2 percent of the IGSCC susceptible welds to be inspected each refueling outage.
- The staff noted that this provides sufficient demonstration that the effects of aging have been and will be adequately managed so that the intended function will be maintained for the subsequent period of extended operation, as required by 10 CFR 54.21(a)(3). This staff determination will be reflected in the staff’s SER for the Peach Bottom SLRA.

During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects.

The staff also audited the description of the SLRA BWR Reactor Water Cleanup System provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR for AMPs.

SLRA AMP B.2.1.16, “Fire Protection”

Summary of Information in the Application. The SLRA states that AMP B.2.1.16, “Fire Protection,” is an existing program with enhancements that will be consistent with the program

elements in GALL-SLR Report AMP XI.M26, “Fire Protection.” To verify this claim of consistency, the staff audited the SLRA AMP. Issues identified but not resolved in this report will be addressed in the SER. During the audit, the staff reviewed the enhancements associated with this AMP. The staff will document its review of the enhancements in the SER.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-XI.M26	Program Basis Document Fire Protection	Revision 1
CC-AA-211	Fire Protection Program	Revision 8
ER-AA-2030	Conduct of Plant Engineering Manual	Revision 18
ST-M-037-399-2	Fire Damper Inspection	Revision 11
ST-M-037-395-2	U/2 Fire Damper Inspection	Revision 2
ST-M-037-395-3	U/3 Fire Damper Inspection	Revision 2
ST-M-037-350-2	Safety Related Door Inspection	Revision 7
N/A	List of Doors That Require Replacement	N/A
N/A	List of Doors That Have Been Replaced	N/A
ST-M-037-311-2	Detailed Visual Inspection of Penetration Seals and Difficult to View Fire Barriers	Revision 10
ST-M-037-311-3	Detailed Visual Inspection of Penetration Seals and Difficult to View Fire Barriers	Revision 11
ST-M-037-313-2	Visual Inspection of Fire Protective Steel Beam Coating	Revision 3

	and Cable Tray Covers	
ST-M-037-314-2	Visual Inspection of Encapsulated Electrical Raceways	Revision 6
Drawing A-484	Barrier Plans Drawing at Elevation 91 Feet 6 Inches	Revision 8
Drawing A-485	Barrier Plans Drawing at Elevation 116 Feet 0 Inches	Revision 4

Document	Title	Revision / Date
Drawing A-486	Barrier Plans Drawing at Elevation 135 Feet 0 Inches	Revision 11
Drawing A-487	Barrier Plans Drawing at Elevation 165 Feet 0 Inches	Revision 1
Drawing A-488	Barrier Plans Drawing at Elevation 195 Feet 0 Inches	Revision 7
Drawing A-489	Barrier Plans Drawing at Elevation 234 Feet 0 Inches	Revision 4
Drawing A-490	Barrier Plans Cooling Water Pump Structure, Emergency Cooling Tower, and Diesel Generator Building	Revision 5
NE-075	Penetration Seals in Hazard Barriers at Peach Bottom Atomic Power Station and Limerick Generating Station	Revision 4

During the audit, the staff verified that the “scope of program,” “preventive actions,” “parameters monitored or inspected,” “detection of aging effects,” “monitoring and trending,” “acceptance criteria,” and “corrective actions” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects.

The staff also audited the description of the SLRA Fire Protection program provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.17, “Fire Water System”

Summary of Information in the Application. The SLRA states that AMP B.2.1.17, “Fire Water System,” is an existing program with an exception and enhancements that will be consistent with the program elements in GALL-SLR Report AMP XI.M27 “Fire Water System.” To verify this claim of consistency, the staff audited the SLRA AMP. Issues identified but not resolved in this report will be addressed in the SER. During the audit, the staff reviewed the exception and enhancements associated with this AMP. The staff will document its review of the exception to the GALL-SLR Report AMP and the enhancements in the SER.

Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-XI.M27	Program Basis Document – Fire Water System	Revision 1
NA	Technical Reviewer Manual – Fire Water System	Revision 9
NA	Fire Protection Program	Revision 17
P-S-51	Design Baseline Document – Fire Protection System	Revision 12
ST-O-37B-313-2	Hose Station Block Valve Operability and Blockage Check	Revision 7
ST-O-37B-322-2	13KV Switchgear Area Sprinkler System Actuation [with marked up changes]	Revision 6
ST-O-37B-323-2	Unit 2 Battery Room, 4KV Switchgear Rooms, and Rad Waste Corridor Area Sprinkler System Actuation [with marked up changes]	Revision 9
RT-O-37B-326-2	Reactor Feedpump Turbine Area Sprinkler System Actuation	Revision 4
RT-O-37B-326-3	Reactor Feedpump Turbine Area Sprinkler System Actuation	Revision 5
RT-O-37B-327-3	Turbine Bearing Sprinkler System Actuation	Revision 6
RT-O-37B-328-2	Sprinkler Alarm Valve Test Potentially Hi-Rad	Revision 4
RT-O-37B-329-2	Common Systems Sprinkler Alarm Valve Test in Non Hi-Rad Areas	Revision 7
RT-O-37B-351-2	2AX001 A Main Transformer Deluge System Functional Test	Revision 8
RT-O-37B-353-3	3cx001 C MAIN Transformer Deluge System Functional Test	Revision 7
RT-O-37B-358-2	Hydrogen Seal Oil Unit Sprinkler Flooding Valve Actuation Test	Revision 8
ST-O-37B-381-2	Underground Fire Main Flow Test	Revision 13
RT-O-37B-382-2	Fire Hydrant Inspection and Flush	Revision 8

Document	Title	Revision / Date
RT-O-37B-383-2	Wet Pipe Sprinkler System Non-ACV Functional Test	Revision 0
ER-AA-5400-1001	Raw Water Corrosion Program Guide	Revision 10
IR 1275720-04-08	Verify Slope and Drainage Points	04/30/2012
IR 1275720-05-08	Verify Flow Testing Procedure Adequacy	04/30/2012
IR 1275720-09-08	Site Check of Dry Pipe System Susceptibility	06/27/2014
IR IR02512545	Unsatisfactory Flow Test Results During Performance of ST-O-37B-381-2	06/10/2015
AR 04163257	First License Renewal Sprinkler Head 50 Year Test – Unit 3 Main Stop Valve/Bypass Valve Platform	08/10/2018
AR 04163262	First License Renewal Sprinkler Head 50 Year Test – Unit 3 Feedwater Heater West Service Platform	08/10/2018
AR 04163273	First License Renewal Sprinkler Head 50 Year Test – Unit 3 Feedwater Heater East Service Platform	08/10/2018
AR 04135918	50 Year Sprinkler Test Plan for First License Renewal	05/09/2018
AR 04131892	Unsatisfactory Flow Test Results During Performance of ST-O-37B-381-2	04/28/2018
AR 04189276	Revise ST-O-37B-381-2 to Utilize Ultrasonic Flow Meters	10/30/2018
WO 04274779	Preventive Maintenance: Clean/Inspect/Rework BS-0421 Internals	02/22/2016
6280-M-318	P&I [Piping and Instrumentation] Diagram Fire Protection System	Revision 48
AR 01153311	Water on Unit 3 M/S Floor [provided by Exelon in relation to questions on leaking sprinklers – 10 drops per minute]	12/16/2010
AR 01343009	Sprinkler Head Leaking in MCU Rebuild Area 165 Elevation Turbine Building [32 props per minute]	03/20/2012

Document	Title	Revision / Date
AR 01663767	Leak from Fire System Sprinkler Unit 2 Main Turbine Lube Oil Sprinkler Alarm Valve Network [3 drops per minute]	05/23/2014
AR 04055356	Two Fire System Sprinkler Heads Leaking in the Main Turbine Lube Oil Room Walkway	09/24/2017

NA	Letter to USNRC Response to Request for Additional Information Related to License Renewal	01/31/2003

During the audit, the staff verified for the program elements that Exelon declared consistent, the “scope of program” and “monitoring and trending” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

The staff also verified Exelon’s claim that aspects of the “preventive actions” and “detection of aging effects” “program elements” not associated with the exceptions identified in the SLRA or by the staff during the audit are consistent with the corresponding program elements in the GALL-SLR Report AMP.

In addition, the staff found that for the “preventive actions,” “parameters monitored or inspected,” “detection of aging effects,” “acceptance criteria,” and “corrective actions” program elements, sufficient information was not available to determine whether they were consistent with the corresponding program elements of the GALL-SLR Report AMP. The staff will consider issuing RAIs in order to obtain the information necessary to verify whether these program elements are consistent with the corresponding program elements of the GALL-SLR Report AMP.

During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER. The staff will consider issuing an RAI in order to obtain the information necessary to determine whether Exelon’s SLRA can be adequate to manage the associated aging effects.

The staff also audited the description of the SLRA Fire Water System program provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.18, “Outdoor and Large Atmospheric Metallic Storage Tanks”

Summary of Information in the Application. The SLRA states that SLRA Section B.2.1.18, “Outdoor and Large Atmospheric Metallic Storage Tanks,” is an existing program with enhancements that will be consistent with the program elements in GALL-SLR Report AMP XI.M29, “Outdoor and Large Atmospheric Metallic Storage Tanks.” To verify this claim of consistency, the staff audited the SLRA AMP. During the audit, the staff reviewed enhancements associated with this AMP. The staff will document its review of the enhancements in the SER.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-XI.M29	Program Basis Document, Outdoor and Large Atmospheric Metallic Storage Tanks	Revision 1
ER-AA-700-404	Aging Management Program for Aboveground Metallic Tanks	Revision 2
Drawing C-60	Field Erected Tank Foundations Fuel Oil, Clarified Water and Demineralized Storage Tanks	Revision 14
Drawing C-61	Field Erected Tank Foundations Refueling and Condensate Storage Tanks	Revision 16
Drawing C-24-33	Elevation 44-0 Diameter by 42-0 Height Refueling Water Tank	October 6, 1969
Drawing C-24-39	Orientation and Bottom Plan Refueling Water Tank	N/A
Drawing C-24-41	Elevation 30-0 Diameter by 42-0 Height Condensate Storage Tank	October 7, 1969
RT-O-100-911-2	Inspection of Aboveground Storage Tanks	Revision 10
WC-AA-120-F-02 (212845)	Project Manager Annotations for Peach Bottom Subsequent License Renewal Refueling Water Storage Tank Inspections	Revision 0
WC-AA-120-F-02 (227500)	Project Manager Annotations for Peach Bottom Subsequent License Renewal Unit 2 Condensate Storage Tank Inspections	Revision 0
WC-AA-120-F-02 (227501)	Project Manager Annotations for Peach Bottom Subsequent License Renewal Unit 3 Condensate Storage Tank Inspections	Revision 0
RT-O-100-911-2	Inspection of Aboveground Tanks	Revision 9
Report No. NUC2014134	Condition Assessment of Coatings Applied to the Exterior of Tanks	Revision 0
LTAM	Long Term Asset Management Strategy for Tanks	Revision 3

During the audit, the staff verified that for the program elements that Exelon declared were consistent, the “scope of program,” “preventive actions,” “parameters monitored or inspected,” “detection of aging effects,” “monitoring and trending,” “acceptance criteria,” and “corrective actions” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects.

The staff also audited the description of the SLRA Outdoor and Large Atmospheric Metallic Storage Tanks program provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.19, “Fuel Oil Chemistry”

Summary of Information in the Application. The SLRA states that AMP B.2.1.19, “Fuel Oil Chemistry,” is an existing program with enhancements that will be consistent with the program elements in GALL-SLR Report AMP XI.M30, “Fuel Oil Chemistry.” To verify this claim of consistency, the staff audited the SLRA AMP. Issues identified but not resolved in this report will be addressed in the SER. During the audit, the staff reviewed the enhancements associated with this AMP. The staff will document its review of the enhancements in the SER.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-XI.M30	Program Basis Document – Fuel Oil Chemistry	Revision 1
P&ID M-377	Diesel Fuel Oil System	09-17-2007
DWG E-5-36	EDG Day Tank	10-15-1973
DWG C-28-16	EDG Main Fuel Oil Tank	03-1970
ECR-PB-94-08147	Diesel Fire Pump Fuel Oil Storage Tank	Revision 0
DWG M-16-22	Diesel Fire Pump Day Tank	08/05/1977
PES-P-006	Diesel Fuel Oil	Revision 11
CY-PB-130-755	Determination of Particulate Contamination in Diesel Fuel Oil	Revision 0

During the audit, the staff verified that the “scope of program,” “preventive actions,” “parameters monitored or inspected,” “monitoring and trending,” “acceptance criteria,” and “corrective

actions” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP. In addition, the staff found that for the “detection of aging effects,” program element, sufficient information was not available to determine whether it was consistent with the corresponding program elements of the GALL-SLR Report AMP. The staff will consider issuing a RAI in order to obtain the information necessary to verify whether this program element is consistent with the corresponding program element of the GALL-SLR Report AMP. The staff will document its evaluation of this potential RAI in the SER.

During the audit, the staff made the following observation: The staff reviewed PB-PBD-AMP- XI.M30 and noted that the "detection of aging effects" portion of the document states that the samples for the diesel generator fuel oil storage tanks are withdrawn from the fuel oil transfer pump suction piping while the transfer pump is in service.

During the audit of the "operating experience" program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER.

The staff also audited the description of the SLRA AMP provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.20, "Reactor Vessel Material Surveillance Program"

Summary of Information in the Application. The SLRA states that AMP B.2.1.20, "Reactor Vessel Material Surveillance" Program (RVMSP) is an existing program that, with an enhancement, will be consistent with the program elements in GALL-SLR Report AMP XI.M31, "Reactor Vessel Material Surveillance." To verify this claim of consistency, the staff audited the SLRA AMP. Issues identified but not resolved in this report will be addressed in the SER. During the audit, the staff reviewed the enhancement associated with this AMP. The staff will document its review of the enhancement in the SER.

Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-XI.M31	Program Basis Document: Reactor Vessel Material Surveillance, GALL-SLR Program XI.M31 – Reactor Vessel Material Surveillance	Revision 1, 09/25/2018
PBAPS UFSAR Section, 4.2.6	Inspection and Testing	Rev. 14, April 2017
PBAPS UFSAR Appendix Q	License Renewal Aging Management UFSAR Supplement	Rev. 14, April 2017
EPRI Proprietary Report No. 1025144[1]	BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan (BWRVIP-86 Revision 1-A)	Revision 1-A, October 2012
EPRI Report 1021553	BWRVIP-87NP, Revision 1: BWR Vessel and Internals Project, Testing and Evaluation of BWR	Revision 1, August 2010

Document	Title	

		Revision / Date
(ADAMS ML102420110)	Supplemental Surveillance Program Capsules D, G, and H	
EPRI Report 1021554 (ADAMS ML102720220)	BWRVIP-111NP, Revision 1: BWR Vessel and Internals Project, Testing and Evaluation of BWR Supplemental Surveillance Program Capsules E, F, and I	Revision 1, August 2010
EPRI Report 1021555 (ADAMS ML102580248)	BWRVIP-113NP: BWR Vessel and Internals Project, River Bend 183 Degree Surveillance Capsule Report	Revision 0, August 2010
EPRI Report 1021556 (ADAMS ML102590092)	BWRVIP-169NP: BWR Vessel and Internals Project, Testing and Evaluation of BWR Supplemental Surveillance Program Capsules A, B, and C	Revision 0, August 2010
GE Nuclear Energy (GE-Nuclear) Report No. GE-NE-B1100716-01 (ADAMS ML12242A007)	Duane Arnold RPV Surveillance Materials and Testing Analysis	Revision 0, July 1997
EPRI Correspondence Letter (ADAMS ML18352A752)	Peach Bottom Unit 2 Surveillance Test Results Report[2]	Dec. 14, 2018
GE-Nuclear Report No. SASR 88-24 (ADAMS ML12242A122)	Peach Bottom Atomic Power Station, Unit 2 Vessel Surveillance Materials Testing and Fracture Toughness Analysis	May 1988
GE-Nuclear Report No. SASR 90-50 (ADAMS ML12242A123)	Peach Bottom Atomic Power Station, Unit 3 Vessel Surveillance Materials Testing and Fracture Toughness Analysis	June 1990
NUREG-2191, Volume 2 (GALL- SLR) Section XI, Chapter XI.M31	Reactor Vessel Materials Surveillance (GALL-SLR AMP XI.M31)	December 2017

Notes: 1.

2.

The proprietary report, as referenced in ADAMS, is addressed in the following ADAMS Accession Numbers: ML13176A096, ML13176A098, ML13176A099, and ML13176A100. A non-proprietary version of the report is available for access by members of the general public at ADAMS ML13176A097.

TheEPRIcorrespondenceletteralertstheNRCthatthePeachBottomUnit230osurveillancecapsule was removed for testing on October 22, 2018, but that submittal of the summary report for the capsule will be delayed, with a reporting date not to exceed April 30, 2020.

During the audit, the staff verified Exelon’s claim that the “preventive actions,” “parameters monitored or inspected,” “detection of aging effects,” “acceptance criteria,” and “corrective actions” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

During the audit, the staff made the following observations:

- The staff noted that in relation to the “scope of program” and “monitoring and trending” elements, Exelon’s RVMSP is an intergraded surveillance program (ISP) that is: (a) designed and implemented by the Electric Power Research Institute (EPRI) Boiling Water Reactor Vessel and Internals Project (BWRVIP), and (b) defined in BWRVIP-86, Revision 1-A. The staff also noted the version of the ISP in BWRVIP-86, Revision 1-A, only covers EPRI’s proposed implementation of the ISP and EPRI-defined surveillance capsule removals through the completion of the initial renewed operating periods. To account for this, Exelon has proposed an enhancement to the AMP (refer to Commitment #20 in SLRA UFSAR Supplement Table A.5) that calls for the applicant to pull a supplemental capsule in each unit during the subsequent period of extended operation.
- The staff reviewed information in GALL-SLR AMP XI.M31; PBAPS Site-Specific Document No. PB-PBD-AMP-XI.M31, Revision 1; EPRI Report No. BWRVIP-86, Revision 1-A; General Electric Nuclear (GE-Nuclear) Report No. SASR 88-24; UFSAR Section 4.2.6; and UFSAR Appendix Q, Section Q.1.2. The staff observed that the “monitoring and trending” element in GALL-SLR AMP XI.M31 includes the following programmatic criteria: (a) the plant-specific surveillance program or ISP will have at least one capsule that has attained or will attain a neutron fluence between one and two times the peak reactor vessel wall location neutron fluence of interest at the end of the subsequent period of extended operation, and (b) if a capsule meeting this criterion has not been tested previously, then the program includes withdrawal and testing (or alternatively the retrieval from storage, reinsertion for additional neutron fluence accumulation, if necessary, and testing) of one capsule addressing the subsequent period of extended operation.
- The staff noted that, to be consistent with these programmatic criteria, the applicant provided its lead factors and capsule removal times for the specified Unit 2 and 3 capsules that are subject to the enhancement in the enhancement tables that were included in SLRA AMP B.2.1.20. The staff also noted that the validity of the lead factor values and the proposed removal times for these capsules was supported by relevant information contained in GE-Nuclear Report No. SASR 88-24, Revision 0.

  The staff will evaluate the basis for this programmatic enhancement in the staff’s evaluation of the AMP, as provided in the final safety evaluation report for the application.

  During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects.

  The staff also audited the description of the RVMSP provided in the UFSAR supplement, Section A.2.1.20. The staff verified that the UFSAR supplement summary description for the RVMSP is consistent with the summary description provided for these types of AMPs in the Table XI-01 of the GALL-SLR Report. The staff also verified that the UFSAR supplement summary description for the

AMP includes the programmatic enhancement of the AMP defined in SLRA Section B.2.1.20 and that this enhancement is reflected in Commitment No. 20 of SLRA UFSAR Supplement Table A.5, “Second License Renewal Commitment List.”

SLRA AMP B.2.1.21, “One-Time Inspection”

Summary of Information in the Application. The SLRA states that AMP B.2.1.21, “One-Time Inspection,” is a new condition monitoring program that will be consistent with the program elements in GALL-SLR Report AMP XI.M32, “One-Time Inspection.” To verify this claim of consistency, the staff audited the SLRA AMP. At the time of the audit, Exelon had not yet fully developed the documents necessary to implement this new program, and the staff’s audit addressed only the program elements described in the applicant’s basis document. The staff will address issues identified but not resolved in this report in the SER.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-XI.M32	Program Basis Document – One-Time Inspection	Revision 2
NRC Integrated Inspection Reports 05000277/2014002 and 05000278/2014002	Peach Bottom Atomic Power Station – NRC Integrated Inspection Report 05000277/2014002 and 05000278/2014002	May 1, 2014
NRC License Renewal Inspection Report 05000277/2013007	Peach Bottom Atomic Power Station – NRC License Renewal Inspection Report 05000277/2013007	Mar 12, 2013
ER-AA-700-301	License Renewal One-Time Inspection Program	Revision 1
PB-AMPBD-OTI	DRAFT One-Time Inspection Sample Basis Document	Revision 0

During the audit, the staff verified Exelon’s claim that the “scope of program,” “preventive actions,” “parameters monitored or inspected,” “monitoring and trending,” “acceptance criteria,” and “corrective actions” program elements of the SLRA One-Time Inspection AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

In addition, the staff found that for the “detection of aging effects,” program element, sufficient information was not available to verify whether it was consistent with the corresponding program elements of the GALL-SLR Report AMP. The staff will consider issuing an RAI in order to obtain the information necessary to verify whether this program element is consistent with the corresponding program element of the GALL-SLR Report AMP. The staff will document its evaluation of this potential RAI in the SER.

During the audit, the staff made the following observations:

- The staff reviewed NRC Integrated Inspection Reports 05000277/2014002 and 05000278/2014002 and noted that, in the reports, no findings were identified.
- The staff reviewed NRC License Renewal Inspection Report 05000277/2013007 and noted that, in the report, no findings were identified.
- The staff reviewed PB-AMPBD-OTI, “One-Time Inspection Sample Basis Document” and noted that the draft document provided the plant-specific technical bases for the various sample selections used in the One-Time Inspection program at Peach Bottom.

During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER.

The staff will consider issuing an RAI to obtain the information necessary to determine whether Exelon’s SLRA AMP for One-Time Inspection can adequately manage the associated aging effects. The staff will document its evaluation of the potential RAI in the SER.
The staff also audited the description of the SLRA AMP for One-Time Inspection provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.22, “Selective Leaching”

Summary of Information in the Application. The SLRA states that AMP B.2.1.22, “Selective Leaching,” is a new program that will be consistent with the program elements in GALL-SLR Report AMP XI.M33, “Selective Leaching.” To verify this claim of consistency, the staff audited the SLRA AMP. At the time of the audit, Exelon had not yet fully developed the documents necessary to implement this new program, and the staff’s audit addressed the program elements described in the applicant’s basis document. The staff will address issues identified but not resolved in this report in the SER.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-	Program Basis Document Selective Leaching	Revision 1

XI.M33		
ER-AA-700-401	Selective Leaching Aging Management	Revision 1
AR 01501324	Lessons Learned from Turbine Building Closed Cooling Water (TBCCW) System Heat Exchanger Eddy Current Testing	04/12/2013

During the audit, the staff verified that the “scope of program,” “preventive actions,” “parameters monitored or inspected,” “detection of aging effects,” “monitoring and trending,” and “acceptance criteria” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

In addition, the staff found that for the “corrective actions” program element, sufficient information was not available to determine whether it was consistent with the corresponding program elements of the GALL-SLR Report AMP. The staff will consider issuing an RAI in order to obtain the information necessary to verify whether this program element is consistent with the corresponding program element of the GALL-SLR Report AMP. The staff will document its evaluation of this potential RAI in the SER.

During the audit, the staff made the following observations:

- The staff reviewed ER-AA-700-401 and noted that Section 4.9.4.4 states (a) the number of additional inspections is equal to the number of failed inspections for each material and environment population with a minimum of five additional visual and mechanical inspections when visual and mechanical inspections did not meet acceptance criteria, or 20 percent of each applicable material and environment combination is inspected, whichever is less, and a minimum of one additional destructive examination when destruction examinations did not meet acceptance criteria; and (b) for expanded inspections on difficult to access surfaces, such as heat exchanger tubes, industry proven technologies found capable of detecting degradation may be used as an initial indicator of the existence of imperfections. If imperfections are identified, then direct visual inspection or destructive examination should be performed to fully assess the material condition.
- The staff reviewed AR 01257959 and noted that during an initial license renewal inspection, two fire protection valves exposed to raw water showed signs of graphitic corrosion ranging in depths from approximately 0.11 to 0.33 inches. The staff also noted that the remaining wall thickness in areas showing the most severe depths of selective leaching attack ranged from approximately 0.95 to 0.98 inches.
- The staff reviewed AR 01501324, which states, “[e]ddy current testing can no longer be used to effectively predict tube leaks in the TBCCW heat exchanger. All TBCCW heat exchangers have severe tube end erosion that has resulted in eddy current testing under calling the severity of the thru wall indications.”

  During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER.

The staff also audited the description of the SLRA AMP provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.23, “ASME Code Class 1 Small-Bore Piping”

Summary of Information in the Application. The SLRA states that AMP B.2.1.23, “ASME Code Class 1 Small-Bore Piping,” is an existing condition monitoring program that will be consistent with the program elements in GALL-SLR Report AMP XI.M35, “ASME Code Class 1 Small-Bore Piping.” To verify this claim of consistency, the staff audited the SLRA AMP. During the audit, the staff reviewed the enhancement associated with this AMP. The enhancement to the GALL- SLR Report AMP is evaluated in the SER.

Audit Activities. During its audit, the staff discussed with the applicant’s staff and reviewed onsite documentation provided by the applicant.

The table below lists documents that were reviewed by the staff and were found relevant to the audit.

Relevant Documents Reviewed

Document	Title	Revision / Date
XI.M35 References Part 1	Basis for Weld Counts, Second License Renewal Project	Rev. 0
AR 04065691	Steam Leak at Weld	10/23/2017
AR 04067473	EOC Review for Failed Weld	10/26/2017
AR 00856352	Inspection of RI-ISI Piping Socket Welds	12/15/2008
AR 04078978	Maintenance Rule System 04 Recommendation	11/29/2017
AR 00479492	Maintenance Rule System 04-1-1 Performance Criteria Exceeded	11/30/2017
AR 02732688	Main Steam D Flow Instrument Lines Small-Bore Piping	10/25/2016

During the audit, the staff verified that the “scope of program,” “preventive actions,” “parameters monitored or inspected,” “detection of aging effects,” “monitoring and trending,” and “acceptance criteria” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects.

The staff also audited the description of the SLRA AMP provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SRP Report.

SLRA AMP B.2.1.24, “External Surfaces Monitoring of Mechanical Components”

Summary of Information in the Application. The SLRA states that AMP B.2.1.24, “External Surfaces Monitoring of Mechanical Components,” is a new program that will be consistent with the program elements in GALL-SLR Report AMP XI.M36, “External Surfaces Monitoring of Mechanical Components.” To verify this claim of consistency, the staff audited the SLRA AMP. During the audit, the staff reviewed the exception to the GALL-SLR Report AMP and the enhancement associated with this AMP. The staff will document its reviews of the exception and the enhancement in the SER.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The table below lists the documents that were reviewed by the staff and were found to be relevant to the audit.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-XI.M36	Program Basis Document – External Surfaces Monitoring of Mechanical Components	Revision 1
ER-AA-335-1005	Standard Approach on How to Evaluate and Inspect Outside Diameter Corrosion on Piping	Revision 4
ER-AA-700-402	External Surfaces Monitoring of Mechanical Components AMP	Revision 1
ER-AA-2030	Conduct of Plant Engineering	Revision 18

During the audit, the staff verified that the “scope of program,” “preventive actions,” “parameters monitored or inspected,” “detection of aging effects,” “monitoring and trending,” “acceptance criteria,” and “corrective actions” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

During the audit of the “operating experience” program element, the staff conducted an independent search of the plant-specific operating experience database as discussed in the operating experience audit report. The staff will evaluate the identified plant-specific operating experience in the SER.

The staff also audited the description of the External Surfaces Monitoring of Mechanical Components program provided in SLRA Section A.2.1.24. The staff verified that it is consistent with the description provided in the GALL-SLR Report Table XI-01.

SLRA AMP B.2.1.25, “Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components”

Summary of Information in the Application. The SLRA states that AMP B.2.1.25, “Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components,” is a new program that will be consistent with the program elements in GALL-SLR Report AMP XI.M38, “Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components.” To verify this claim of consistency, the staff audited the SLRA AMP. At the time of the audit, Exelon had not yet fully developed the documents necessary to implement this new program, and the staff’s audit addressed only the program elements described in the applicant’s basis document. The staff will address issues identified but not resolved in this report in the SER.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-XI.M38	Program Basis Document Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components	Revision 1
ER-AA-700-403	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Aging Management Program	Revision 0
PB-AMRBD-MEAE	Materials, Environments, and Aging Effects Aging Management Review Basis Document	Revision 2

During the audit, the staff verified that the “scope of program,” “preventive actions,” “monitoring and trending,” “acceptance criteria,” and “corrective actions” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

In addition, the staff found that for the “parameters monitored or inspected” and “detection of aging effects” program elements, sufficient information was not available to determine whether they were consistent with the corresponding program elements of the GALL-SLR Report AMP. The staff will consider issuing RAIs in order to obtain the information necessary to verify whether these program elements are consistent with the corresponding program elements of the GALL-SLR Report AMP. The staff will document its evaluation of these potential RAIs in the SER.

During the audit, the staff made the following observations:

- The staff reviewed ER-AA-700-403 and noted that Section 4.7.3.2 states the number of additional inspections to be performed, if a component does not meet acceptance criteria, is no fewer than five additional inspections for each inspection that did not meet acceptance criteria, or 20 percent of each applicable material, environment, and aging effect combination, whichever is less.
- The staff reviewed PB-AMRBD-MEAE and noted that (a) Section 4.3.12, “PBAPS Internal and External Environment Summary,” states that raw water (potable) has been filtered and chlorinated and is therefore not susceptible to MIC; and (b) Section 4.5.1, “Treated Water (EPRI Mechanical Tools Appendix A),” states that MIC is only a potential aging mechanism for treated water where contamination with microbes has occurred.

  During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER.

  The staff also audited the description of the SLRA AMP provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.26, “Lubricating Oil Analysis”

Summary of Information in the Application. The SLRA states that AMP B.2.1.26, Lubricating Oil Analysis,” is an existing program that is consistent with the program elements in GALL-SLR Report AMP XI.M39, “Lubricating Oil Analysis.” To verify this claim of consistency, the staff audited the SLRA AMP.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP- XI.M39	Lubricating Oil Analysis	Revision 1
MA-AA-716-006	Control of Lubricants Program	Revision 14

MA-AA716-230	Predictive Maintenance Program	Revision 11
MA-AA-716-230- 1001	Oil Analysis Interpretation Guideline	Revision 20

During the audit, the staff verified that the “scope of program,” “preventive actions,” “parameters monitored or inspected,” “detection of aging effects,” “monitoring and trending,” “acceptance criteria,” and “corrective actions” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER.

The staff also audited the description of the SLRA AMP Lubricating Oil Analysis provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.27, “Monitoring of Neutron-Absorbing Materials Other Than Boraflex”

Summary of Information in the Application. The SLRA states that AMP XI.M40, “Monitoring of Neutron-Absorbing Materials Other Than Boraflex,” is an existing program that is consistent with the program elements in GALL-SLR Report AMP XI.M40, “Monitoring of Neutron-Absorbing Materials Other Than Boraflex.” To verify this claim of consistency, the staff audited the SLRA AMP.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
N/A	Peach Bottom Atomic Power Station Units 2 and 3 Updated Final Safety Analysis Report (UFSAR)	Revision 26
PB-PBD-AMP-XI.M40	Program Basis Document – Monitoring of Neutron-Absorbing Materials Other Than Boraflex	Revision 0
RT-R-004-971-2	Two-Year Surveillance Program for Netco Snap-In Alcan Neutron Absorbing Material, for the first Ten Year Interval (Unit 2)	Revision 2
RT-R-004-971-3	Two-Year Surveillance Program for Netco Snap-In Alcan Neutron Absorbing Material, for the first Ten Year Interval (Unit 3)	Revision 2

RT-R-004-972-2	Ten Year Surveillance Program for NETCO Snap- In Alcan Neutron Absorbing Material (Unit 2)	Revision 0
RT-R-004-972-3	Ten Year Surveillance Program for NETCO Snap- In Alcan Neutron Absorbing Material (Unit 2)	Revision 0
NF-AA-610	On-Site Wet Storage of Spent Nuclear Fuel	Revision 15
N/A	Peach Bottom Atomic Power Station Units 2 and 3 Updated Safety Analysis Report (UFSAR)	Revision 26

During the audit, the staff verified that the “scope of program,” “preventive actions,” “parameters monitored or inspected,” “detection of aging effects,” “monitoring and trending,” “acceptance criteria,” and “corrective actions” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

During the audit, the staff made the following observation:

 The staff reviewed fleet procedure NF-AA-610 and noted that there is a procedural requirement to trend coupon test results if projected degradation of the neutron absorbing material is unable to maintain the required 5 percent sub-criticality margin. This procedural requirement is “[f]or stations that had their renewed licenses approved to GALL Report Revision 2...”; however, it was not clear whether this would also apply to plants licensed under GALL-SLR.

During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER.

The staff also audited the description of the SLRA AMP/TLAA title provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.28, “Buried and Underground Piping and Tanks”

Summary of Information in the Application. The SLRA states that AMP B.2.1.28, “Buried and Underground Piping and Tanks,” is an existing program with enhancements that will be consistent with the program elements in GALL-SLR Report AMP XI.M41, “Buried and Underground Piping and Tanks.” To verify this claim of consistency, the staff audited the SLRA AMP. Issues identified but not resolved in this report will be addressed in the SER. During the audit, the staff reviewed the enhancements associated with this AMP. The staff will document its review of the enhancements in the SER.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-XI.M41	Buried and Underground Piping and Tanks Program Basis Document	Revision 2
EC-622830	2018 PBAPS (Peach Bottom Atomic Power Station) Cathodic Protection Improvements	04/25/2018
RT-O-57F-910-2	Cathodic Protection System Inspection	02/28/2018
RT-O-57F-910-2	Cathodic Protection System Inspection	01/29/2018
6280-C-16	Specification for Installation of Underground Piping for the PBAPS Units 2 and 3 Philadelphia Electric Company	07/31/1968
6280-C-28	Specification for Underground Tanks for the PBAPS Units 2 and 3 for the Philadelphia Electric Company	10/08/1969
6280-M-306	Specification for External Surface Treatment of Underground Metallic Pipe for the PBAPS Units 2 and 3 for the Philadelphia Electric Company	07/18/1968
ER-AA-5400-1002	Underground Piping and Tank Examination Guide	Revision 8

During the audit, the staff verified that the “scope of program,” “monitoring and trending,” and “corrective actions” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

In addition, the staff found that for the “preventive actions,” “parameters monitored or inspected,” “detection of aging effects,” and “acceptance criteria” program elements, sufficient information was not available to determine whether they were consistent with the corresponding

During

the audit, the staff made the following observations:

program elements of the GALL-SLR Report AMP. The staff will consider issuing RAIs in order to obtain the information necessary to verify whether these program elements are consistent with the corresponding program elements of the GALL-SLR Report AMP. The staff will document its evaluation of these potential RAIs in the SER.

- The staff reviewed PB-PBD-AMP-XI.M41 and noted that (a) buried stainless steel piping is coated with either a coal tar based somastic coating or a coal tar enamel with felt wrap coating (with the exception of the 10-inch diameter stainless steel line from the torus dewatering tank to the condensate transfer pump suction line); (b) buried and underground carbon steel piping and tanks are coated with either a coal tar based somastic coating or a coal tar enamel with felt wrap coating; (c) the emergency diesel generator fuel oil tanks are coated with coal tar based bituminous coating; (d) original design specifications specified that bedding material be installed within six inches of buried steel and stainless steel coated pipe and comprised of sound well graded granular material with aggregate size less than 3/8-inch; and (e) soil samples have shown relatively low levels of chlorides (less than 15 ppb (parts per billion)).

- The staff reviewed EC-622830, Attachment 1, Appendix I, and noted that the results of twenty soil corrosivity samples show that (a) soil resistivity ranged from 3,000 to 145,000 ohm-cm with an average value of 40,521 ohm-cm; (b) oxygen reduction values ranged from 263 to 390 millivolts; (c) none of the samples had detectable sulfides; (d) soil pH ranged from 7.1 to 9.8; (e) all observed samples were moist to wet; (f) anaerobic sulfate reducing bacteria were identified in thirteen of twenty tested samples; and (g) chlorides ranged from approximately one to 100 ppm (parts per million).
- The staff reviewed AR 04055916 and noted that (a) the buried 6 to 8 foot portion of the 10-inch diameter stainless steel cross-tie line between the torus dewatering tank and the Unit 3 condensate storage tank was found to be uncoated; (b) one to 2wo feet of the subject piping is exposed to native fill; (c) five to six feet of the subject piping is exposed to cementitious fill; and (d) in order to meet subsequent license requirements, the subject piping should be coated 10 years prior to the subsequent period of extended operation.
- The staff reviewed AR 01137854 and noted that a pinhole leak was identified on top of a weld on the 20-inch emergency service water (ESW) supply buried piping. The leak was due to internal corrosion as the external surfaces of the piping did not show signs of external corrosion.
- The staff reviewed AR 01255154 and noted that during buried piping inspections the most severe external corrosion was 3/32-inch, which did not threaten the minimal wall thickness of 0.245-inch for 0.5-inch nominal wall thickness pipe.
- The staff reviewed AR 02513031 and noted that during buried fire protection piping inspections (a) no external corrosion was identified; and (b) as found coatings were well bonded to the piping.
- The staff reviewed RT-O-57F-910-2 (both January and February 2018) and noted that 15 out of 16 rectifiers met voltage and current limit acceptance criteria.
- The staff reviewed ER-AA-5400-1002 and noted that specific details on the installation and use of electrical resistance corrosion rate probes will be in accordance with the vendor, manufacturer, and NACE qualified cathodic protection expert (i.e., NACE CP4, “Cathodic Protection Specialist” qualification) recommendations.

During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER.

The staff also audited the description of the SLRA AMP provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.29, “Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks”

Summary of Information in the Application. SLRA states that AMP B.2.1.29, “Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks,” is a new program with exceptions that will be consistent with the program elements in GALL-SLR Report AMP XI.M42, “Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks.” To verify this claim of consistency, the staff audited the SLRA AMP. At the time of the audit, Exelon had not yet fully developed the documents necessary to implement this new program, and the staff’s audit addressed only the program elements described in the applicant’s basis document. Issues identified but not resolved in this report will be addressed in the SER. During the audit, the staff reviewed the exceptions associated with this AMP. The staff will document its review of the exceptions to the GALL-SLR Report AMP in the SER.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-XI.M42	Program Basis Document Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks	Revision 2
ER-AA-330-014	Exelon Safety-Related (Service Level III) Coatings	Revision 2
M-010-002	Residual Heat Removal (RHR) Heat Exchanger Maintenance	Revision 17
M-C-756-001	High Pressure Coolant Injection (HPCI) Turbine Inspection	Revision 32

During the audit, the staff verified that the “preventive actions,” “parameters monitored or inspected,” “monitoring and trending,” and “acceptance criteria” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

The staff also verified that aspects of the “detection of aging effects” and “corrective actions” program elements not associated with the exceptions identified in the SLRA or by the staff during the audit are consistent with the corresponding program elements in the GALL-SLR Report AMP.

In addition, the staff found that for the “scope of program” program element, sufficient information was not available to determine whether it was consistent with the corresponding program elements of the GALL-SLR Report AMP. The staff will consider issuing an RAI in order to obtain the information necessary to verify whether this program element is consistent with the corresponding program element of the GALL-SLR Report AMP. The staff will document its evaluation of this potential RAI in the SER.

During the audit, the staff made the following observations:

- The staff reviewed AR 04049466 and noted that during development of the Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks aging management program, it was identified that the high-pressure service water side of the RHR heat exchanger water box did not appear to be coated.
- The staff reviewed ER-AA-330-014 Section 4.8.5, “Service Level (III) ISG Periodic Inspection Requirements,” which states “[t]he training and qualification of individuals involved in coating inspections and evaluating degraded conditions is conducted in accordance with an ASTM International standard endorsed in RG 1.54 (such as ASTM D7167-05) including staff guidance associated with a particular standard. For cementitious coatings/linings inspectors should have a minimum of 5 years of experience inspecting or testing concrete structures or cementitious coatings/linings or a degree in the civil/structural discipline and a minimum of 1 year of experience.”

  During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER. The staff will consider issuing an RAI in order to obtain the information necessary to determine whether Exelon’s SLRA AMP can be adequate to manage the associated aging effects. The staff will document its evaluation of the potential RAI in the SER.

  The staff also audited the description of the SLRA AMP provided in the UFSAR supplement. The staff found that sufficient information was not available to determine whether the description provided in the UFSAR supplement was an adequate description of the SLRA AMP. The staff will consider issuing an RAI in order to obtain the information necessary to verify the sufficiency of the UFSAR supplement program description. The staff will document its evaluation of the potential RAI in the SER.

  SLRA AMP B.2.1.30, “ASME Section XI, Subsection IWE”

  Summary of Information in the Application. The SLRA states that AMP B.2.1.30, “ASME Section XI, Subsection IWE,” is an existing program, with enhancements and exception, that will be consistent with the program elements in GALL-SLR Report AMP XI.S1, “ASME Section XI, Subsection IWE.” To verify this claim of consistency, the staff audited the SLRA AMP. Issues identified but not resolved in this report will be addressed in the SER. During the audit, the staff reviewed the exception and enhancements associated with this AMP. The staff will document its review of the exceptions to the GALL-SLR Report AMP and the enhancements in the SER.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-XI.S1	Program Basis Document: ASME Section XI, Subsection IWE	Revision 1
ER-AA-330	Conduct of Inservice Inspection Activities	Revision 13
ER-AA-330-007	Visual Examination of Section XI Class MC Surfaces and Class CC Liners	Revision 11
ER-AA-330-009	ASME Section XI Repair/Replacement Program	Revision 13
CC-AA-102	Design Input and Configuration Change Impact Screening	Revision 30
CC-MA-102-1001	Design Input and Impact Screening: Implementation	Revision 14
MA-AA-736-600	Torqueing and Tightening of Bolted Connections	Revision 5
PES-S-010	Standard: Fasteners	Revision 0 (Rev 1 markup)
FPSA-02	Fastener Procurement Standard for ASME Section III Fasteners	Revision 0
FPSB-02	Fastener Procurement Standard for ASTM Safety Related Fasteners	Revision 0
FPSD-02	Fastener Procurement Standard for Dedicated Safety Related Fasteners	Revision 0
ST-M-007-900-2 & ST-M-007- 900-3	Drywell Airgap Drains Flow Test (Once/Operating Cycle) – Peach Bottom Unit 2/3 Surveillance Test (verifies drywell airgap drain liners are clear)	Revision 2
PBT05.G06	Augmented Inspection Plan – Fourth Ten-Year Inspection Interval, PBAPS Unit 2 and Unit 3: Augmented Containment Inspection Program No. AUG-C3 - Monitor Sludge Accumulation on Torus Floor	Revision 5
EXLNPB113-REPT-001	Review of Containment Fatigue Analyses for Peach Bottom Second License Renewal (Non-Safety- Related)	Revision 0, 12/7/2016

Document	Title	

		Revision / Date
1400630.304	Fatigue Exemption of the Peach Bottom Drywell (including mechanical and electrical penetrations) – Structural Integrity Associates Calculation	Revision 0, 10/16/2017
GEH-7480-316805-HE2-ISI	Peach Bottom Atomic Power Station – P3R19 (Sept 2013) ISI Final Report	10/16/2013
7480-191304- HE2-ISI	Peach Bottom Atomic Power Station – P2R21 (Oct 2016) ISI Final Report	October 2016
GEH-REPORT-PB-ISI-14- 183901	Peach Bottom Atomic Power Station – P2R20 (Nov 2014) ISI Final Report	November 2014
7480-189821- HE3-ISI	Peach Bottom Atomic Power Station – P3R20 (Oct 2015) ISI Final Report	10/16/2015
RCN-043	Sept-Oct 2015 P3R20 Torus Project: Underwater Cleaning, Coating Inspection and Repair, Peach Bottom Unit 3, Underwater Construction Corporation	November 2015
RCN-036	Oct-Nov 2014 P2R20 Torus Project: Underwater Cleaning, Coating Inspection and Repair, Peach Bottom Unit 3, Underwater Construction Corporation	December 2014
MA-PB-793-001	Visual Examination of Containment Vessels and Internals	Revision 3
RT-M-007-901-2 & RT-M-007- 901-3	Debris Loading Measurement and Computation in Torus	Revision 2

During the audit, the staff verified that for the program elements that Exelon declared were consistent, the “scope of program,” “monitoring and trending,” “acceptance criteria,” and “corrective actions” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

The staff also verified Exelon’s claim that aspects of the “parameters monitored or inspected” program element not associated with the exception identified by Exelon are consistent with the corresponding program element in the GALL-SLR Report AMP.

In addition, the staff found that for the “preventive actions” and “detection of aging effects” program elements, sufficient information was not available to determine whether they were consistent with the corresponding program elements of the GALL-SLR Report AMP. The staff will use the voluntary SLRA supplement information committed to by Exelon during the audit, or the staff will consider issuing RAIs in order to obtain the information necessary to verify whether these program elements are consistent with the corresponding program elements of the GALL- SLR Report AMP. The staff will document its evaluation of the supplemental information or potential RAIs in the SER.

During the audit, the staff noted that the SLRA has credited the ASME Section XI, Subsection IWE AMP to manage flow blockage due to fouling for the stainless steel ECCS suction strainers exposed to treated water in the torus. This component, material, environment, and aging effect/mechanism combination is not included in the GALL-SLR Report. The staff will use voluntary SLRA supplement information provided by Exelon or consider issuing an RAI to assess the capability of the AMP for aging management of this component, material, environment and aging effect/mechanism combination. The staff will document its evaluation of the supplemental information or potential RAI in the SER.

During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER.

The staff also audited the description of the SLRA AMP “ASME Section XI, Subsection IWE,” provided in the UFSAR supplement. The staff found that sufficient information was not available to determine whether the description provided in the UFSAR supplement was an adequate description of the SLRA AMP “ASME Section XI, Subsection IWE.” The staff will use voluntary SLRA supplement information provided by Exelon or consider issuing an RAI in order to obtain the information necessary to verify the sufficiency of the UFSAR supplement program description. The staff will document its evaluation of the supplemental information or potential RAI in the SER.

SLRA AMP B.2.1.31, “ASME Section XI, Subsection IWF”

Summary of Information in the Application. The SLRA states that AMP B.2.1.31, “ASME Section XI, Subsection IWF,” is an existing program with enhancements that will be consistent with the program elements in GALL-SLR Report AMP XI.S3, “ASME Section XI, Subsection IWF.” To verify this claim of consistency, the staff audited the SLRA AMP. Issues identified but not resolved in this report will be addressed in the SER. During the audit, the staff reviewed the enhancements associated with this AMP. The staff will document its review of the enhancements in the SER.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
ER-AA-330	Conduct of Inservice Inspection Activities	Revision 013
ER-AA-330-003	Inservice Inspection of Section XI Component Supports	Revision 013
ER-AA-335-016	VT-3 Visual Examination of Component Supports, Attachments, and Interiors of Reactor Vessels	Revision 10
ER-AA-330	ISI Classification Basis Document, Fourth Ten-Year Inspection Interval	09/02/2014
PB-PBD-AMP-XI.S3	Program Basis Document – ASME Section XI, Subsection IWF	Revision 1
M-3403-1, -2, and -3	Drawing M-3403-1, -2, and -3	N/A

Document	Title	Revision / Date
ER-AA-330-009	ASME Section XI Repair/Replacement Program	Revision 14
ER-AA-335-016	VT-3 Visual Examination of Component Supports, Attachments, and Interiors of Reactor Vessels	Revision 11
N/A	ISI Program Plan Fourth Ten-Year Inspection Interval	09/16/2014

During the audit, the staff verified that for the program elements that Exelon declared were consistent, the “scope of program,” “parameters monitored or inspected,” “acceptance criteria,” and “corrective actions” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP. In addition, the staff found that for the “preventive actions,” “detection of aging effects,” and “monitoring and trending,” program elements sufficient information was not available to determine whether they were consistent with the corresponding program elements of the GALL-SLR Report AMP. The staff will consider issuing RAIs in order to obtain the information necessary to verify whether these program elements are consistent with the corresponding program elements of the GALL-SLR Report AMP. The staff will document its evaluation of these potential RAIs in the SER.

During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER.

The staff also audited the description of the SLRA AMP “ASME Section XI, Subsection IWF” provided in the UFSAR supplement. The staff found that sufficient information was not available to determine whether the description provided in the UFSAR supplement was an adequate description of the SLRA AMP “ASME Section XI, Subsection IWF.” The staff will consider issuing an RAI in order to obtain the information necessary to verify the sufficiency of the UFSAR supplement program description. The staff will document its evaluation of these potential RAIs in the SER.

SLRA AMP B.2.1.32, “10 CFR Part 50, Appendix J”

Summary of Information in the Application. The Peach Bottom Atomic Power Station (PBAPS) Units 2 and 3 SLRA states that AMP B.2.1.32, “10 CFR Part 50, Appendix J,” is an existing program that is consistent with the program elements in GALL-SLR Report AMP XI.S4, “10 CFR Part 50, Appendix J.” To verify this claim of consistency, the staff audited the SLRA AMP.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP XI.S4	10 CFR Part 50, Appendix J Peach Bottom Atomic Power Station, Second License Renewal Project (AMP Basis Document)	Revision 2

Document	Title	Revision / Date
PBAPS UFSAR	Sections 4.6 and 5.2, Main Steam Line Isolation (MSIV) Valves, Primary Containment (respectively)	Revision 26
PBAPS Technical Specifications (TS)	Sections 3.6 and 5.5.12, Containment Systems, Primary Containment Leak Rate Testing Program (respectively)	N/A*
Annual, U2 & U3	PBAPS; Appendix J Program Health Report; Control Doc.: ER-AA-380	2016
1st Tri-Annual, U2 & U3	PBAPS; Appendix J Program Health Report; Control Doc.: ER-AA-380	2015
2nd Tri-Annual, U2 & U3	PBAPS; Appendix J Program Health Report; Control Doc.: ER-AA-380	2014
2nd Tri-Annual, U2 & U3	PBAPS; Appendix J Program Health Report; Control Doc.: ER-AA-380	2015
3rd Tri-Annual, U2 & U3	PBAPS; Appendix J Program Health Report; Control Doc.: ER-AA-380	2014
ST-J-07A-600-2	R1174669-2014 ILRT Excerpts	Revision 8
ST-J-07A-600-3	R0333868-2015 ILRT Excerpts	Revision 4
ECR 15-00314	LLRT Scope Reduction (RHR/Low Pressure Coolant Injection, Core Spray, and Standby Liquid Control Systems)	09/18/201
N/A*	Exclusions from LRT (Penetrations/associated Components: N-12; N-13A/B; N-16A/B; N-35E, F, G; N-37A-D; N-38A-D; N-39A/B, N-210A/B, N-211A/B, N- 212, N-213A/B, N-214, N-216, N-221, N-223, N-224 U2, N-226A-D, N-227, N-228A-D, N-229 U2, N-230, N-233 U2, N-234 U2, N-234A/B U3, N-235 U3, N-236A/B u3)	N/A*
ML15196A559	Peach Bottom Atomic Power Station, Units 2 and 3 – Issuance of Amendments; Re: Extension of Type A and Type C LRT Frequencies	09/08/2015
ECR 16-00346	MSIV Poppet Skirt Modification	10/31/2016

ER-AA-380	Primary Containment LRT Program (Implementing Document)	Revision 11
2nd Half	Appendix J Program Health Metric	2017

Document	Title	Revision / Date
XI.S4 Industry OpE	PBAPS U2, U3 License Renewal Project: 10 CFR 50, Appendix J AMP	N/A*
ER-AA-380-1002**	PBAPS U2, U3 License Renewal Project: 10 CFR 50, Appendix J AMP	N/A*
R1003365**	Integrated Leakage Rate Test – Planning and Implementation Guide	Revision 4
MA-AA-716-017**	Replace Resilient Parts (Plunger, Disc, & "O" Ring)	04/23/2015
S-188**	Station Rework Reduction Program	Revision 8
PBAPS SLRA	Drywell Vessel Pour Sequence	1975
	Section B.2.1.32 and Sections B.2.1.1, B.2.1.2, B.2.1.5, B.2.1.9, B.2.1.14, B.2.1.21, B.2.1.23, B.2.1.24, B.2.1.25, B.2.1.30	Revision 0

*N/A not available
**Requested by Staff following the OE Audit

During the audit, the staff verified that the "scope of program," "preventive actions," "parameters monitored or inspected," "detection of aging effects," "monitoring and trending," "acceptance criteria," and "corrective actions" program element(s) of the SLRA AMP are consistent with the corresponding element(s) of the GALL-SLR Report AMP.

During the audit, the staff made the following observations:

- The staff reviewed PBAPS SLRA and confirmed that Sections B.2.1.1, B.2.1.2, B.2.1.5, B.2.1.9, B.2.1.14, B.2.1.21, B.2.1.23, B.2.1.24, B.2.1.25, B.2.1.30 referenced by PBAPS SLRA Section B.2.1.32 are also listed in PB-PBD-AMP XI.S4, planned to be used as the relevant Aging Management Programs (AMPs) to manage the effects of aging for the components excluded from 10 CFR Part 50 Appendix J, local leakage rate tests (LLRTs). The excluded components are identified in UFSAR Table 5.2.2, "Containment Penetrations, Compliance with 10 CFR50, Appendix J." The staff's individual AMR line item audit reviews, based on listings in Table 2 system sections and associated Table 1 references for components excluded from the 10 CFR Part 50, Appendix J LLRTs, are documented in the appropriate "In Office Audit Report" sections.

During the audit of the "operating experience" program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER.

The staff also audited the description of the SLRA AMP 10 CFR Part 50, Appendix J provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.33, “Masonry Walls”

Summary of Information in the Application. SLRA states that AMP B.2.1.33, “Masonry Walls,” is an existing condition monitoring program with enhancements that will be consistent with the program elements in GALL-SLR Report AMP XI.S5, “Masonry Walls.” To verify this claim of consistency, the staff audited the SLRA AMP.

During the audit, the staff reviewed the enhancements associated with this AMP. The staff will document its review of the enhancements in the SER.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-XI.S5	Aging Management Program Basis Document – Masonry Walls	Revision 1 8/2/2017
ER-AA-450	Structures Monitoring	Revision 6
ER-PB-450	Peach Bottom Structures Monitoring	Revision 0
ER-PB-450-1006	Peach Bottom Structures Monitoring Instructions	Revision 4
AR 02657801	Large cracks in masonry wall TB 3 135 ELEV	4/19/2016
AR 02657343	Large cracks in the floor and walls	4/18/2016
AR 04134239-03	Revise ER-AA-450 Section 6.1.5	11/28/2018

During the audit, the staff verified that for the program elements that Exelon declared were consistent, the “scope of program,” “preventive actions,” “parameters monitored or inspected,” “detection of aging effects,” “monitoring and trending,” “acceptance criteria,” and “corrective actions” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER.

The staff also audited the description of the SLRA AMP provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.34, Structures Monitoring

Summary of Information in the Application. The SLRA states that AMP B.2.1.34, “Structures Monitoring,” is an existing program with enhancements that will be consistent with the program elements in GALL-SLR Report AMP XI.S6, “Structures Monitoring.” To verify this claim of consistency, the staff audited the SLRA AMP. Issues identified but not resolved in this report will be addressed in the SER. During the audit, the staff reviewed the enhancements associated with this AMP. The staff will document its review of the enhancements in the SER.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-XI.S6	Program Basis Document: Structures Monitoring	Revision 1
ER-PB-450	Peach Bottom Structures Monitoring Program (New)	Revision 0
ER-PB-450-1006	Peach Bottom Structures Monitoring Instruction	Revision 4
ER-PB-716-1000	Control of Bolting/Torqueing/Tensioning	Revision 0
PES-S-003	In-Storage Maintenance of Nuclear Material	Revision 10
Specification C-41	Structural Steel	Revision 0
ER-AA-450	Structures Monitoring	Revision 6
SA-AA-117	Excavation, Training, Shoring	Revision 21
MA-AA-736-600	Torqueing and Tightening of Bolted Connections	Revision 8
P-T-01	Structural: Design Baseline Document (incl. Sec. 3.3)	Revision 9
Dwg. Figure 1	Groundwater Monitoring Locations	February 2010
17D0989	Report of Groundwater Sampler Spring 2017	04/27/17
17L0736	Report of Groundwater Sampler Winter 2017	12/27/17
18B1256	Report of Groundwater Sampler Early 2018	03/09/18
UFSAR	PBAPS Unit 2 and 3 Updated Safety Analysis Report	Revision 26 April 2017

During the audit, the staff verified that, for the program elements that Exelon declared were consistent, the “parameters monitored or inspected,” “monitoring and trending,” “acceptance criteria,” and “corrective actions” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP. In addition, the staff found that, for the “scope of program,” “preventive actions,” and “detection of aging effects” program elements, sufficient information was not available to determine whether they were consistent with the corresponding program elements of the GALL-SLR Report AMP. The staff will consider issuing RAIs in order to obtain the information necessary to verify whether these program elements are consistent with the corresponding program elements of the GALL-SLR Report AMP. The staff will document its evaluation of these potential RAIs in the SER.

During the audit the staff made the following observation:

- The staff reviewed Report No(s). 17L0736, 17D0989 and 18B1256, and noted that several monitoring wells have recorded chlorides level above the GALL-SLR Report threshold for aggressive groundwater/soil; thus, structures near these locations may be exposed to a non-seasonal aggressive groundwater/soil environment. The staff will consider issuing an RAI and document its evaluation in the SER.

During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER.

The staff also audited the description of the SLRA Structures Monitoring Program provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.35, “Inspection of Water Control Structures Associated with Nuclear Power Plants”

Summary of Information in the Application. The SLRA states that AMP B.2.1.5, “Inspection of Water Control Structures Associated with Nuclear Power Plants,” is an existing program with enhancements. To verify this claim of consistency, the staff audited the SLRA AMP. Issues identified but not resolved in this report will be addressed in the SER. During the audit, the staff reviewed the enhancements associated with this AMP. The enhancements are evaluated in the SER.

Audit Activities. During its audit, the staff reviewed onsite documentation provided by the applicant. The table below lists the documents that were reviewed by the staff and were found relevant to the audit.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-XI.S7	GALL-SLR Program XI.S7- Inspection of Water Control Structures Associated with Nuclear Power Plants	Revision 1 06/19/2018
Inspection Report	Conowingo –FERC Dam Inspection Report	03/08/2016

During the audit, the staff verified that the “scope of program,” “preventive actions,” “parameters monitored or inspected,” “detection of aging effects,” “monitoring and trending,” and “acceptance criteria” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff’s evaluation of the identified plant-specific operating experience will be addressed in the SER. In light of the plant-specific operating experience, in order to obtain the information necessary to determine whether the applicant’s SLRA AMP can be adequate to manage the associated aging effects, the staff will consider issuing an RAI. The staff’s evaluation of the potential RAI will be documented in the Safety Evaluation Report.

The staff also audited the description of the SLRA AMP provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.36, “Protective Coating Monitoring and Maintenance”

Summary of Information in the Application. The SLRA states that AMP XI.S8, “Protective Coating Monitoring and Maintenance,” is an existing program with an enhancement that will be consistent with the program elements in GALL-SLR Report AMP XI.S8, “Protective Coating Monitoring and Maintenance.” To verify this claim of consistency, the staff audited the SLRA AMP. Issues identified but not resolved in this report will be addressed in the SER. During the audit, the staff reviewed an enhancement associated with this AMP. The staff will document its review of the enhancement in the SER.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-XI.S8	Program Basis Document - Protective Coating Monitoring and Maintenance	Revision 1
CC-AA-205	Control of Undocumented/Unqualified Coatings Inside the Containment	Revision 10
ER-AA-330-008	Exelon Safety-Related (Service Level I) Protective Coatings	Revision 12
MA-PB-793-001	Visual Examination of Containment Vessels and Internals	Revision 3
NE-00047	Specification for Torus Underwater Inspection and Repair at Peach Bottom Atomic Power Station	Revision 7
PMRQ	20S019: Torus Dewatering/Cleaning/Inspection	N/A

234247-01		
PMRQ 234248-01	30S019: Torus Dewatering/Cleaning/Inspection	N/A
ST-N-080-900-2	Visual Examination of Drywell and Torus Surfaces	Revision 4

Document	Title	Revision / Date
ST-N-080-900-3	Visual Examination of Drywell and Torus Surfaces	Revision 4
ER-AA-330-007	Visual Examination of Section XI Class MC Surfaces and Class CC Liners	Revision 11
ER-AA-335-018	Visual Examination of ASME IWE Class MC and Metallic Liners of IWL Class CC Components	Revision 12
ER-AA-330-007	Visual Examination of Section XI Class MC Surfaces and Class CC Liners	Revision 11

During the audit, the staff verified that the “scope of program,” “preventive actions,” “parameters monitored or inspected,” “acceptance criteria,” and “corrective actions” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

In addition, the staff found that for the “detection of aging effects,” “monitoring and trending,” program elements sufficient information was not available to determine whether they were consistent with the corresponding program elements of the GALL-SLR Report AMP. The staff will consider issuing RAIs in order to obtain the information necessary to verify whether these program elements are consistent with the corresponding program elements of the GALL-SLR Report AMP. The staff will document its evaluation of these potential RAIs in the SER.

During the audit, the staff made the following observations:

- The staff reviewed AR 2413128 and noted that degraded coating along the “belly band” region of the torus extends from 1 inch above to 6 inches into the waterline.
- The staff reviewed AR 1691387 and noted that the measured total organic carbon (TOC) in the Unit 2 torus water had increased after re-coating the Unit 2 torus. The staff also noted that the suspected cause for the rise in TOC was the curing agent used to apply the new coating to the Unit 2 torus.

- The staff reviewed AR 1192421 and noted that the main steam safety relief valve (MSSRV) discharge temperature is greater than the coatings qualified temperature. If the MSSRVs lift, they could result in approximately 100 additional pounds of unqualified coatings in containment.
- The staff reviewed PMID RQ 234247-01 and noted that the inspection frequency for coatings in the torus are at least every 4 years/2 refueling outages for above and below the waterline.
- The staff reviewed the proposed UFSAR supplement in the SLRA, and noted that it did not state the program would be based on Regulatory Guide 1.54, "Service Level I, II, III, and In-Scope License Renewal Protective Coatings Applied to Nuclear Power Plants."
- The staff reviewed the proposed enhancement to the program and noted that it did not specify the standard to which coatings inspection personnel will be certified.

During the audit of the "operating experience" program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER. The staff will consider issuing an RAI in order to obtain the information necessary to determine whether Exelon's SLRA Protective Coating Monitoring and Maintenance program can be

adequate to manage the associated aging effects. The staff will document its evaluation of the potential RAI in the SER.

The staff also audited the description of the SLRA Protective Coatings Monitoring and Maintenance program provided in the UFSAR supplement. The staff found that sufficient information was not available to determine whether the description provided in the UFSAR supplement was an adequate description of the SLRA Protective Coating Monitoring and Maintenance program. The staff will consider issuing an RAI in order to obtain the information necessary to verify the sufficiency of the UFSAR supplement program description. The staff will document its evaluation of the potential RAI in the SER.

SLRA AMP B.2.1.37, "Electrical Insulation for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements"

Summary of Information in the Application. The SLRA states that AMP B.2.1.37, "Electrical Insulation for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements" is an existing program with enhancements that will be consistent with the program elements in GALL-SLR Report AMP XI.E1, "Electrical Insulation for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements." In addition, the SLRA stated that no exceptions were taken to the GALL-SLR Report AMP XI.E1. To verify this claim of consistency, the staff audited the SLRA AMP.

During the audit, the staff reviewed the enhancements associated with this AMP. The staff will document its review of the enhancements in the SER.

Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-XI.E1	Electrical Insulation for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Revision 1/July 6, 2018
XI.E1 Plant OpE		August 10, 2018
IEPSON Report No. NE-11-32-1	Cable and Connection Inspection Summary Report	Revision 0
M-C-700 209	Cleaning and Inspection of Control Panels	Revision 1
M-C-700-220	480 Volt Load Center Inspection and Cleaning	Revision 7

During the audit, the staff verified that the "scope of program," "preventive actions," "parameters monitored or inspected," "detection of aging effects," "monitoring and trending," "acceptance

criteria," and "corrective actions" program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

During the audit of the "operating experience" program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER.

The staff also audited the description of the SLRA AMP provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report Table XI-01.

SLRA AMP B.2.1.38, "Electrical Insulation for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits"

Summary of Information in the Application. The SLRA states that AMP B.2.1.38, "Electrical Insulation for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits," is an existing program with enhancements that will be consistent with the program elements in GALL-SLR Report AMP XI.E2, "Electrical Insulation for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits." To verify this claim of consistency, the staff audited the SLRA AMP. During the audit, the staff reviewed the enhancements associated with this AMP. The staff will document its review of the enhancements in the SER.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP- XI.E2	Electrical Insulation for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits	Revision 1
ER-AA-300-150	Cable Condition Monitoring Program	Revision 5
ER-AA-2030	Conduct of Equipment Reliability Manual	Revision 20
SI2R-63F-050- A1CE	Main Stack Rad Monitor RY-0-17-050A Electronic Calibration Check	Revision 11
ST-I-063-201-2	RX BLDG Vent Exhaust RAD Monitor Calibration and Functional Test for RIS-2-17-452A and C	Revision 6

During the audit, the staff verified that the “scope of program,” “preventive actions,” “parameters monitored or inspected,” “detection of aging effects,” “monitoring and trending,” “acceptance

criteria,” and “corrective actions” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects.

The staff also audited the description of the SLRA AMP provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.39, “Electrical Insulation for Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements”

Summary of Information in the Application. The SLRA states that AMP B.2.1.39, “Electrical Insulation for Inaccessible Medium-Voltage Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements,” is an existing program with enhancements and exceptions that will be consistent with the program elements in GALL-SLR Report AMP XI.E3A, “Electrical Insulation for Inaccessible Medium-Voltage Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements.” To verify this claim of consistency, the staff audited the SLRA AMP. Issues identified but not resolved in this report will be addressed in the SER. During the audit, the staff reviewed the exceptions and enhancements associated with this AMP. The staff will document its review of the exceptions to the GALL-SLR Report AMP and the enhancements in the SER.

Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP- XI.3A	Electrical Insulation for Inaccessible Medium- Voltage Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements – Program Basis Document	Rev. 1
ER-AA-300-150	Cable Condition Monitoring Program	Rev.5
XI.E3A Plant OpE	Electrical Insulation for Inaccessible Medium- Voltage Power Cables Not Subject to 10 CFR 50.49 EQ Requirements	08/10/2018
PB-AMPBD-E3	Manhole Inspection Frequency Basis Document	Rev. 0
PNLOC 1605H	Storm Sewer / MySmartcovers	09/20/2018

During the audit, the staff verified that the "scope of program," "parameters monitored or inspected," "detection of aging effects," "monitoring and trending," "acceptance criteria," and "corrective actions" program elements of the SLRA AMP are consistent with the corresponding

elements of the GALL-SLR Report AMP. The staff also verified Exelon's claim that aspects of the "preventive action" program element not associated with the exceptions identified in the SLRA are consistent with the corresponding program element in the GALL-SLR Report AMP. In addition, the staff found that sufficient information was not available to determine if the "preventive actions" program element, with the exceptions identified by the applicant, is consistent with the corresponding program element of the GALL-SLR Report AMP. The exceptions rely on level monitoring system to inspect water accumulation in manholes every 5 years, instead of annually as recommended in the GALL-SLR Report AMP XI.E3A. The staff will potentially issue an RAI in order to obtain the information necessary to determine if the exceptions will satisfy the criteria of 10 CFR 54.21(a)(3). The staff will document its evaluation of this potential RAI in the SER.

During the audit of the "operating experience" program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff also audited the description of the SLRA AMP provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.40, “Electrical Insulation for Inaccessible Instrument and Control Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements”

Summary of Information in the Application. The SLRA states that AMP B.2.1.40, “Electrical Insulation for Inaccessible Instrument and Control Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements,” is a new program with exceptions that will be consistent with the program elements in GALL-SLR Report AMP XI.E3B, “Electrical Insulation for Inaccessible Instrument and Control Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements.” To verify this claim of consistency, the staff audited the SLRA AMP. Issues identified but not resolved in this report will be addressed in the SER. During the audit, the staff reviewed the exceptions associated with this AMP. The staff will document its review of the exceptions to the GALL-SLR Report AMP in the SER.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
ER-AA-300-150	Cable Monitoring Program	Rev. 5
XI.E3B Plant OpE	Electrical Insulation for Inaccessible Instrumentation and Control Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	08/10/2018
PNLOC 1605H	Storm Sewer / MySmartcovers	09/20/2018
PB-AMPBD-E3	Manhole Inspection Frequency Basis Document	Rev. 0

Document	Title	Revision / Date
PB-PBD-AMP-XI.E3B-PBD	Electrical Insulation for Inaccessible Instrumentation and Control Cables Not Subject to 10 CFR50.49 Environmental Qualification Requirements – Program Basis Document	Rev.1

During the audit, the staff verified that the “scope of program,” “parameters monitored or inspected,” “detection of aging effects,” “monitoring and trending,” “acceptance criteria,” and “corrective actions” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP. The staff also verified Exelon's claim that aspects of the "preventive actions" program element not associated with the exceptions identified in the SLRA are consistent with the corresponding program elements in the GALL-SLR Report AMP. In addition, the staff found that sufficient information was not available to determine if the "preventive actions" program element, with the exceptions identified by the applicant, is consistent with the corresponding program element of the GALL-SLR Report AMP. The exceptions rely on a level monitoring system to inspect water accumulation in manholes every 5 years, instead of annually as recommended in the GALL-SLR Report AMP XI.E3A. The staff will potentially issue an RAI in order to obtain the information necessary to determine if the exceptions will satisfy the criteria of 10 CFR 54.21(a)(3). The staff will document its evaluation of this potential RAI in the SER.

During the audit of the "operating experience" program element, the staff's independent database search did not identify any operating experience that would indicate that the AMP may not be adequate to manage the associated aging effects.

The staff also audited the description of the SLRA AMP provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.41, "Electrical Insulation for Inaccessible Low-Voltage Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements"

Summary of Information in the Application. The SLRA states that AMP B.2.1.41, "Electrical Insulation for Inaccessible Low-Voltage Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements," is a new program with exceptions that will be consistent with the program elements in GALL-SLR Report AMP XI.E3C, "Electrical Insulation for Inaccessible Low-Voltage Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements." To verify this claim of consistency, the staff audited the SLRA AMP. Issues identified but not resolved in this report will be addressed in the SER. During the audit, the staff reviewed the exceptions associated with this AMP. The staff will document its review of the exceptions to the GALL-SLR Report AMP in the SER.

Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-	Electrical Insulation for Inaccessible Low-Voltage Power	Rev. 1

AMP- XI.E3C	Cables Not Subject to 10 CFR50.49 Environmental Qualification Requirements – Program Basis Document	
XI.E3C Plant OpE	Electrical Insulation for Inaccessible Low-Voltage Power Cables Not Subject to 10 CFR50.49 Environmental Qualification Requirements Plant OpE	08/10/2018
ER-AA-300-150	Cable Monitoring Program	Rev. 5
PB-AMPBD-E3	Manhole Inspection Frequency Basis Document	Rev. 0
PNLOC 1605H	Storm Sewer / MySmartcovers	09/20/2018

During the audit, the staff verified that the “scope of program,” “parameters monitored or inspected,” “detection of aging effects,” “monitoring and trending,” “acceptance criteria,” and “corrective actions” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP. The staff also verified Exelon’s claim that aspects of the “preventive actions” program element not associated with the exceptions identified in the SLRA are consistent with the corresponding program elements in the GALL-SLR Report AMP. In addition, the staff found that sufficient information was not available to determine if the “preventive actions” program element, with the exceptions identified by the applicant, is consistent with the corresponding program element of the GALL-SLR Report AMP. The exceptions rely on a level monitoring system to inspect water accumulation in manholes every 5 years, instead of annually as recommended in the GALL-SLR Report AMP XI.E3A. The staff will potentially issue an RAI in order to obtain the information necessary to determine if the exceptions will satisfy the criteria of 10 CFR 54.21(a)(3). The staff will document its evaluation of this potential RAI in the SER.

During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects.

The staff also audited the description of the SLRA AMP provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.2.1.42, “Metal Enclosed Bus”

Summary of Information in the Application. The SLRA states that AMP B.2.1.42, “Metal Enclosed Bus,” is a new program that will be consistent with the program elements in GALL- SLR Report AMP XI.E4, “Metal Enclosed Bus.” To verify this claim of consistency, the staff audited the SLRA AMP. At the time of the audit, Exelon had not yet fully developed the

documents necessary to implement this new program, and the staff’s audit addressed only program elements described in the applicant’s basis document.

Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-XI.E4	Peach Bottom Atomic Power Station, Second License Renewal Project – Metal Enclosed Bus Program Basis Document	12/14/17
6280-E-7	Purchase Specification – Metal Enclosed Bus	08/13/1971
6280-E7-40-216-S	Drawing - Bus Duct Arrangement 15 kV 3000A CU	05/23/1974
6280-E7-40-8 (sh 2)	Drawing - Bus Duct Arrangement	01/10/1979
6280-E7-40-8 (sh 3)	Drawing - Bus Duct Arrangement	09/23/1974
M-054-003	4.16 kV/13.2 kV Non-Segmented Bus Inspection/Maintenance	Revision 3
ER-AA-300-140	License Renewal Metal Enclosed Bus Program	Revision 2

During the audit, the staff verified Exelon's claim that the "scope of program," "preventive actions," "parameters monitored or inspected," "detection of aging effects," "monitoring and trending," "acceptance criteria," and "corrective actions" program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

During the audit, the staff made the following observation:

 The staff reviewed the AMP basis document PB-PBD-AMP-XI.E4 and noted that this document, as well as the SLRA AMP B.2.1.42, "Metal Enclosed Bus," excluded elastomers from this program. The staff discussed this exclusion with Exelon personnel during breakout sessions and requested photos of these components. The staff confirmed lack of elastomers (gaskets) on the in-scope metal enclosed bus sections by reviewing drawings 6280-E7-40-216-S, 6280-E7-40-8 (sh 2), 6280-E7- 40-8 (sh 3), as well as photos provided by Exelon in the portal.

During the audit of the "operating experience" program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER.

The staff also audited the description of the SLRA section A. 2.1.42, “Metal Enclosed Bus,” provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report Table XI-01.

SLRA AMP B.2.1.43, “Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements”

Summary of Information in the Application. The SLRA states that AMP B.2.1.43, “Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements,” is a new program that will be consistent with the program elements in GALL-SLR Report AMP XI.E6, “Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements.” To verify this claim of consistency, the staff audited the SLRA AMP. At the time of the audit, Exelon had not yet fully developed the documents necessary to implement this new program, and the staff’s audit addressed only the program elements described in the applicant’s basis document.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
ER-AA-300-120	Electrical Cable Connections not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program – Implementation Document	Revision 4
PB-AMPBD-E6	Electrical Cable Connections not Subject to 10 CFR 50.49 Environmental Qualification Requirements – Sample Basis Document	Revision 0
PB-PBD-AMP- XI.E6	Peach Bottom Atomic Power Station Second License Renewal Project - Electrical Cable Connections not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Revision 2
MA-AA-716-230- 1003	Thermography Program Guide	Revision 5
S-8506-A	Standard - Electrical Bolted Connections	02/01/2009

During the audit, the staff verified Exelon’s claim that the “scope of program,” “preventive actions,” “parameters monitored or inspected,” “detection of aging effects,” “monitoring and trending,” and “acceptance criteria” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

During the audit, the staff made the following observations:

- The staff reviewed AMP basis document PB-PBD-AMP-XI.E6 and noted that only two general types of connections were listed to be included in the program (bolted and crimped). The sample basis document PB-AMPBD-E6 lists more inclusive types, such as, splice, butt, bolted, crimp type, ring lugs, connectors, and terminal blocks. Subsequent to the breakout session discussions, Exelon revised PB-PBD-AMP-XI.E6 and PB-AMPBD-E6 to clarify that the program will encompass all connections types

utilized at the site and the sampling basis will include all connections such as, splice,

butt, bolted, crimp type, ring lugs, connectors, and terminal blocks.

- The staff reviewed AMP basis document PB-PBD-AMP-XI.E6 and noted that although

this is a one-time inspection program, trending is not included for tests that are trendable and may have to be repeated periodically based on the initial finding results. Subsequent to the breakout session discussions, Exelon revised PB-PBD-AMP-XI.E6 to include trending for tests that are trendable and are deemed necessary to be repeated as periodic tests.

During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER.

The staff also audited the description of the SLRA section A.2.1.43, “Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements,” provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report Table XI-01.

SLRA AMP B.2.2.1, “Wooden Pole”

Summary of Information in the Application. The SLRA states that AMP B.2.2.1, “Wooden Pole,” is an existing plant-specific program with enhancement. The staff audited the SLRA AMP to determine consistency with SRP-SLR Section A.1.2.3, “Aging Management Program Elements.”

Audit Activities. During its audit, the staff reviewed onsite documentation provided by the applicant. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-PS- 1	GALL-SLR Program PS-1- Wooden Pole	Revision 1 09/13/2017
PS-1	Wooden Pole PS-1 References	-
ER-AA-700-1001	Susquehanna Substation Wooden Pole Inspection Activity	Revision 1 10/04/2013

During the audit, the staff verified Exelon’s stated consistency with SRP-SLR Section A.1.2.3 for the “scope of program,” “preventive actions,” “parameters monitored or inspected,” “detection of aging effects,” “monitoring and trending,” “acceptance criteria,” and “corrective actions” program elements of the SLRA AMP.

During the audit of the “operating experience” program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER.

The staff also audited the description of the SLRA AMP provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B3.1.1, “Fatigue Monitoring Program”

Summary of Information in the Application. The SLRA states that AMP B3.1.1, “Fatigue Monitoring Program,” is an existing program with enhancements that will be consistent with the program elements in GALL-SLR Report AMP X.M1, “Fatigue Monitoring.” To verify this claim of consistency, the staff audited the SLRA AMP. During the audit, the staff reviewed the enhancements associated with this AMP. The staff will document its review of the enhancements in the SER.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-X.M1	Program Basis Document, Fatigue Monitoring	Revision 1
FP-PBAP-404	Cycle Counting and Cycle-Based Fatigue Report for the Transient and Fatigue Monitoring System for Peach Bottom Atomic Power Station Units 2 and 2 (SIR-99-122)	Revision 6, August 2017
FP-PBAP-405	SI:FatiguePro 4.0 SBF Transfer Functions for Peach Bottom Atomic Power Station Units 2 and 3 Environmental Fatigue Monitoring System	Revision 6, June 2017
FP-PBAP-406	Software Verification and Validation Report for Peach Bottom Plant-Specific SI:FatiguePro 4.0 Software	Revision 2, September 2017
ERC-PB-11-00367-000	Fatigue Program Updates for License Renewal	October 20, 2011
1400630.301	Peach Bottom Second License Renewal (SLR), 60 and 80 Year Cycle and Fatigue Projections	Revision 1
1400630.302	Peach Bottom Second License Renewal (SLR), Peach	Revision 0

	Bottom Fatigue Usage Assessment	
1400630.302	Peach Bottom Second License Renewal (SLR), Peach Bottom Environmentally-Assisted Fatigue Screening	Revision 0

During the audit, the staff verified that the "scope of program," "preventive actions," "parameters monitored or inspected," "detection of aging effects," "monitoring and trending," "acceptance criteria," and "corrective actions" program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

During the audit of the "operating experience" program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER.

The staff also audited the description of the SLRA Fatigue Monitoring Program provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA AMP B.3.1.2, "Neutron Fluence Monitoring"

Summary of Information in the Application. The SLRA states that AMP B.3.1.2, "Neutron Fluence Monitoring" is an existing program with an enhancement that will be consistent with the program elements in GALL-SLR Report AMP X.M2, "Neutron Fluence Monitoring." To verify this claim of consistency, the staff audited the SLRA AMP. Issues identified but not resolved in this report will be addressed in the SER. During the audit, the staff reviewed the enhancement associated with this AMP. The staff will document its review of the enhancement in the SER.

Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP-X.M2	Program Basis Document: Neutron Fluence Monitoring	Rev. 1, 01/10/2018
AR 02713499	Action Request Report, 2018 Withdrawal of Unit 2 120o RPV Surveillance Capsule	09/08/2016
AR 03962434	Action Request Report, 2018 Withdrawal of Unit 2 120o RPV Surveillance Capsule	01/12/2017
AR 01666001	Action Request Report, Duane Arnold ISP Surveillance Data Applicable to PBAPS Unit	05/30/2014
OE 302507 [1]	P-T Curves Non-Conservative Based on Integrated	01/10/2013

	Surveillance Capsule Analysis Results	
OE 293244 [1]	Non-Conservative Fluence Inputs to Technical Specification P-T Limit Curves	01/19/2012
OE 252123 [1]	Non-Conservative Technical Specification P-T Limit Curves Identified During Thermal Power Optimization Project Review	12/06/2011
OE 234035 [1]	Reactor Coolant System P-T Limits	09/17/2008
CC-AA-102	Design Input and Configuration Change Impact Screening	
ER-AA-370	Reactor Coolant Pressure Boundary (RCPB) Integrity	

Document	Title	Revision / Date
NF-AB-105	Managing Cycle Design Inputs and Requirements	
General Electric-Hitachi Record: GE 003N7847	Peach Bottom Atomic Power Station Units 2 and 3, 80- Year Subsequent License Renewal, Task T0301: RPV Fracture Toughness Evaluation	Rev. 0, Dec. 2016
Transware Record: EXL-PB0-001-R-005 / EXL-PB0-002-R- 005	Transware Fluence Evaluation Report: PBAPS Unit 3 Vessel Internal Components Fluence Evaluations	06/04/2014
Transware Record: EXL-PB0-001-R-003 / EXL-PB0-002-R- 003	Transware Fluence Evaluation Report: PBAPS Unit 2 Vessel Internal Components Fluence Evaluations	06/04/2014
EPRI Proprietary Report No. 1019053	BWRVIP-145-A: BWR Vessel and Internals Project, Evaluation of Susquehanna Unit 2 Tope Guide and Core Shroud Materials Samples Using RAMA Fluence Methodology. [2]	June 2009
General Electric-Hitachi Proprietary Report No. NEDC-32983-P-A [3]	Licensing Topical Report: General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations	Rev. 2, January 2006

Notes: 1. The record represents generic operating experience that was assessed for applicability to the units.
2. “-A” of the BWRVIP designation referenced in the title designates the report and methodology has been

approved by the staff.
3. “-A” in the Report Number designates the report and methodology has been approved by the staff.

During the audit, the staff verified Exelon’s claim that the “scope of program,” “preventive actions,” “parameters monitored or inspected,” “detection of aging effects,” “monitoring and trending,” “acceptance criteria,” and “corrective actions” program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

During the audit, the staff made the following observations:

 The staff reviewed General Electric-Hitachi Company Record No. GE 003N7847, and Transware, Enterprises, Inc. Record Nos. EXL-PB0-001-R-005 / EXL-PB0-002-R-005, and EXL-PB0-001-R-003 / EXL-PB0-002-R-003 and verified that Exelon is using two different vendors to perform fluence projections for RPV and RVI components in Unit 2 and Unit 3: (a) GE-Hitachi (GEH) for the neutron fluence projections for PBAPS RPV components, and (b) Transware Enterprises, Inc., use of EPRI's RAMA methodology for performance of the neutron fluence projections for the PBAPS RVI components. The staff did not have any inquiries in relation to the staff's review of these records or the contents of these records.

 The staff reviewed PBAPS Record Nos. CC-AA-102, ER-AA-370, and NF-AB-105, and verified the applicant has appropriate procedure controls in place to perform appropriate component design, core design, and operating characteristic and specification reviews for preparing design reports and providing appropriate design inputs to those vendors that may be contracted to perform neutron fluence evaluations of the RPV or RVI components in the PBAPS unit designs. The staff did not have any inquiries in relation to the staff's review of these records or the contents of these records.

During the audit of the "operating experience" program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff also reviewed generic operating experience that was identified by the applicant as being potentially applicable to this AMP. The staff will evaluate the identified plant- specific and generic operating experience in the SER.

The staff also audited the description of the SLRA Neutron Fluence Monitoring AMP provided in the SLRA UFSAR Supplement Section A.3.1.2. The staff verified this description is consistent with the description provided in the Table X-01 of GALL-SLR Report for GALL-SLR AMP X.M2, "Neutron Fluence Monitoring."

SLRA AMP B.3.1.3, "Environmental Qualification of Electric Equipment"

Summary of Information in the Application. The SLRA states that AMP B.3.1.3, "Environmental Qualification of Electric Equipment," is an existing program with enhancements that will be consistent with the program elements in GALL-SLR Report AMP X.E1, "Environmental Qualification of Electric Equipment." To verify this claim of consistency, the staff audited the SLRA AMP. During the audit, the staff reviewed the enhancements associated with this AMP. The staff will document its review of the enhancements in the SER.

Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date

PB-PBD-AMP-X.E1	Environmental Qualification of Electric Equipment – Program Basis Document	Rev. 1
AR 4106712-09	Aging Management Program (AMP) Effectiveness Review - Peach Bottom Environmental Qualification Activities AMP	Rev. 1
CC-AA-203	Environmental Qualification Program	Rev.1
EQ-PB-011	Environmental Qualification - Okonite 600 V Power & Control Cable and 5 kV Power Cable	Rev. 1
EQ-PB-016	Environmental Qualification - Brand Rex Cable	Rev. 1

During the audit, the staff verified that for the program elements that Exelon declared were consistent, the "scope of program," "preventive Actions," "parameters monitored or inspected," "detection of aging effects," "monitoring and trending," "acceptance criteria," and "corrective actions" program elements of the SLRA AMP are consistent with the corresponding elements of the GALL-SLR Report AMP.

During the audit of the "operating experience" program element, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects.

The staff also audited the description of the SLRA AMP provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

2.2 Time Limited Aging Analyses (TLAAs)

SLRA TLAA Section 4.1, "Identification and Evaluation of Time-Limited Aging Analyses"

Summary of Information in the Application. SLRA Section 4.1, "Identification and Evaluation of Time-Limited Aging Analyses (TLAAs)," discusses the applicant's methodology for identifying those plant analyses, evaluations, calculations, or assessments (AECAs) that qualify as TLAAs, consistent with the definition for TLAAs provided in 10 CFR 54.3(a), and for identifying those TLAAs that must be included and evaluated in the SLRA in accordance with 10 CFR 54.21(c)(1). SLRA Section 4.1 provides: (a) a list of those AECAs that qualify as TLAAs and have been identified as TLAAs in accordance with the requirement in 10 CFR 54.21(c)(1), and (b) a pointer to the sections or subsections in SLRA Chapter 4 that provides the applicant's evaluation of the TLAAs and the basis for dispositioning the TLAAs in accordance with 10 CFR 54.21(c)(1)(i), (ii), or (iii).

Section 4.1 of the SLRA also summarizes the applicant's review that was performed to identify any regulatory exemptions that have been granted for the current licensing basis (CLB) in accordance with the requirements in 10 CFR 50.12 and are based on a TLAA, and the results of its regulatory exemption review, as required by 10 CFR 54.21(c)(2).

The staff audited SLRA Section 4.1, applicable information in the UFSAR, and supporting information, documents, and records to verify that Exelon has provided a comprehensive list of AECAs that qualify as TLAAs in accordance with 10 CFR 54.3(a) and has identified these AECAs as TLAAs in accordance with 10 CFR 54.21(c)(1). The staff also audited this information to: (a) verify that the applicant has appropriately identified regulatory exemptions granted in the CLB under the requirements of 10 CFR 50.12 that are based on a TLAA, as required in accordance with 10 CFR 54.21(c)(2); and (b) verify, for those 50.12 exemptions that are based on a TLAA (if any), that the applicant has provided an appropriate evaluation of the exemptions in the SLRA justifying their continuation during the subsequent period of extended operation. As part of these efforts, the staff performed a search of the NRC's ADAMS document control database for any regulatory exemptions that may have been granted in the CLB under the requirements of 10 CFR 50.12 for the reactor units that are within the scope of the SLRA. The staff will address any issues identified but not resolved in this audit report in the SER.

Audit Activities. During its audit, the staff interviewed the applicant's staff and reviewed documentation provided by the applicant. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-TLAABD	Peach Bottom Atomic Power Station Units 2 and 3, License Renewal Project, TLAA Basis Document – Part 2 -TLAA Evaluation (Pages 4.1-1 through 4.1-9, and Attachment 6, "List of Reports Considered to PTLAAs)	Revision 0

Document	Title	Revision / Date
LR-P-007	Peach Bottom Plant-Specific Exemptions Granted Pursuant to 10 CFR 50.12	Revision 0

SLRA TLAA Section 4.1, "Identification and Evaluation of Time-Limited Aging Analyses"

Summary of Information in the Application. SLRA Section 4.1, "Identification and Evaluation of Time-Limited Aging Analyses (TLAAs)," discusses the applicant's methodology for identifying those plant analyses, evaluations, calculations, or assessments (AECAs) that qualify as TLAAs, consistent with the definition for TLAAs provided in 10 CFR 54.3(a), and for identifying those TLAAs that must be included and evaluated in the SLRA in accordance with 10 CFR 54.21(c)(1). SLRA Section 4.1 provides: (a) a list of those AECAs that qualify as TLAAs and have been identified as TLAAs in accordance with the requirement in 10 CFR 54.21(c)(1), and (b) a pointer to the sections or subsections in SLRA Chapter 4 that provides the applicant's evaluation of the TLAAs and the basis for dispositioning the TLAAs in accordance with 10 CFR 54.21(c)(1)(i), (ii), or (iii).

Section 4.1 of the SLRA also summarizes the applicant's review that was performed to identify any regulatory exemptions that have been granted for the current licensing basis (CLB) in accordance with the requirements in 10 CFR 50.12 and are based on a TLAA, and the results of its regulatory exemption review, as required by 10 CFR 54.21(c)(2).

The staff audited SLRA Section 4.1, applicable information in the UFSAR, and supporting information, documents, and records to verify that Exelon has provided a comprehensive list of AECAs that qualify as TLAAs in accordance with 10 CFR 54.3(a) and has identified these AECAs as TLAAs in accordance with 10 CFR 54.21(c)(1). The staff also audited this information to: (a) verify that the applicant has appropriately identified regulatory exemptions granted in the CLB under the requirements of 10 CFR 50.12 that are based on a TLAA, as required in accordance with 10 CFR 54.21(c)(2), and (b) verify, for those 50.12 exemptions that are based on a TLAA (if any), that the applicant has provided an appropriate evaluation of the exemptions in the SLRA justifying their continuation during the subsequent period of extended operation. As part of these efforts, the staff performed a search of the NRC's ADAMS document control database for any regulatory exemptions that may have been granted in the CLB under the requirements of 10 CFR 50.12 for the reactor units that are within the scope of the SLRA. The staff will address any issues identified but not resolved in this audit report in the SER.

Audit Activities. During its audit, the staff interviewed the applicant's staff and reviewed documentation provided by the applicant. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-TLAABD	Peach Bottom Atomic Power Station Units 2 and 3, License Renewal Project, TLAA Basis Document – Part 2 - TLAA Evaluation (Pages 4.1-1 through 4.1-9, and Attachment 6, "List of Reports Considered to PTLAAs)	Revision 0

Document	Title	Revision / Date
LR-P-007	Peach Bottom Plant-Specific Exemptions Granted Pursuant to 10 CFR 50.12	Revision 0

Summary of Audit Review for Identified TLAAs

During the audit of the SLRA Section 4.1, relevant information in the UFSAR, and supporting information, the staff verified that Exelon may not have identified all AECAs that qualify as a TLAA in accordance with 10 CFR 54.3(a). The following items summarize the staff's observations relative to AECAs that required further discussions with the applicant:

- The staff noted that, in Basis Document PB-TLAABD, Revision 0, Appendix A, the applicant includes a reference to General Electric-Hitachi (GEH) Report No. GEH-0000- 0151-0155). The report includes an analysis of 16 existing flaws that were detected in the upper reactor pressure vessel (RPV) head of Unit 2. The applicant identifies that the flaw analysis justifies crack stability of the flaws of a 60-year life but qualifies that the analysis does not need to be identified as a TLAA because the component will be re- inspected in the 5th 10-Year ISI internal for the impacted unit. The staff will seek further justification on why it would preclude identification of this analysis as a TLAA, particularly if the 60-year flaw analysis was being used as the basis for a safety decision to re- inspect the upper head at a particular time in the 5th 10-Year ISI interval. In contrast to the reference of this GEH report, row 158 of the Basis Document appendix identifies that an analogous year 2002 flaw evaluation of similar indications in Unit 3 RPV upper head is a TLAA for the LRA.
- The staff noted that, in Basis Document PB-TLAABD, Revision 0, Appendix A, the applicant includes a reference to site Record No. PEAM-MPLUS-9, Rev. 000 (a year 2013 record) and identifies that the neutron flux evaluation in the record for the RPV shell plates, nozzles, and welds in Units 2 and 3 qualifies as a TLAA for the units. However, later in the appendix, the applicant identifies that an updated fluence analysis (Record No. 349-1-VC-39, Sht. 0001, Rev. 000) was performed in 2015 for Unit 3. For the year 2015 fluence analysis for Unit 3 in Record No. 349-1-VC-39, Sht. 0001, Rev. 000, the applicant concluded the analysis is not a TLAA because it is not contained or incorporated by reference in the CLB. The staff will seek further clarification as to whether the more recent, Year 2015 fluence analysis for Unit 3, is superseding the previous Year-2013 fluence analysis referenced for Unit 3 RPV in site Record No. PEAM-MPLUS-9, Rev. 000 and, if so, why the 2015 analysis would not need to be identified as a TLAA for the SLRA.
- The staff noted that, in Basis Document PB-TLAABD, Revision 0, Appendix A, the applicant includes a reference to site Record 99-02244, Revision 1. Site Record 99- 02244 includes a flaw evaluation of an indication that was detected in one of the unit's jet pump adapter welds. The applicant states that the evaluation in the site record does not qualify as TLAA because it does not meet Criterion 3 for defining TLAAs in 10 CFR 54.3(a). However, the applicant does not explain why the analysis does not meet Criterion 3 in 10 CFR 54.3(a). The staff will seek further justification as to why the evaluation in site Record 99-02244, Rev. 01, is not considered to meet Criterion 3 for TLAA identification in 10 CFR 54.3(a).

During the audit, the staff discussed these AECAs with the applicant during a scheduled audit breakout teleconference conducted on December 13, 2018. These matters will be reflected in one or more potential RAIs to the applicant and in the staff's evaluation of SLRA Section 4.1.

Summary of Audit Review of Exemptions that May Meet the Criteria in 10 CFR 54.21(c)(2)

During the audit of the SLRA Section 4.1, information in the UFSAR, and supporting information, the staff verified that the CLB does not include any regulatory exemptions granted in accordance with 10 CFR 50.12 that are based on a TLAA, such that the exemptions would need to be identified in the SLRA and evaluated in accordance with the requirements of
10 CFR 54.21(c)(2).

SLRA TLAA Section 4.2.2, "Reactor Vessel Upper Shelf Energy Analyses"

Summary of Information in the Application. SLRA Section 4.2.2, "Reactor Vessel Upper Shelf Energy (USE) Analyses" (henceforth the TLAA on USE), discusses the neutron fluence- dependent analyses that are included in the current licensing basis (CLB) to evaluate potential drops in the upper shelf energy fracture toughness properties of ferritic steel components

that were used to fabricate the reactor pressure vessels (RPVs). Exelon identified that, collectively, these analyses constitute a TLAA for the subsequent license renewal application (SLRA) and dispositioned the analyses in accordance with 10 CFR 54.21(c)(1)(ii).

To verify that Exelon provided a basis to support its disposition of the TLAA, the staff audited the TLAA. The staff will address issues identified but not resolved in this report in the SER.

Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
SLRA Section 4.2.1	Reactor Vessel and Internals Neutron Fluence Analyses	Revision 0
SLRA Section 4.2.2	Reactor Vessel Upper Shelf Energy (USE) Analyses	Revision 0
PB-TLAABD	Peach Bottom Atomic Power Station Units 2 and 3, License Renewal Project, TLAA Basis Document – Part 1 – TLAA Identification, Attachment 7, PBAPS First LRA TLAA and SLRA TLAA Comparison	Revision 0
PB-TLAABD	Peach Bottom Atomic Power Station Units 2 and 3, License Renewal Project, TLAA Basis Document –	Revision 0

	Part 2 – TLAA Evaluation, Section 4.2, Reactor Vessel and Internals Neutron Embrittlement Analyses	
EPRI BWRVIP Report No. 1008872	BWRVIP-74-A: BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines for License Renewal	Revision 0, June 2003
PECO Energy Correspondence Letter	Peach Bottom Atomic Power Station, Units 2 and 3, Limerick Generating Station, Units 1 and 2, Response to Generic Letter 92- 01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity"	August 15, 1995

Document	Title	Revision / Date
NRC Correspondence Letter to PECO Energy Company	Closeout for PECO Energy Company (PECO) Response to Generic Letter 92-01, Revision 1, Supplement 1, Peach Bottom Atomic Power Plant, Units 2 and 3"	September 25, 1996
GE-Hitachi Nuclear Energy Proprietary Report No. 003N7847 (Class II Report)	Project Task Report, Exelon Nuclear, LLC, Peach Bottom Atomic Power Station, Units 2 and 3, 80-Yer Subsequent License Renewal, Task T0301: RPV Fracture Toughness Evaluation	Revision 0, December 2016
GE-Nuclear Report No. SASR 88-24 (ADAMS ML12242A122)	Peach Bottom Atomic Power Station, Unit 2 Vessel Surveillance Materials Testing and Fracture Toughness Analysis	May 1988
GE-Nuclear Report No. SASR 90-50 (ADAMS ML12242A123)	Peach Bottom Atomic Power Station, Unit 3 Vessel Surveillance Materials Testing and Fracture Toughness Analysis	June 1990
GE-Nuclear Proprietary Report No. NEDC-33556P	Safety Analysis Report for Exelon Peach Bottom Atomic Power Station, Units 2 and 3, Constant Pressure Power Uprate	Revision 0, September 2012
NRC Correspondence Letter and Safety Evaluation to Exelon Nuclear (ADAMS ML14133A046)	Peach Bottom Atomic Power Station, Units 2 and 3 – Issuance of Amendments RE: Extended Power Uprate (TAC Nos. ME9631 and ME9632)	August 25, 2014
GE-Nuclear Proprietary Report No. NEDC-33873P	Safety Analysis Report for Exelon Peach Bottom Atomic Power Station, Units 2 and 3, Thermal Power Optimization	Revision 0 February 2017
NRC Correspondence Letter and Safety Evaluation to Exelon Nuclear (ADAMS ML17286A013)	Peach Bottom Atomic Power Station, Units 2 and 3 – Issuance of Amendments RE: Measurement Uncertainty Recapture Power Uprate (CAC Nos. MF9289 and MF9290; EPID L-2017-LLS-0001)	November 15, 2017

Report

Nos. 003N7847, Revision 0, and 004N6849, Revision 0.

During the audit of the TLAA, the staff verified that Exelon has provided its basis that supports its disposition of 10 CFR 54.21(c)(1)(ii).
During the audit, the staff made the following observations based on its in-house audit review of relevant information in SLRA Sections 4.2.1 and 4.2.2, Basis Document Report PB-TLAA-BD, Parts 1 and 2, the applicant's previous responses to Generic Letters (GLs) 92-01, Revision 1, and 92-01, Revision 1, Supplement 1, and GE-Hitachi Nuclear Energy (GE) Proprietary Class II

- The staff observed that GE Proprietary Report No. 003N7847, Revision 0, serves as the licensing basis document for the applicant's TLAA on USE. The staff observed that GE Proprietary Report No. 004N6849, Revision 0, is used only to address potential uncertainties in the neutron fluence values that were reported for the RPV beltline components in SLRA Section 4.2.1.1. The staff observed that the 004N6849 report does not serve as the licensing basis document for the TLAA on USE. Based on these observations, the staff noted that GE Proprietary Report No. 003N7847, Revision 0, establishes the Unit 2-specific and Unit 3-specific RPV components that need to be included in the scope of the TLAA on USE.
- The staff observed that the scope of the TLAA on USE covers the following RPV components in Unit 2 that are made from ferritic steel materials: (a) RPV shell plates located in the RPV lower and lower intermediate shells, (b) RPV axial welds located in the lower and lower intermediate shells, and (c) the RPV circumferential weld adjoining the lower shell course to the lower intermediate shell. The staff observed that, based on the design of the Unit 2 RPV and information reviewed by the staff, the TLAA on USE does not include any upper intermediate shell plates or welds as extended beltline components within the scope of the TLAA.[1]
- The staff observed that the scope of the TLAA on USE covers the following RPV components in Unit 3 that are made from ferritic steel materials: (a) RPV shell plates located in the RPV lower and lower intermediate shells, (b) RPV axial welds located in the lower and lower intermediate shells, (c) the RPV circumferential weld adjoining the lower shell course to the lower intermediate shell, (d) RPV shell plates in the intermediate shell, which are extended beltline components for the Unit 3 assessment, (e) RPV axial welds in the intermediate shell, which are extended beltline components for the Unit 3 assessment, and (f) the lower intermediate shell-to-intermediate shell circumferential weld, which is an extended beltline component for the Unit 3 assessment.
- The staff observed that, in the SLRA, the applicant only provided end-of-second-renewed-life USE or equivalent margins analysis (EMA) values for those Unit-specific RPV base metal and weld components which are considered to be the most limiting for the TLAA on USE assessment. Based on its review of the applicant's response to Generic Letter 92-01, Revision 1, Supplement, and supporting EPRI BWRVIP records, the staff observed that Exelon's original licensing basis did not have a sufficient amount of Charpy-impact test data to

establish the un-irradiated upper shelf energy plateaus for all RPV beltline base metal and weld materials. The staff will reference this observation in its evaluation of the TLAA on USE, as documented in the SER.

- The staff observed that the EMA input parameter values listed in SLRA Tables 4.2.2-3 – 4.2.2-6 for specified Unit 2 and 3 RPV and ISP surveillance materials were consistent with those provided in GE-Nuclear Proprietary Report No. NEDC-33873P. The staff also

[1] For Unit 3, Exelon's corresponding nomenclature of this shell in the Unit 3 RPV design is referred to as the intermediate shell.

observed that these EMA input parameters were approved in the staff's November 15, 2017, safety evaluation for the license amendment granting the measurement uncertainty recapture power uprates for the reactor units (ADAMS Accession No. ML17286A013).

During the audit of the operating experience associated with the TLAA, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff did not identify any additional aging effects or mechanisms (i.e., other than loss of fracture toughness due to neutron irradiation embrittlement) that would have an impact on the applicant's evaluation of the TLAA.

The staff also audited the description of the SLRA TLAA on USE provided in the UFSAR supplement. The staff verified this description provided an adequate description of why a limiting BWRVIP-74-A equivalent margins analysis (EMA) was needed as the basis for meeting the USE requirements in 10 CFR Part 50, Appendix G, and how the EMAs for the reactor units have been projected to the end of the subsequent period of extended operation in accordance with the requirement in 10 CFR 54.21(c)(1)(ii).

SLRA TLAA Section 4.2.3, "Reactor Vessel Adjusted Reference Temperature (ART) Analyses"

Summary of Information in the Application. SLRA Section 4.2.3, "Reactor Vessel Adjusted Reference Temperature (ART) Analyses," discusses the analyses for the reactor vessel. Exelon dispositioned the TLAA in accordance with 10 CFR 54.21(c)(1)(ii). To verify that Exelon provided a basis to support its disposition of the TLAA, the staff audited the TLAA.

Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision /

		Date
PB-TLAABD, Part 1	Time-Limited Aging Analysis (TLAA) Basis Document – Part 1 – TLAA Identification	Revision 0
PB-TLAABD, Part 2	Time-Limited Aging Analysis (TLAA) Basis Document – Part 2 – TLAA Evaluation, Section 4.2.3, Reactor Vessel Adjusted Reference Temperature (ART) Analyses	Revision 0

During the audit of the TLAA, the staff verified that Exelon has provided its basis that supports its disposition of 10 CFR 54.21(c)(1)(ii).

During the audit of the operating experience associated with the TLAA, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects.

The staff also audited the description of the SLRA Reactor Vessel Adjusted Reference Temperature (ART) Analyses provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA TLAA Section 4.2.4, “Reactor Vessel Pressure-Temperature (P-T) Limits”

Summary of Information in the Application. SLRA Section 4.2.4, “Reactor Vessel Pressure- Temperature (P-T) Limits,” discusses the analyses for the reactor pressure vessel (RPV) P-T limit curves. Exelon dispositioned the TLAA in accordance with 10 CFR 54.21(c)(1)(iii).

To verify that Exelon provided a basis to support its disposition of the TLAA, the staff audited the TLAA.

Audit Activities. During its audit, the staff reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
Renewed License No. DPR-44; Appendix A	Technical Specifications for Peach Bottom Atomic Power Station Unit 2; Section 5.6.7, “Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)”	Amendment No. 305
Renewed License No. DPR-56; Appendix A	Technical Specifications for Peach Bottom Atomic Power Station Unit 3; Section 5.6.7, “Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)”	Amendment No. 309

During the audit of the TLAA, the staff verified that Exelon has provided its basis that supports its disposition of 10 CFR 54.21(c)(1)(iii).

During the audit of the operating experience associated with the TLAA, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. No significant plant specific operating experience associated with TLAA Section 4.2.4 was noted by the staff during its review.

The staff also audited the description of the “SLRA Reactor Vessel Pressure-Temperature (P-T) Limits” TLAA provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the SRP-SLR.

SLRA TLAA Section 4.2.5, “Reactor Vessel Circumferential Weld Failure Probability Analyses”

Summary of Information in the Application. SLRA Section 4.2.5, “Reactor Vessel Circumferential Weld Failure Probability Analyses,” discusses the analyses for the reactor pressure vessel (RPV) circumferential welds. Exelon dispositioned the TLAA in accordance with 10 CFR 54.21(c)(1)(iii).

To verify that Exelon provided a basis to support its disposition of the TLAA, the staff audited the TLAA.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

During the audit of the TLAA, the staff verified that Exelon has provided its basis that supports its disposition of 10 CFR 54.21(c)(1)(iii).

During the audit of the operating experience associated with the TLAA, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. No significant plant specific operating experience associated with TLAA Section 4.2.5 was noted by the staff during its review.

The staff also audited the description of the SLRA “Reactor Vessel Circumferential Weld Failure Probability Analyses” TLAA provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the SRP-SLR.

SLRA TLAA Section 4.2.6, “Reactor Vessel Axial Weld Failure Probability Analyses”

Summary of Information in the Application. SLRA Section 4.2.6, “Reactor Vessel Axial Weld Failure Probability Analyses,” discusses the analyses for the reactor pressure vessel (RPV) axial welds. Exelon dispositioned the TLAA in accordance with 10 CFR 54.21(c)(1)(ii).

To verify that Exelon provided a basis to support its disposition of the TLAA, the staff audited the TLAA.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

During the audit of the TLAA, the staff verified that Exelon has provided its basis that supports its disposition of 10 CFR 54.21(c)(1)(ii).

Document	Title	Revision / Date
GE Hitachi Nuclear Energy 003N7847	Project Task Report Exelon Nuclear, LLC Peach Bottom Atomic Power Stations Units 2 and 3 80-Year Subsequent License Renewal Task T0301: PRP Fracture Toughness Evaluation	Revision 0, December 2016

Document	Title	Revision / Date
GE Hitachi Nuclear Energy 003N7847	Project Task Report Exelon Nuclear, LLC Peach Bottom Atomic Power Stations Units 2 and 3 80-Year Subsequent License Renewal Task T0301: PRP Fracture Toughness Evaluation	Revision 0, December 2016

During the audit of the operating experience associated with the TLAA, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will document its review of relevant operating experience in the SER.

The staff also audited the description of the SLRA “Reactor Vessel Axial Weld Failure Probability Analyses” TLAA provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the SRP-SLR.

SLRA TLAA Section 4.2.7, “Reactor Vessel Reflood Thermal Shock Analysis”

Summary of Information in the Application. SLRA Section 4.2.7, “Reactor Vessel Reflood Thermal Shock Analysis,” discusses the analysis for the reactor vessel. Exelon dispositioned the TLAA in accordance with 10 CFR 54.21(c)(1)(ii).

To verify that Exelon provided a basis to support its disposition of the TLAA, the staff audited the TLAA.

Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-TLAABD, Part 1	Time-Limited Aging Analysis (TLAA) Basis Document – Part 1 – TLAA Identification	Revision 0
PB-TLAABD, Part 2	Time-Limited Aging Analysis (TLAA) Basis Document – Part 2 – TLAA Evaluation, Section 4.2.7, Reactor Vessel Reflood Thermal Shock Analysis	Revision 0

During the audit of the TLAA, the staff verified that Exelon has provided its basis that supports its disposition of 10 CFR 54.21(c)(1)(ii).

During the audit of the operating experience associated with the TLAA, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects.

The staff also audited the description of the SLRA Reactor Vessel Reflood Thermal Shock Analysis provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA TLAA Section 4.2.8, "Core Shroud Reflood Thermal Shock Analysis"

Summary of Information in the Application. SLRA Section 4.2.8, "Core Shroud Reflood Thermal Shock Analysis," discusses the analysis for the core shroud. Exelon dispositioned the TLAA in accordance with 10 CFR 54.21(c)(1)(i).

To verify that Exelon provided a basis to support its disposition of the TLAA, the staff audited the TLAA.

Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	

		Revision / Date
PB-TLAABD, Part 1	Time-Limited Aging Analysis (TLAA) Basis Document – Part 1 – TLAA Identification	Revision 0
PB-TLAABD, Part 2	Time-Limited Aging Analysis (TLAA) Basis Document – Part 2 – TLAA Evaluation, Section 4.2.8, Core Shroud Reflood Thermal Shock Analysis	Revision 0

During the audit of the TLAA, the staff verified that Exelon has provided its basis that supports its disposition of 10 CFR 54.21(c)(1)(i).

During the audit of the operating experience associated with the TLAA, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects.

The staff also audited the description of the SLRA Core Shroud Reflood Thermal Shock Analysis provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA TLAA Section 4.2.9, “Core Plate Rim Hold-Down Bolt Loss of Preload Analysis”

Summary of Information in the Application. SLRA Section 4.2.9, “Core Plate Rim Hold-Down Bolt Loss of Preload Analyses,” discusses the applicant’s analyses for evaluating loss of preload in the tensioning force used to secure the core plate rim hold-down bolts (CPRH-DBs) used in the core plate assembly designs. Exelon dispositioned these analyses as TLAA and dispositioned the TLAAs in accordance with 10 CFR 54.21(c)(1)(i).

To verify that Exelon provided a basis to support its disposition of the TLAA (henceforth referred to as the CPRH-DB TLAA), the staff audited the TLAAs. The staff will address issues identified but not resolved in this report in the SER.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
SLRA Section 4.2.1.2	Reactor Vessel Internals Neutron Fluence Analyses TLAA	Revision 0
SLRA Sections 4.2.9	Core Plate Rim Hold-Down Bolt Loss of	Revision 0

Document	Title	Revision / Date
and A.4.2.9	Preload Analysis	
General Electric Letter (from Brian Frew) to the EPRI BWRVIP (Randy Stark)	Relaxation of Core Plate Rim Hold-down Bolts	June 26, 2006
EPRI Proprietary Report No. 107284	BWR Vessel and Internals Project, BWR Core Plate Inspection and Evaluation Guidelines (BWRVIP-25)	December 1996
NRC Letter and Safety Evaluation	Final Safety Evaluation of BWRVIP Vessel and Internals Project, "BWR Vessel and Internals Project, BWR Core Plate Inspection and Flaw Evaluation Guideline (BWRVIP-25," EPRI Report TR-107284, December 1996 (TAC No. M97802)	Dec. 19, 1999
NRC Letter and Safety Evaluation	Safety Evaluation for Referencing of BWR Vessel and Internals Project, BWR Core Plate Inspection and Flaw Evaluation Guidelines (BWRVIP-25) Report for Compliance with the License Renewal Rule (10 CFR Part 54) and Appendix B, BWR Core Plate Demonstration of Compliance with the Technical Requirements of the License Renewal Rule (10 CFR 54.21)	Dec. 7, 2000
GE Energy Nuclear Correspondence Letter	Relaxation of Core Plate Rim Hold-down Bolts	June 29, 2006
EPRI Proprietary Report No. 3002005594	BWRVIP-25, Revision 1: BWR Vessel and Internals Project, BWR Core Plate Inspection and Evaluation Guidelines[1]	December 1996

Notes: 1. Electric Power Research Institute (EPRI) Proprietary Report 3002005594 (BWRVIP-25, Revision 1) was submitted for staff review and approval in a letter to the NRC document control desk dated September 26, 2016. EPRI's proprietary responses to the requests for additional information (RAIs) issued on the BWRVIP-25, Revision 1, methodology were submitted to the staff October 12, 2018. At the time of the staff's audit of the TLAA, the methodology in the report and the responses to the RAIs on the methodology were still pending approval by the staff.

During the audit of the TLAA, the staff verified that Exelon has provided its basis that supports its disposition of 10 CFR 54.21(1)(i). However, the staff found that sufficient information was not available to complete its review of Exelon's basis for its TLAA disposition. In order to obtain the necessary information, the staff will consider issuing one or more requests for additional information (RAIs). The staff will document its evaluation of any potential RAIs issued on the topic of this TLAA in the SER.

During the audit the staff made the following observations:

- During the staff's review of SLRA Sections 4.2.1.2 and 4.2.9, and information in EPRI Proprietary Report BWRVIP-25, Revision 1, the staff observed that the applicant is basing its 10 CFR 54.21(c)(1)(i) disposition of the TLAA on a comparison to the proprietary CP-RHDB stress relaxation methodology specified in Appendix I of the EPRI BWRVIP-25, Revision 1, report and use of an CP-RHDB fluence value that has been averaged over the entire length of the bolts. The staff noted that the methodology in BWRVIP-25, Revision 1, is currently undergoing a staff review and has yet to receive staff approval or endorsement. Therefore, the staff observed that it may need further justification from the applicant as to why BWRVIP-25, Revision 1, provides an acceptable methodology and basis for projecting the analysis to the end of the subsequent period of extended operation.

During the audit of the operating experience associated with the TLAA, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff did not identify any additional aging effects (i.e., other than loss of preload due to neutron irradiation enhanced creep) that would impact the applicant's TLAA for the core plate rim hold-down bolts.

The staff also audited the description of the SLRA TLAA provided in the UFSAR supplement. The staff will evaluate the adequacy of the UFSAR supplement summary description of the TLAA in the staff's safety evaluation report for the application. This will include the staff evaluation of the applicant's basis for using an average neutron fluence value as the basis for dispositioning the TLAA under 10 CFR 54.21(c)(1)(i), rather than the peak 70 effective full power years (EFPY) fluence value reported for the bolts in SLRA Section 4.2.1.

SLRA TLAA Section 4.2.10, "Jet Pump Slip Joint Repair Clamp Loss of Preload Analysis"

Summary of Information in the Application. SLRA Section 4.2.10, "Jet Pump Slip Joint Repair Clamp Loss of Preload Analysis," discusses the analysis that was performed to assess potential drops in the preloaded tensioning forces of jet pump slip joint repair clamps (JPSJRCs) that were installed in Unit 2 in either 2004, 2008, or 2014. Exelon dispositioned the TLAA (henceforth referred to as the JPSJRC Preload TLAA) in accordance with 10 CFR 54.21(c)(1)(i).

To verify that Exelon provided a basis to support its disposition of the TLAA, the staff audited the TLAA.

Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
SLRA Section 4.2.1.2	Reactor Vessel Internals Neutron Fluence Analyses TLAA	Revision 0
SLRA Sections 4.2.10 and A.4.2.10	Jet Pump Slip Joint Repair Clamp Loss of Preload Analysis	Revision 0
EPRI Proprietary Report No.	BWRVIP-41, Revision 4: BWR Vessel and Internals Project, BWR Jet Pump Assembly	Revision 4, September 2014

	Inspection and	

Document	Title	Revision / Date
3002003093	Evaluation Guidelines (BWRVIP-41)	
NRC Letter (ADAMS ML18130A050)	Transmittal Letter Regarding Final Proprietary Safety Evaluation for Electric Power Research Institute Topical Report BWRVIP-41, Revision 4, “BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines (CAC No. MF4887; EPID L- 2014-TOP-0008)	June 26, 2018
NRC Proprietary Safety Evaluation (ADAMS ML18129A054; Non-proprietary SE is given in ML18130A024)	Final Proprietary Safety Evaluation for BWRVIP-41, Revision 4, “BWRVIP Jet Pump Assembly Inspection and Flaw Evaluation Guidelines	June 26, 2018
GE Nuclear Energy Proprietary Class III Report No. GENE- 0000-0031-1507-01	Jet Pump Slip Joint Clamp Repair, Structural Evaluation, Peach Bottom 2 & 3 and Limerick 1 & 2 Nuclear Power Stations	Revision 0, September 2004
Transware Enterprises Inc. Proprietary Report No. EXL-PB0-002- R-003	Peach Bottom Atomic Power Station Unit 2 Vessel Internal Components Fluence Evaluation Projection to 70 EFPY	Revision 1, March 2018

During the audit of the TLAA, the staff verified that Exelon has provided its basis that supports its disposition of 10 CFR 54.21(c)(1)(i).

During the audit the staff made the following observations:

- The staff reviewed SLRA Sections 4.2.1.2 and 4.2.10. The staff noted that, in SLRA Section 4.2.10, the applicant identifies that the TLAA is only applicable to JPSJRCs (i.e., a total of 11 repair clamps) that were installed in the Unit 2 jet pump assembly in either 2004, 2008, or 2014. The staff verified that the applicant has yet to install any JPSJRCs in the jet pump assembly of Unit 3.

- The staff verified that the applicant’s neutron fluence value for the JPSJRCs was calculated using EPRI BWRVIP’s RAMA software. The staff’s review of the neutron fluence TLAA for reactor internals in SLRA Section 4.2.1.2 will be performed, in part, to confirm whether the application of RAMA software technology is valid for calculating the neutron fluence values for these components at the end of the subsequent period of extended operation (i.e., at 70 EFPY.

During the audit of the operating experience associated with the TLAA, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff did not identify any additional aging effects or mechanisms (i.e., other than loss of preload due to neutron irradiation-enhanced stress relaxation) that would have an impact on the applicant’s evaluation of the TLAA.

The staff also audited the description of the SLRA's JPSJRC Preload TLAA provided in the UFSAR supplement. The staff noted that the UFSAR supplement summary description for the TLAA is consistent with the UFSAR supplement criteria provided in the SRP-SLR Section 4.7.2.2 for analyses that qualify as plant-specific TLAAs.

SLRA TLAA Section 4.2.11, "Jet Pump Auxiliary Spring Wedge Loss of Preload Analysis"

Summary of Information in the Application. SLRA Section 4.2.11, "Jet Pump Auxiliary Spring Wedge Loss of Preload Analysis," discusses the analysis that was performed to assess potential drops in the preloaded tensioning forces of jet pump auxiliary spring wedges (JPASWs) that were installed to provide lateral support for specified jet pump mixers whose design was modified to include the spring wedges. Exelon dispositioned the TLAA (henceforth referred to as the JPASW Preload TLAA) in accordance with 10 CFR 54.21(c)(1)(i).

To verify that Exelon provided a basis to support its disposition of the TLAA, the staff audited the TLAA.

Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
SLRA Section 4.2.1.2	Reactor Vessel Internals Neutron Fluence Analyses TLAA	Revision 0
SLRA Sections 4.2.11 and A.4.2.11	Jet Pump Auxiliary Spring Wedge Loss of Preload Analysis	Revision 0
EPRI Proprietary Report No. 3002003093	BWRVIP-41, Revision 4: BWR Vessel and Internals Project, BWR Jet Pump Assembly Inspection and Evaluation Guidelines (BWRVIP- 41)	Revision 4, September 2014
NRC Letter (ADAMS ML18130A050)	Transmittal Letter Regarding Final Proprietary Safety Evaluation for Electric Power Research Institute Topical Report BWRVIP-41, Revision 4, "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines (CAC No. MF4887; EPID L- 2014-TOP-0008)	June 26, 2018
NRC Proprietary Safety Evaluation (ADAMS ML18129A054; Non-proprietary SE is given in ML18130A024)	Final Proprietary Safety Evaluation for BWRVIP- 41, Revision 4, "BWRVIP Jet Pump Assembly Inspection and Flaw Evaluation Guidelines	June 26, 2018
GE Nuclear Energy	Peach Bottom 2, 3: Jet Pump Auxiliary Spring	Revision 0,

Proprietary Class III Report No. GENE- B13-02317-00-01	Wedge Stress Analysis	September 2001
Transware Enterprises Inc.	Peach Bottom Atomic Power Station Unit 2 Vessel Internal Components Fluence Evaluation	Revision 1, March 2018

Document	Title	Revision / Date
Proprietary Report No. EXL-PB0-002- R-003	Projection to 70 EFPY	[redacted]

During the audit of the TLAA, the staff verified that Exelon has provided its basis that supports its disposition of 10 CFR 54.21(c)(1)(i).

During the audit the staff made the following observations:

- The staff reviewed SLRA Sections 4.2.1.2 and 4.2.11. The staff noted that, in SLRA Section 4.2.11, the applicant identifies that the TLAA is only applicable to the following JPAFWs that were installed and remain in service as part of the inlet mixer assembly of the jet pumps: (a) Unit 2 jet pumps 10, 12, 14, 18, and 19, and (b) Unit 3 jet pumps 14.[2]

- The staff verified that the applicant's neutron fluence value for the JPASWs was calculated using EPRI BWRVIP's RAMA software. The staff's review of the neutron fluence TLAA for reactor internals in SLRA Section 4.2.1.2 will be performed, in part, to confirm whether the application of RAMA software technology is valid for calculating the neutron fluence values for these components at the end of the subsequent period of extended operation (i.e., at 70 effective full power years EFPY).

During the audit of the operating experience associated with the TLAA, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff did not identify any additional aging effects or mechanisms (i.e., other than loss of preload due to neutron irradiation-enhanced stress relaxation) that would have an impact on the applicant's evaluation of the TLAA.

The staff also audited the description of the SLRA's JASW Preload TLAA provided in the UFSAR supplement. The staff noted that the UFSAR supplement summary description for the TLAA is consistent with the UFSAR supplement criteria provided in the SRP-SLR Section 4.7.2.2 for analyses that qualify as plant-specific TLAAs.

SLRA TLAA Section 4.2.12, "Jet Pump Riser Repair Clamp Loss of Preload Analysis"

Summary of Information in the Application. SLRA Section 4.2.12, "Jet Pump Riser Repair Clamp Loss of Preload Analysis," discusses the analysis that was performed to assess potential drops in the preloaded tensioning forces of jet pump riser repair clamps

(JPRRCs) that were installed on the risers of two jet pump assemblies in 1998. Exelon dispositioned the TLAA (henceforth referred to as the JPRRC Preload TLAA) in accordance with 10 CFR 54.21(c)(1)(i). To verify that Exelon provided a basis to support its disposition of the TLAA, the staff audited the TLAA.

Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

[2] The Unit 2 JPASWs were installed in either 2004, 2006, or replaced in 2014. A JPASW was installed on Unit 2 jet pump 20 in 2006 but removed in 2014. The specified Unit 3 JPASW was installed in year 2001. An additional JPASW was installed on Unit 3 jet pump 09 in 2011, but was removed from the jet pump in 2017.

Relevant Documents Reviewed

Document	Title	Revision / Date
SLRA Section 4.2.1.2	Reactor Vessel Internals Neutron Fluence Analyses TLAA	Revision 0
SLRA Sections 4.2.12 and A.4.2.12	Jet Pump Riser Repair Clamp Loss of Preload Analysis	Revision 0
EPRI Proprietary Report No. 3002003093	BWRVIP-41, Revision 4: BWR Vessel and Internals Project, BWR Jet Pump Assembly Inspection and Evaluation Guidelines (BWRVIP-41)	Revision 4, September 2014
NRC Letter (ADAMS ML18130A050)	Transmittal Letter Regarding Final Proprietary Safety Evaluation for Electric Power Research Institute Topical Report BWRVIP-41, Revision 4, "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines (CAC No. MF4887; EPID L- 2014-TOP-0008)	June 26, 2018
NRC Proprietary Safety Evaluation (ADAMS ML18129A054; Non-proprietary SE is given in ML18130A024)	Final Proprietary Safety Evaluation for BWRVIP-41, Revision 4, "BWRVIP Jet Pump Assembly Inspection and Flaw Evaluation Guidelines	June 26, 2018
GE Nuclear Energy Proprietary Class III Report No. GENE- B13-01915-01	PECo Nuclear, Peach Bottom Atomic Power Station Unit 3, Structural Analysis, Jet Pump Riser Structural Enhancement	Revision 0, March 6, 1998
Transware Enterprises Inc. Proprietary Report No. EXL-PB0-002- R-003	Peach Bottom Atomic Power Station Unit 2 Vessel Internal Components Fluence Evaluation Projection to 70 EFPY	Revision 1, March 2018

During the audit of the TLAA, the staff verified that Exelon has provided its basis that supports the disposition of the TLAA in accordance 10 CFR 54.21(c)(1)(i).

During the audit the staff made the following observations:

- The staff reviewed SLRA Sections 4.2.1.2 and 4.2.12. The staff noted that, in SLRA Section 4.2.12, the applicant identifies that the TLAA is only applicable to JPRRCs that were installed on two jet pumps in Unit 3 in 1998 and remain in service: (a) the riser for Unit 3 jet pump pair 01/02, and (b) the riser for jet pump pair 13/14. The staff noted that these JPRRCs were installed to repair and structurally replace specific jet pump riser elbow-to-thermal sleeve welds that are known by the applicant to have applicable defects in them. The staff also noted that, in its TLAA, the applicant reports that these defects were detected as a result of in-service inspections that were performed on the specified jet pump riser welds in 1997.

- The staff verified that the applicant's neutron fluence value for the JPRRCs was calculated using EPRI BWRVIP's RAMA software. The staff's review of the neutron fluence TLAA for reactor internals in SLRA Section 4.2.1.2 will be performed, in part, to confirm whether the application of RAMA software technology is valid for calculating the neutron fluence values for these components at the end of the subsequent period of extended operation (i.e., at 70 effective full power years (EFPY)).

During the audit of the operating experience associated with the TLAA, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff did not identify any additional aging effects or mechanisms (i.e., other than loss of preload due to neutron irradiation-enhanced stress relaxation) that would have an impact on the applicant's evaluation of the TLAA.

The staff also audited the description of the SLRA's JRRC Preload TLAA provided in the UFSAR supplement. The staff noted that the UFSAR supplement summary description for the TLAA is consistent with the UFSAR supplement criteria provided in the SRP-SLR Section 4.7.2.2 for analyses that qualify as plant-specific TLAAs.

SLRA TLAA Section 4.2.13, "Replacement Core Plate Plug Extended Life Irradiation – Enhanced Stress Relaxation"

Summary of Information in the Application. SLRA Section 4.2.13, "Replacement Core Plate Plug Extended Life Irradiation –Enhanced Stress Relaxation," discusses the analyses for the extended life core support plugs. Exelon dispositioned the TLAA in accordance with 10 CFR 54.21(c)(1)(ii). To verify that Exelon provided a basis to support its disposition of the TLAA, the staff audited the TLAA. The staff will address issues identified but not resolved in this report in the SER.

Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
GE-NE-B13-02100-00-02	Stress Analysis for Extended Life Core Support Plugs for Exelon Nuclear Peach Bottom Atomic Power Station Units 2 & 3	Revision 0, June 2001
ER-PB-331-1001	Peach Bottom Reactor Pressure Vessel & Internals Program Basis and Implementation Document	Revision 6
GEH-004N2986	Peach Bottom Core Plate Plug Life Extension to 55 Years	Revision 1, June 07, 2017

During the audit of the TLAA, the staff verified that Exelon has provided its basis that supports its disposition of 10 CFR 54.21(c)(1)(ii). However, the staff found that sufficient information was not available to complete its review of Exelon's basis for its TLAA disposition. In order to obtain the necessary information, the staff will consider issuing an RAI. The staff will document its evaluation of this potential RAI in the SER.

During the audit, the staff made the following observations:

 The staff reviewed GE-NE-B13-02100-00-02 and noted that the document contains a description of the extended life core support plugs and mandrel spring.

 The staff reviewed GEH-004N2986 and noted that the document provides the initial installation pre-load and a reference used to determine the reduction in preload at 55 effective full power years (EFPY).

During the audit of the operating experience associated with the TLAA, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. No significant plant specific operating experience associated with TLAA Section 4.2.13 was noted by the staff during its review.

The staff also audited the description of the SLRA "Replacement Core Plate Plug Extended Life Irradiation – Enhanced Stress Relaxation" TLAA provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the SRP-SLR.

SLRA TLAA Section 4.2.14, "First License Renewal Application Core Shroud IASCC and Embrittlement Analysis"

Summary of Information in the Application. SLRA Section 4.2.14, "First License Renewal Application Core Shroud IASCC and Embrittlement Analysis," discusses the analyses for the core shroud. Exelon dispositioned the TLAA in accordance with 10 CFR 54.21(c)(1)(iii).

To verify that Exelon provided a basis to support its disposition of the TLAA, the staff audited the TLAA.

Audit Activities. During its audit, the staff reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PBAPS TLAA Technical Report	Peach Bottom Atomic Power Station Units 2 & 3 License Renewal Project, TLAA Technical Report	Revision 1
PB-PBD-AMP-XI.M9	Program Basis Document, BWR Vessel Internals	Revision 1
N/A	Potential TLAA – From Section 4.3.2.2 of First LRA	N/A

During the audit of the TLAA, the staff verified that Exelon has provided its basis that supports its disposition of 10 CFR 54.21(c)(1)(iii).

During the audit of the operating experience associated with the TLAA, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will document its review of relevant operating experience in the SER.

The staff also audited the description of the SLRA "First License Renewal Application Core Shroud IASCC and Embrittlement Analysis" TLAA provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the SRP-SLR.

SLRA TLAA Section 4.2.15, "Unit 3 Core Spray Replacement Piping Bolting Loss of Preload Evaluation"

Summary of Information in the Application. SLRA Section 4.2.15, "Unit 3 Core Spray Replacement Piping Bolting Loss of Preload Evaluation," discusses the analysis for the Unit 3 Core Spray Replacement Piping Bolting. Exelon dispositioned the TLAA in accordance with 10 CFR 54.21(c)(1)(ii).

To verify that Exelon provided a basis to support its disposition of the TLAA, the staff audited the TLAA.

Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-TLAABD, Part 1	Time-Limited Aging Analysis (TLAA) Basis Document – Part 1 – TLAA Identification	Revision 0
PB-TLAABD,	Time-Limited Aging Analysis (TLAA) Basis Document – Part 2 –	Revision 0

Part 2	TLAA Evaluation, Section 4.2.15, Unit 3 Core Spray Replacement Piping Bolting Loss of Preload Evaluation	

During the audit of the TLAA, the staff verified that Exelon has provided its basis that supports its disposition of 10 CFR 54.21(c)(1)(ii).

During the audit of the operating experience associated with the TLAA, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects.

The staff also audited the description of the SLRA Unit 3 Core Spray Replacement Piping Bolting Loss of Preload Evaluation provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the GALL-SLR Report.

SLRA TLAA Section 4.3.5, “Environmental Fatigue Analyses for RPV and Class 1 Piping”

Summary of Information in the Application. SLRA Section 4.3.5, “Environmental Fatigue Analyses for RPV and Class 1 Piping,” discusses the environmental fatigue analyses for reactor pressure vessel (RPV) and ASME Code Class 1 piping components. Exelon dispositioned the TLAA in accordance with 10 CFR 54.21(c)(1)(iii).

To verify that Exelon provided a basis to support its disposition of the TLAA, the staff audited the TLAA. The staff will address issues identified but not resolved in this report in the SER.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-TLAABD, Part 1	Time-Limited Aging Analysis (TLAA) Basis Document – Part 1 – TLAA Identification	Revision 0
PB-TLAABD, Part 2	Time-Limited Aging Analysis (TLAA) Basis Document – Part 2 – TLAA Evaluation, Section 4.3.5, Environmental Fatigue analysis for RPV and Class 1 Piping	Revision 0
SIR-99-091	Report on System Review and Recommendations or a Transient and Fatigue Monitoring System at Peach Bottom	Revision 0

	Atomic Power Station	
602-S-VC-23	Fatigue Analysis for Limiting Piping Components	Revision 0
1400630.301	60- and 80-Year Fatigue Projections	Revision 1
1400630.302	Peach Bottom Fatigue Usage Assessment	Revision 0
1400630.303	Peach Bottom Environmentally-Assisted Fatigue Screening	Revision 0
ER-AA-470	Fatigue and Transient Monitoring Program	Revision 7
ST-J-080-940-2	Reactor Pressure Vessel Fatigue Monitoring Record	Revision 9
SIR-99-122	Cycle Counting and Cycle-based Fatigue Report for the Transient and Fatigue Monitoring System for Peach Bottom Atomic Power Station Units 2 and 3	Revision 6

During the audit of the TLAA, the staff verified that Exelon has provided its basis that supports its disposition of 10 CFR 54.21(c)(1)(iii). However, the staff found that sufficient information was not available to complete its review of Exelon's basis for its TLAA disposition. In order to obtain the necessary information, the staff will consider issuing RAIs. The staff will document its evaluation of these potential RAIs in the SER.

During the audit of the operating experience associated with the TLAA, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff did not identify any additional aging effects that would have an impact on the Exelon's evaluation of the TLAA.

The staff also audited the description of the SLRA environmental fatigue analyses for RPV and Class 1 piping provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the SRP-SLR.

SLRA TLAA Section 4.3.6.1 "Generic BWR Fatigue Analyses for Various Reactor Vessel Internal Components"

Summary of Information in the Application. SLRA Section 4.3.6.1, "Generic BWR Fatigue Analyses for Various Reactor Vessel Internal Components," discusses the generic BWR fatigue analyses for reactor vessel internals. Exelon dispositioned the TLAA in accordance with 10 CFR 54.21(c)(1)(ii). To verify that Exelon provided a basis to support its disposition of the TLAA, the staff audited the TLAA. The staff will address issues identified but not resolved in this report in the SER.

Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-TLAABD,	Time-Limited Aging Analysis (TLAA) Basis Document – Part 1 –	Revision 0

Part 1	TLAA Identification	
PB-TLAABD, Part 2	Time-Limited Aging Analysis (TLAA) Basis Document – Part 2 – TLAA Evaluation, Section 4.3.6.1, Generic BWR Fatigue Analyses for Various Reactor Vessel Internal Components	Revision 0
NEDC-33566P	GE Hitachi Nuclear Energy Safety Analysis Report for Exelon Peach Bottom Atomic Power Station Units 2 and 3 Constant Pressure Power Uprate	Revision 0
NEDC-33873P	GE Hitachi Nuclear Energy Safety Analysis Report for Peach Bottom Atomic Power Station Units 2 and 3 Thermal Power Optimization	Revision 0

During the audit of the TLAA, the staff verified that Exelon has provided its basis that supports its disposition of 10 CFR 54.21(c)(1)(ii). However, the staff found that sufficient information was not available to complete its review of Exelon's basis for its TLAA disposition. In order to obtain the necessary information, the staff will consider issuing an RAI. The staff will document its evaluation of this potential RAI in the SER.

During the audit of the operating experience associated with the TLAA, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff did not identify any additional aging effects that would have an impact on the Exelon's evaluation of the TLAA.

The staff also audited the description of the SLRA generic BWR fatigue analyses for various reactor vessel internal components provided in the UFSAR supplement. The staff verified this description is consistent with the guidance in the SRP-SLR.

SLRA TLAA Section 4.3.6.2, "Generic BWR Fatigue Analyses for the Core Shroud"

Summary of Information in the Application. SLRA Section 4.3.6.2, "Generic BWR Fatigue Analyses for the Core Shroud," discusses the generic BWR fatigue analyses for the core shroud. Exelon dispositioned the TLAA in accordance with 10 CFR 54.21(c)(1)(i). To verify that Exelon provided a basis to support its disposition of the TLAA, the staff audited the TLAA.

Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-TLAABD, Part 1	Time-Limited Aging Analysis (TLAA) Basis Document - Part 1 - TLAA Identification	Revision 0

PB-TLAABD, Part 2	Time-Limited Aging Analysis (TLAA) Basis Document - Part 2 - TLAA Evaluation, Section 4.3.6.2, Generic BWR Fatigue Analyses for the Core Shroud	Revision 0
NEDC-33566P	GE Hitachi Safety Analysis Report for Exelon Peach Bottom Atomic Power Station Units 2 and 3 Constant Pressure Power Uprate	Revision 0
NEDC-33873P	GE Hitachi Safety Analysis Report for Peach Bottom Atomic Power Station Units 2 and 3 Thermal Power Optimization	Revision 0
004N2968	GE Hitachi Report on Peach Bottom Atomic Power Station Units 2 and 3 Shroud Fatigue Information	Revision 1

During the audit of the TLAA, the staff verified that Exelon has provided its basis that supports its disposition of 10 CFR 54.21(c)(1)(i).

During the audit of the operating experience associated with the TLAA, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff did not identify any additional aging effects that would have an impact on the Exelon's evaluation of the TLAA.

The staff also audited the description of the SLRA generic BWR fatigue analyses for the core shroud provided in the UFSAR supplement. The staff verified this description is consistent with the guidance in the SRP-SLR.

SLRA TLAA Section 4.3.6.3, "Core Shroud Support Fatigue Analysis Reevaluation"

Summary of Information in the Application. SLRA Section 4.3.6.3, "Core Shroud Support Fatigue Analysis Reevaluation," discusses the fatigue analysis for the core shroud support. Exelon dispositioned the TLAA in accordance with 10 CFR 54.21(c)(1)(iii). To verify that Exelon provided a basis to support its disposition of the TLAA, the staff audited the TLAA. The staff will address issues identified but not resolved in this report in the SER.

Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-TLAABD, Part 1	Time-Limited Aging Analysis (TLAA) Basis Document - Part 1 - TLAA Identification	Revision 0
PB-TLAABD, Part 2	Time-Limited Aging Analysis (TLAA) Basis Document - Part 2 - TLAA Evaluation, Section 4.3.6.3, Core Shroud Support Fatigue Analysis Reevaluation	Revision 0
SIR-98-030	Thermal Events at Peach Bottom Atomic Power Station	4/3/1998

NEDC-33566P	GE Hitachi Safety Analysis Report for Exelon Peach Bottom Atomic Power Station Units 2 and 3 Constant Pressure Power Uprate	Revision 0
NEDC-33873P	GE Hitachi Safety Analysis Report for Peach Bottom Atomic Power Station Units 2 and 3 Thermal Power Optimization	Revision 0
1400630.303	Peach Bottom Environmentally-Assisted Fatigue Screening	Revision 0

During the audit of the TLAA, the staff verified that Exelon has provided its basis that supports its disposition of 10 CFR 54.21(c)(1)(iii). However, the staff found that sufficient information was not available to complete its review of Exelon's basis for its TLAA disposition. In order to obtain the necessary information, the staff will consider issuing RAIs. The staff will document its evaluation of the potential RAIs in the SER.

During the audit of the operating experience associated with the TLAA, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff did not identify any additional aging effects that would have an impact on the Exelon's evaluation of the TLAA.

The staff also audited the description of the SLRA fatigue analysis for the core shroud support provided in the UFSAR supplement. The staff verified this description is consistent with the guidance in the SRP-SLR.

SLRA TLAA Section 4.3.6.5, "Replacement Steam Dryer Stress Report and Fatigue Evaluation"

Summary of Information in the Application. SLRA Section 4.3.6.5 discusses Exelon's TLAA for the fatigue analysis of the replacement reactor pressure vessel steam dryer. Exelon stated that the steam dryer at each Peach Bottom unit was replaced to support the extended power uprate (EPU) operation. Exelon stated that the replacement steam dryer was evaluated under the EPU condition in 2014 to ensure compliance with the structural design requirements of the 2007 Edition and 2008 Addenda of the ASME Code, Section III, Subsection NG. Exelon noted that, because the evaluation resulted in a calculated cumulative usage factor (CUF) value based on a specified number of design cycles and the number of cycles assumed for design transients over license term, this calculated CUF is considered as TLAA and that the re-evaluation is required for the subsequent period of extended operation. Exelon dispositioned this TLAA in accordance with 10 CFR 54.21(c)(1)(i). To verify that Exelon provided a basis to support its disposition of the TLAA, the staff audited the TLAA.

Audit Activities. During its audit, the staff reviewed onsite documentation provided by Exelon. The table below lists the documents that were reviewed by the staff and were found relevant to the audit.

Relevant Documents Reviewed

Document	Title	Revision / Date

PB-TLAABD, Part 1	TLAA Basis Document – Part 1 – TLAA Identification	Revision 0 / September 2017
PB-TLAABD, Part 2	TLAA Basis Document – Part 2 – TLAA Evaluation, Section 4.3.6.5, Replacement Steam Dryer Stress Report and Fatigue Evaluation	Revision 0
Package ADAMS Accession No. ML122860201	License Amendment Request (LAR) - Extended Power Uprate (EPU) for Peach Bottom, Units 2 and 3	09/28/2012
EPRI Report No. 3002010541	BWRVIP-139, Revision 1-A: BWR Vessel and Internals Project, Steam Dryer Inspection and Flaw Evaluation Guidelines	11/2017

During the audit of the TLAA, the staff verified that Exelon has provided its basis that supports its disposition of 10 CFR 54.21(c)(1)(i).

During the audit, the staff made the following observations:

 The staff reviewed the TLAA basis document including the LAR document associated with both the EPU operation and the steam dryers' replacement activity at Peach

Bottom. The staff noted that the replacement steam dryers were evaluated for compliance with the structural design requirements of the ASME Code, Section NG, under the EPU condition. The staff verified that the CUF calculated using the assumed transient cycles met the ASME Code limit. The staff noted that, consistent with the acceptance criteria defined in Section 4.3.2.1.1.1 of SRP-SLR and the review procedures defined in Section 4.3.3.1.1.1 of the SRP-SLR, this provides sufficient demonstration that the TLAA is acceptable in accordance with the acceptance criterion defined in 10 CFR 54.21(c)(1)(i). This staff determination will be reflected in the staff's safety evaluation report for the Peach Bottom SLRA.

During the audit of the operating experience associated with the TLAA, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff did not identify any additional aging effects that would have an impact on the Exelon's evaluation of the TLAA.

The staff also audited the description of the SLRA Replacement Steam Dryer Stress Report and Fatigue Evaluation provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the SRP-SLR for TLAAs.

SLRA TLAA Section 4.4, "Environmental Qualification of Electric Equipment"

Summary of Information in the Application. SLRA Section 4.4, "Environmental Qualification of Electric Equipment," discusses the thermal, radiation, and cyclical analyses for plant electrical and I&C components. Exelon dispositioned the TLAAs in accordance with 10 CFR 54.21(c)(1)(iii). To verify that Exelon provided a basis to support its disposition of the TLAA, the staff audited the TLAA.

Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-PBD-AMP- X.E1	Environmental Qualification of Electric Equipment – Program Basis Document	Rev. 1
AR 4106712-09	Aging Management Program (AMP) Effectiveness Review - Peach Bottom Environmental Qualification Activities AMP	Rev. 1
CC-AA-203	Environmental Qualification Program	Rev.1
EQ-PB-011	Environmental Qualification - Okonite 600 V Power & Control Cable and 5 kV Power Cable	Rev. 1
EQ-PB-016	Environmental Qualification - Brand Rex Cable	Rev. 1
PB-TLAABD	Peach Bottom Atomic Power Station Units 2 and 3 License Renewal Project – TLAA Basis Document – Part 2 – TLAA Evaluation	Rev. 0

During the audit of the TLAA, the staff verified that Exelon has provided its basis that supports its disposition of 10 CFR 54.21(c)(1)(iii).

During the audit of the operating experience associated with the TLAA, the staff independently searched the plant specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER.

The staff also audited the description of the SLRA TLAA, "Environmental Qualification of Electric Equipment," provided in the UFSAR Supplement A.4.4.1. The staff verified this description is consistent with the description provided in the SRP-SLR for TLAAs.

SLRA TLAA Subsection 4.6.1, "Primary Containment Structures, Penetrations, and Associated Components with Fatigue Analyses"

Summary of Information in the Application. SLRA Section 4.6, "Primary Containment Fatigue Analyses" and Subsection 4.6.1, "Primary Containment Structures, Penetrations, and Associated Components with Fatigue Analyses" discuss the analyses for the Peach Bottom Atomic Power Station (PBAPS) Units 2 and 3 Torus Shell, Torus Penetrations, Torus Vents, Safety Relief Valve (SRV) Discharge Piping, Other Piping Attached to the Torus, Drywell-to- Torus Vent Bellows, Replacement RHR and Core Spray Suction Strainers. Exelon dispositioned the TLAAs in accordance with 10 CFR 54.21(c)(1)(iii). To verify that Exelon provided a basis to support its disposition of the TLAA, the staff audited the TLAA. The staff will address issues identified but not resolved in this report in the SER.

Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-TLAABD, Part 1	PBAPS Time-Limited Aging Analysis (TLAA) Basis Document – Part 1 - TLAA Identification	Revision 0
PB-TLAABD, Part 2	PBAPS TLAA Basis Document – Part 2 - TLAA Evaluation SLRA Section 4.6, Primary Containment Fatigue Analyses	Revision 0
PBAPS SLRA	Section 4.6.1, Primary Containment Structures, Penetrations and Associated components with fatigue Analyses	Revision 0
PBAPS SLRA	Section B.3.1.1, Fatigue Monitoring	Revision 0
PBAPS SLRA	Section 3.5.2.2.1.5, Cumulative Fatigue Damage	Revision 0
PBAPS SLRA	Sections 3.5.2.2.1.3, Loss of Material Due to General, Pitting and Crevice Corrosion	Revision 0
PBAPS SLRA	Section B.2.1.30, ASME Section XI, Subsection IWE	Revision 0

Document	Title	Revision / Date
PBAPS UFSAR	Appendix Q.5.4.1, "Fatigue Analyses of Containment Pressure Boundaries: Analysis of Tori, Torus Vents, and Torus Penetrations	Revision 26
EXLNPB113- REPT-001	Review of Containment Fatigue Analyses for Peach Bottom Second License Renewal	Revision 0
Addendum 2 to Revision 1 to Spec No. NE- 265	Nuclear Safety Related Specifications for ECCS Suction Strainers for the Limerick Generating Station Units 1 and 2 and Peach Bottom Atomic Power Station Units 2 and 3	February 1998
PM-1006	RHR Strainer Supports	Revision 2
PM-1004	Core Spray Strainer Supports	Revision 2
10104-22-0	Sargent and Lundy Design Report, Unit 2 ECCS Pump Suction Strainer – Ring Girder Stiffeners	December 1998
10104-22-01	Sargent and Lundy Design Report "ECCS Pump Suction Strainer Ring Girder	May 1998
PBM-040	CSC (Containment Suppression Chamber) Modification – Fatigue Evaluation of Torus	Revision 2
PBM-024	Fatigue Evaluation for Vent System for LOCA	December

		1998
P-1-Q-614	Mark I Long-Term Program Plant Unique Analyses	Revision 1A and Revisions 0, 1, 2
MISC-ME-DR- 040	PBAPS ECCS Suction Strainer Assemblies	Revision 5

During the audit of the TLAA, the staff verified that Exelon has provided its basis that supports disposition in accordance with 10 CFR 54.21(c)(1)(i), (ii), or (iii). However, the staff found that sufficient information was not available to complete its review of Exelon's basis for its TLAA disposition. In order to obtain the necessary information, the staff will use the voluntary SLRA supplement information committed to by Exelon during the audit or consider issuing RAIs. The staff will document its evaluation of the supplement and potential RAI(s) in the SER.

During the audit, the staff made the following observations:

- The staff reviewed PBAPS EXLNPB113-REPT-001, "Review of Containment Fatigue Analyses for Peach Bottom Second License Renewal," a document relevant to Section 4.6 of the PBAPS SLRA, and noted that it states that generic fatigue evaluations/waivers may be considered for the Torus Electrical Penetration Assemblies, Drywell Shell, and Drywell Head. The staff also reviewed PBAPS SLRA PB-TLAABD, Part 2, and noted that it references ASME Section III, Subsections A and B, 1965, which in its Subsection N-415.1 states that an analysis for cyclic operation is not required if specific operation conditions are met (e.g., the number of cycles related to the cycling of vessel pressure from atmospheric to

operating, the number of specified significant pressure fluctuations, temperature difference between any two points during normal, startup, and shutdown operations, etc.). The staff could not locate any applicable waivers for fatigue parameter evaluations for the noted plant operating conditions as discussed in PBAPS EXLNPB113-REPT-001. It is not clear how PBAPS would meet the evaluations of those activities for waiver of fatigue for Torus Electrical Penetration Assemblies, Drywell Shell, Drywell Head, or any other primary containment structure, penetration, and associated component subject to fatigue waiver conditions.

- The staff reviewed PBAPS SLRA Section 4.6.1 and noted that it states that two monitored locations (i.e., Torus (CS)/Torus Shell and Torus Penetrations (CS)/Torus Shell) are bounding the design CUFs. For the "SRV Discharge Piping" and "Other Piping Attached to the Torus," with CUF 0.202 and the "Replacement RHR" and various subcomponents of "Core Spray Suction Strainers," 0.193 and 0.367, Section 4.6.1 of the SLRA dispositions these as 10 CFR 54.21(c)(1)(iii). The staff also reviewed Appendix Q.5.4.1, "Fatigue Analyses of Containment Pressure Boundaries: Analysis of Tori, Torus Vents, and Torus Penetrations," Revision 26 of the UFSAR, which states that locations of low usage factor (< 0.4) are dispositioned per 10 CFR 54.21(c)(1)(i). It is unclear why these low CUF locations, dispositioned in the SLRA in accordance with 10 CFR 54.21(c)(1)(iii), are inconsistent with Appendix Q.5.4.1 of the UFSAR. It is also unclear how the two identified monitoring locations could adequately assess the number and severity of loading cycles from thermal, pressure, and seismic transients for the “SRV Discharge Piping” and “Other Piping Attached to the Torus” and “Replacement RHR” and various components of “Core Spray Suction Strainers” during the SPEO.

During the audit of the operating experience associated with the TLAA, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER.
The staff will use the voluntary SLRA supplement information committed to by Exelon during the audit or consider issuing RAI(s) in order to obtain the information necessary to determine whether Exelon’s SLRA TLAA Section 4.6, “Primary Containment Fatigue Analyses,” and SLRA TLAA Subsection 4.6.1, “Primary Containment Structures, Penetrations, and Associated Components with Fatigue Analyses,” can be adequate to manage the associated aging effects. The staff will document its evaluation of the supplement and potential RAI(s) in the SER.

During the audit, the staff made the following observations:

 The staff reviewed Section 4.6 and Subsection 4.6.1 of the PBAPS SLRA and noted that the bounding design CUFs for PBAPS Torus Shell and Penetrations are 0.942 and 0.992 respectively. Table 4.3.1-3 of the SLRA assigns CUF values of 0.862 and 0.591 respectively for the two monitored locations. Section 4.3.1, “Transient Cycle and Cumulative Usage Projections for 80 Years,” of the SLRA, however, states that PBAPS has experienced a declining trend in transient accumulation over time, and the trend provides an accurate basis for future transient projections where each transient was evaluated to determine if the recent 15-year trend had a consistent cycle accumulation rate. It is not clear whether PBAPS used the 15-year declining rate for most transients to extrapolate the projected number of future occurrences beginning January 1, 2016, and ending at the end of the units’ 80-year life, and that then resulted in CUF reductions of 15 and 40 percent, respectively. In addition, Sections 3.5.2.2.1.3, “Loss of Material Due to General, Pitting and Crevice Corrosion,” and B.2.1.30, “ASME Section XI, Subsection IWE,” discuss an underwater examination that identified a local area of pitting/general corrosion with 0.126

inches of metal loss of the nominal 0.675-inch-thick torus shell plate. It is not clear whether loss of material (corrosion fatigue) was considered in the projected CUF evaluations for the selected location, and, if it occurs, what measures PBAPS plans to take for loss of material that potentially could reduce the fatigue life of affected components.

The staff also audited the description of the SLRA TLAA Section 4.6, “Primary Containment Fatigue Analyses,” and Subsections 4.6.1 “Primary Containment Structures, Penetrations, and Associated Components with Fatigue Analyses,” provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the SRP-SLR.

SLRA TLAA Subsection 4.6.2, “Containment Process Line Penetration Bellows”

Summary of Information in the Application. SLRA Section 4.6, “Primary Containment Fatigue Analyses,” Subsection 4.6.2, “Containment Process Line Penetration Bellows” discusses the analyses for the PBAPS Units 2 and 3 containment penetration bellows. Exelon dispositioned the TLAAs in accordance with 10 CFR 54.21(c)(1)(i). To verify that Exelon provided a basis to support its disposition of the TLAA, the staff audited the TLAA. The staff will address issues identified but not resolved in this report in the SER.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-TLAABD, Part 1	PBAPS, Time-Limited Aging Analysis (TLAA) Basis Document - Part 1 - TLAA Identification	Revision 0
PB-TLAABD, Part 2	PBAPS TLAA Basis Document Basis Document - Part 2 - TLAA Evaluation SLRA Section 4.6, Primary Containment Fatigue Analyses	Revision 0
PBAPS SLRA	Section 4.6.2, Containment Process Line Penetration Bellows	Revision 0
PBAPS UFSAR	Appendix M, Containment Report	Revision 26
EXLNPB113- REPT-001	Review of Containment Fatigue Analyses for Peach Bottom Second License Renewal	Revision 0
1400630.301	PBAPS Second License Renewal (SLR), 60 and 80 year Cycle and fatigue Projections, Structural Integrity, Associates, Inc.	Revision 1
Design Specification 1187-P-314(Q)	Design Specification for Replacement Containment Expansion Joints for Nuclear Service for the PBAPS	Revision 6

Document	Title	Revision / Date
Specs 6280-M- 122	Specification for Containment Expansion Joints for the PBAPS	Revision 5

During the audit of the TLAA, the staff verified that Exelon has provided its basis that supports disposition in accordance with 10 CFR 54.21(c)(1)(i), (ii), or (iii). However, the staff found that sufficient information was not available to complete its review of Exelon’s basis for its TLAA disposition. In order to obtain the necessary information, the staff will use the voluntary SLRA supplement information committed to by Exelon during the audit or consider issuing RAI(s). The staff will document its evaluation of the supplement and potential RAI(s) in the SER.

- During the audit, the staff noted that Section 4.6, Subsection 4.6.2, of the PBAPS SLRA states that "[t]he design specification for the original bellows specified 200 'startup-shutdown' cycles (as defined in [ASME Code] Section III) and a minimum of 1,500 'normal operating' cycles (as defined in [ASME Code] Section III)." It also states that the Unit 3 RHR supply and return line penetration bellows were replaced during 1988 and 1989; however, "[t]he design specification for the [Unit 3] replacement penetration bellows specified 1,500 normal operating cycles" but did not specify 200 startup-shutdown cycles. In addition, it states that over an 80-year period Units 2 and 3 are projected to experience 186 and 140 "Heatup-Cooldown" transient cycles, respectively, "which are less than the specified 200 startup-shutdown transient cycles for the original containment bellows." It also states that for "both the original and replaced containment bellows, the specified 1500 'normal operating cycles' associated with a DBA is significantly greater than an assumed one DBA per unit." The section then concludes by stating that the "primary containment process line bellows fatigue analyses remain valid through the second period of extended operation" and dispositions these TLAAs per 10 CFR 54.21(c)(1)(i).
- The audited PB-TLAABD, Part 2, references "Specification 6280-M-122, Specification for Containment Expansion Joints for the Peach Bottom Atomic Power Station Units 2 and 3," dated January 6, 1969, and "Design Specification for Replacement Containment Expansion Joints for Nuclear Service," dated September 2, 1987, confirm PBAPS SLRA Section 4.6.2 statements for cyclic loading of bellows. The "Design Specification for Replacement Containment Expansion Joints for Nuclear Service," however, states the "effects of relative end point displacement[s] resulting from thermal and seismic movements shall be considered in the fatigue evaluation" of the bellows. It is not clear whether the applicant replaced the bellows assemblies at both units, or just at Unit 3. If only bellows at Unit 3 were replaced, it is not clear whether their design satisfies the design basis cyclic loading of anticipated severities and number of cycles. It is also not clear how "relative end point displacement[s] resulting from thermal [...] movements" would account for startup-shutdown cyclic loadings required in the original design consistent with ASME code Section III, Paragraph N-412 (n)(1) and (n)(3), noted in Specification 6280-M-122. In addition, it is not clear how PBAPS equivalences the severities and the numbers of cycles associates with DBAs to those of normal cycles of operation.

  During the audit of the operating experience associated with the TLAA, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER.

The staff also audited the description of the SLRA TLAA Section 4.6, "Primary Containment Fatigue Analyses," Subsection 4.6.2, "Containment Process Line Penetration Bellows," provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the SRP-SLR.

TLAA Section 4.7.1, “Crane Cyclic Loading Analyses”

Summary of Information in the Application. SLRA Section 4.7.1, “Crane Cyclic Loading Analyses,” discusses the analyses for the following cranes:

- reactor building cranes
- emergency diesel generator bridge cranes
- turbine building cranes
- circulating water pump structure cranes

Exelon dispositioned the TLAAs in accordance with 10 CFR 54.21(c)(1)(i). To verify that Exelon provided a basis to support its disposition of the TLAA, the staff audited the TLAA. The staff will address issues identified but not resolved in this report in the SER.

Audit Activities. During its audit, the staff interviewed Exelon’s staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-TLAABD	TLAA Basis Document – Part 1 –TLAA Identification	Revision 0
PB-TLAABD	TLAA Basis Document – Part 2 –TLAA Identification	Revision 0
Specification 6280-M- 13B	Specification for Reactor Building Cranes for PBAPS	Revision 1
Specification 6280-M- 13A	Specification for Turbine Building Cranes for PBAPS	Revision 2
MA-PB-763-415	High Pressure Turbine Disassemble and Inspection	Revision 4
ANSI B30.2	Overhead and Gantry Cranes	1976
MA-PB-716-021	Rigging and Handling of Heavy Loads	Revision 0
Specification 6280-M- 25	Specification for Miscellaneous Bridge and Jib Cranes for PBAPS Units 2 and 3	Revision 2
Specification 6280-M- 24A	Intake Structure Crane for PBAPS Units 2 and 3	Revision 1
NUREG-1769	Safety Evaluation Report Related to the License Renewal of Peach Bottom Atomic Power Station, Units 2 and 3	March 2003
PBAPS UFSAR	Updated Final Safety Analysis Report	Revision 26
CMAA-70	Specification for Electric Overhead Traveling Cranes	1975

Document	Title	

		Revision / Date
CMAA-70	Specification for Top Running Bridge and Gantry Type Multiple Girder Electric Overhead Traveling Cranes	1999

During the audit of the TLAA, the staff verified that Exelon has provided its basis that supports its disposition of 10 CFR 54.21(c)(1)(i). However, the staff found that sufficient information was not available to complete its review of Exelon's basis for its TLAA disposition. In order to obtain the necessary information, the staff will consider issuing an RAI. The staff will document its evaluation of this potential RAI in the SER.

During the audit, the staff made the following observation:

- The staff reviewed the SRP-SLR Sections 4.7.1 and Table 4.7.1-2 and noted that, for the turbine building cranes, the number of cycles projected for 80 years of operation is 7,340 cycles. The staff reviewed table 4.7.1-2 and added the listed expected number of lifts over 80 years for each load, and it noted that the total number of lift cycles may be 1,140 instead of 7,340. The staff will consider issuing an RAI in order to obtain the information necessary to verify the correct number of total lift cycles for the turbine building cranes.

During the audit of the operating experience associated with the TLAA, the staff independently searched the plant-specific database to identify any previously unknown or recurring aging effects. The staff will evaluate the identified plant-specific operating experience in the SER. The staff also audited the description of the crane cyclic loading TLAAs provided in the UFSAR supplement. The staff verified this description is consistent with the description provided in the SRP-SLR.

SLRA TLAA 4.7.4, "Fracture Mechanics Analysis of ISI Reportable Indications For Group I Piping: As Forged Laminar Tear in a Unit 3 Main Steam Elbow Near Weld 1-B-3BC-LDO Discovered During Preservice UT"

Summary of Information in the Application. SLRA Section TLAA 4.7.4 describes the Fracture Mechanics Analysis of ISI-Reportable Indications for Group I Piping: As-Forged Laminar Tear in a Unit 3 Main Steam Elbow Near Weld 1-B-3BC-LDO Discovered During Preservice UT. The staff did not perform an in-house audit of this TLAA because the applicant's basis in SLRA Section 4.7.4 for dispositioning the TLAA in accordance with 10 CFR 54.21(c)(1)(ii) was sufficient for the staff's review without the need for an audit of background information on the TLAA basis.

SLRA TLAA 4.7.5, "Unit 3 Core Spray Replacement Piping Fatigue and Leakage Assessment"

Summary of Information in the Application. SLRA Section TLAA 4.7.5 describes the Unit 3 Core Spray Replacement Piping Fatigue and Leakage Assessment. The staff did not perform an in- house audit of this TLAA because the applicant's basis in SLRA Section 4.7.5 for dispositioning the TLAA in accordance with 10 CFR 54.21(c)(1)(i) was sufficient for the staff's review without the need for an audit of background information on the TLAA basis.

2.3 Further Evaluations

SLRA AMR Further Evaluation 3.5.2.2.2.6, "Reduction of Strength and Mechanical Properties of Concrete Due to Irradiation"

Summary of Information in the Application. During the audit, the staff reviewed plant documentation associated with the following:

 SLRA Section 3.5.2.2.2.6 (AMR 3.5.1-097) addresses the further evaluation for the aging effect of reduction of strength and mechanical properties of concrete due to irradiation.

The table below lists the documents that were reviewed by the staff and were found relevant to the review of this item. These documents were provided by Exelon.

Relevant Documents Reviewed

Document	Title	Revision / Date
PBAPS UFSAR	Updated Final Safety Analysis Report	Revision 26
EPRI Report 3002002676	Expected Condition of Reactor Cavity Concrete after 80 Years of Radiation Exposure	February 2014
EPRI Report 3002008128	Structural Disposition of Neutron Radiation Exposure in BWR Vessel Support Pedestals	July 2016
PM-0832	Radiation Through Bioshield Wall and Streaming Through Penetrations	February 24, 2015
Drawing S-191	Reactor Pedestal and Sacrificial Shield Development Unit 2	Revision 16
Drawing S-192	Reactor Pedestal and Sacrificial Shield Section Unit 2	Revision 14
Drawing S-199	Drywell Interior Platforms Plan Elevation 13'-0" & 154'-9"	Revision 25
EPRI Report 3002014882	An Assessment of the Integrity of BWR Vessel Structural Steel Supports for Long-Term Operations	December 2018

During the audit, the staff made the following observations:
The staff reviewed PBAPS UFSAR Section C.4.6, and noted that it states the following:

 The sacrificial shield was designed without considering the concrete for any structural purpose, except the lower 10 ft. of the wall. The forces considered were: seismic forces, pipe loading, pipe restraints, platform loads, and jet load reaction. The 27-in thick cylindrical structure consists of 12 steel columns equally spaced and continually tied by a 1/4-in thick steel plate on the inside and outside of the columns.

The staff notes that steel components near the reactor pressure vessel (RPV) could be subject to the aging effect of loss of fracture toughness due to embrittlement caused by radiation. Based on the information in the UFSAR stated above, a review of site drawings S-191, S-192, and S-199 and interviews with Exelon's personnel, the staff noted that there are several steel

components, such as steel columns, steel plates, and RPV lateral restraints that are part of the shield wall structure. The staff notes that these structural steel components may be subject to the aging effect of loss of fracture toughness due to radiation embrittlement; however, the SLRA does not address this aging effect for these components. The staff may need additional information regarding the susceptibility of these components to this aging effect. During a teleconference on January 22, 2019, the applicant stated that it would supplement the SLRA to address this aging effect. The staff's review of the SLRA supplement will be documented in SER Section 3.5.2.2.2.6.

SLRA AMR Further Evaluation 3.6.2.2.3, "Loss of Material Due to Wind-Induced Abrasion, Loss of Conductor Strength Due to Corrosion, and Increased Resistance of Connection Due to Oxidation or Loss of Preload for Transmission Conductors, Switchyard Bus, and Connections"

Summary of Information in the Application. During the audit, the staff reviewed plant documentation associated with the following:

- SLRA Table 3.6.2 item corresponding to SLRA Table 3.6.1-004, "transmission conductors" composed of aluminum, and steel exposed to air-outdoor
- SLRA Table 3.6.2 item corresponding to SLRA Table 3.6.1-005, "transmission connectors" composed of aluminum, and steel exposed to air-outdoor
- SLRA Table 3.6.2 item corresponding to SLRA Table 3.6.1-006, "switchyard bus and connections" composed of aluminum, copper, bronze, stainless steel, and galvanized steel exposed to air-outdoor
- SLRA Table 3.6.2 item corresponding to SLRA Table 3.6.1-007, "transmission conductors" composed of aluminum, and steel exposed to air-outdoor

Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

Relevant Documents Reviewed

Document	Title	Revision / Date
N/A	Peach Bottom Atomic Power Station, Units 2 and 3 Screening Report – Electrical Commodities	Revision 1

R1272069	Recurring Task Work Order – Perform Thermography on Start Up	22/09/2015
PB-AMRBD-MEAE	Peach Bottom Atomic Power Station, Units 2 and 3 – Materials, Environments, and Aging Effects Aging Management Review Basis Document	Revision 2
6280 E-1	Single Line Diagram	Revision 57
EPP-4036	PECO Substation Rigid Bus	04/15/2009
EPP-2030	Engineering Practice –Overhead Transmission Line Weather and Mechanical Design Conditions	11/09/2004

The staff reviewed Exelon's further evaluation 3.6.2.2.3, "Transmission Conductors, Switchyard Bus, and Connections." This input will be used in SER Section 3.6.2.2.3

SLRA AMR Further Evaluation 3.6.2.3.2, "High-Voltage Electrical Insulators"

Summary of Information in the Application. During the audit, the staff reviewed plant documentation associated with the following:

- SLRA Table 3.6.2, item corresponding to SLRA Table 3.6.1-002, "high-voltage electrical

  insulators" composed of porcelain, malleable iron, aluminum, galvanized steel, and

  cement exposed to air-outdoor

- SLRA Table 3.6.2, item corresponding to SLRA Table 3.6.1-003, "high-voltage electrical

  insulators" composed of porcelain, malleable iron, aluminum, galvanized steel, and cement exposed to air-outdoor

  Audit Activities. During its audit, the staff interviewed Exelon's staff and reviewed documentation provided by Exelon. The staff reviewed the following relevant documents.

  Relevant Documents Reviewed

Document	Title	Revision / Date
PB-AMRBD-MEAE	Peach Bottom Atomic Power Station, Units 2 and 3 – Materials, Environments, and Aging Effects Aging Management Review Basis Document	Revision 2

6280 E-1	Single Line Diagram	Revision 57
N/A	Peach Bottom Atomic Power Station, Units 2 and 3 Screening Report – Electrical Commodities	Revision 1
R1272069	Recurring Task Work Order – Perform Thermography on Start Up	22/09/2015

During the audit, the staff made the following observation:

 The staff reviewed Exelon's further evaluation 3.6.2.3.2 – High-Voltage Electrical Insulators and noted that SLRA concluded that no AMP is required for these components. During a breakout session with Exelon, the staff discussed the operating experience as well as predictive maintenance performed at Peach Bottom. Exelon subsequently revised the SLRA and added additional discussions for the technical basis of the conclusion. This input will be used in SER Section 3.6.2.3.2.

2.4 Scoping and Screening Methodology and Results The following SLRA Sections were audited:

2.1 “Scoping and Screening Methodology
2.3 “Scoping and Screening Results: Mechanical 2.4 “Scoping and Screening Results: Structures 2.5 “Scoping and Screening Results: Electrical

Summary of Information in the Application. The SLRA Section 2.1 “Scoping and Screening Methodology” states in part:

The initial step in the scoping process was to define the entire plant in terms of systems and structures. Each of these systems and structures were evaluated

against the scoping criteria in 10 CFR 54.4(a)(1), (a)(2), and (a)(3), to determine if the system or structure performs or supports a safety-related intended function, if system or structure failure could prevent the satisfactory accomplishment of a safety-related function, or if the system or structure performs functions that demonstrate compliance with the requirement of one of the five second license renewal regulated events. The intended function(s) that are the bases for including systems and structures within the scope of second license renewal were also identified.

A mechanical system was included within the scope of second license renewal if any portion of the system met the scoping criteria of 10 CFR 54.4.

A structure was included within the scope of second license renewal if any portion of the structure met the scoping criteria of 10 CFR 54.4. Structures were then further evaluated to determine those structural components that are required to perform or support the identified structure intended function(s).

Electrical and Instrumentation and Control (I&C) systems were scoped like mechanical systems and structures per the scoping criteria in 10 CFR 54.4(a)(1), (a)(2), and (a)(3). Electrical and I&C components within the in scope electrical and I&C systems were included within the scope of second license renewal.

To verify this approach, the staff audited the above listed SLRA Sections.

Audit Activities. During the NRC audit of the scoping and screening methodology and results, the staff focused on those systems identified on the Peach Bottom PRA Risk Summary. The staff reviewed SLRI documentation provided by the applicant on the on-line portal.

The table below lists the documents that were reviewed by the staff and were found relevant to the audit.

Relevant Documents Reviewed

Document	Title	Revision / Date
PB-SSBD-A1	10 CFR 54.4(a)(1) Safety Related Systems	Rev. 2
PB-SSBD-A2	10 CFR 54.4(a)(2) System Scoping Criteria	Rev. 1
PB-SSBD-AOT	Abnormal Operational Transients	Rev. 2
PB-SSBD-ATWS	10 CFR 54.4(a)(3) ATWS Systems	Rev. 1
PB-SSBD-EQ	10 CFR 54.4(a)(3) Environmental Qualification Systems	Rev. 2
PB-SSBD-FP	Fire Protection	Rev. 1
PB-SSBD-SBO	Station Blackout	Rev. 2

During the audit, the staff reviewed the subsequent license renewal scoping and screening results, procedures and reports, and interviewed the applicant’s staff during breakout sessions.

3. Applicant Personnel Contacted During Audit

Name	Affiliation
David Distel, PM	Exelon
Paul Weymuller	Exelon
Donald Warfel	Exelon
Peter Tamburro	Exelon
Leah Ritz	Exelon
Michael Baker	Exelon
James Annett (now retired)	Exelon
Scott Kauffman	Exelon

Mark Miller	Exelon
Michael Coakley	Exelon
John Hufnagel (now retired)	Exelon
Albert Piha	Exelon
Mary Kowalski	Exelon
Benjamin Jordan	Exelon

4. Exit Meeting

An exit meeting was held with the applicant on April 29, 2019, to discuss the results of the in-office regulatory audit. The staff is considering the issuance of RAIs and requests for confirmation of information to support the completion of the staff's SLRA review.

September 27, 2019 – Letter dated September 27, 2019 from Lisa Regner, Acting Branch Chief Plant Licensing Branch III Division of Operating Reactor Licensing Office of nuclear Reactor Regulation to Bryan Hanson, Senior Vice President, Exelon Generation Company, President and Chief Nuclear Officer Exelon Nuclear with the subject of BRAIDWOOD STATION, UNITS 1 AND 2; BYRON STATION, UNIT NOS. 1 AND 2; CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 AND 2; CLINTON POWER STATION, UNIT NO. 1; DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3; JAMES A. FITZPATRICK NUCLEAR POWER PLANT; LASALLE COUNTY STATION, UNITS 1 AND 2; LIMERICK GENERATING STATION, UNITS 1 AND 2; NINE MILE POINT NUCLEAR STATION, UNITS 1 AND 2; PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3; QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2; AND R. E. GINNA NUCLEAR POWER PLANT - PROPOSED ALTERNATIVE TO USE ENCODED PHASED ARRAY ULTRASONIC EXAMINATION TECHNIQUES (EPID L-2019-LLR-0011)

By letter dated February 15, 2019 (Agencywide Documents and Access Management System (ADAMS) Accession No. ML19049A001), as supplemented by letter dated June 25, 2019 (ADAMS Accession No. ML19176A343), Exelon Generation Company, LLC (the licensee) submitted a request in accordance with paragraph 50.55a(z)(1) of Title 10 of the *Code of Federal Regulations* (10 CFR) for a proposed alternative to the requirements of 10 CFR 50.55a and the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) at Braidwood Station, Units 1 and 2 (Braidwood); Byron Station, Unit Nos. 1 and 2 (Byron); Calvert Cliffs Nuclear Power Plant, Units 1 and 2 (Calvert Cliffs); Clinton Power Station, Unit No. 1 (Clinton); Dresden Nuclear Power Station, Units 2 and 3 (Dresden); James A. FitzPatrick Nuclear Power Plant (FitzPatrick); LaSalle County Station, Units 1 and 2 (LaSalle); Limerick Generating Station, Units 1 and 2 (Limerick); Nine Mile Point Nuclear Station, Units 1 and 2 (NMP); Peach Bottom Atomic Power Station, Units 2 and 3 (Peach Bottom); Quad Cities Nuclear Power Station, Units 1 and 2 (Quad Cities); and R. E. Ginna Nuclear Power Plant (Ginna). The proposed alternative would allow the licensee to use encoded phased array ultrasonic examination techniques in lieu of radiography for ferritic steel and austenitic stainless-steel piping welds for each of these facilities.

The application also requested to use the proposed alternative at Three Mile Island Nuclear Station (TMI), Unit 1. However, the licensee withdrew the request for TMI by letter dated June 17, 2019 (ADAMS Accession No. ML19169A031).

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that the licensee has adequately addressed the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC staff authorizes the use of the proposed alternative for the remainder of the current 10-year inservice inspection intervals at Braidwood Units 1 and 2, Byron Unit Nos. 1 and 2, Calvert Cliffs Units 1 and 2, Clinton, Dresden Units 2 and 3, FitzPatrick, LaSalle Units 1 and 2, Limerick

Units 1 and 2, NMP Units 1 and 2, Peach Bottom Units 2 and 3, Quad Cities Units 1 and 2, and Ginna, as specified in the licensee's June 25, 2019, letter. In addition, the NRC staff authorizes the use of the proposed alternative for the duration of the fourth 10-year inservice inspection interval at Clinton and the sixth 10-year inservice inspection interval at Ginna, as specified in the licensee's June 25, 2019, letter.

All other ASME Code requirements for which relief has not been specifically requested and approved remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Proposed Alternative and Basis for use:

The licensee proposed to use the encoded PAUT in lieu of the RT required by the ASME Code for volumetric examination of ferritic steel or austenitic stainless-steel piping welds during repair and replacement activities. Ultrasonic and radiographic testing are two volumetric examination techniques that are commonly used to inspect welds in nuclear power plants and in other industries. The two techniques use different physical mechanisms to detect and characterize discontinuities, which results in several key differences in sensitivity and discrimination capability between the two techniques.

The proposed alternative includes requirements for qualification of the encoded PAUT procedures, equipment, and personnel by the performance demonstration using representative piping conditions and flaws. The licensee stated that this approach will demonstrate the ability of the encoded PAUT to detect and accurately size flaws that are both acceptable and unacceptable to the defined acceptance standards. Section 5.1 of the application, as supplemented, provides a detailed description of the licensee's proposed alternative.

The licensee stated that the technical basis for the proposed alternative was developed from numerous codes and code cases, associated industry experience, research articles, and results of welds examinations by the ultrasonic and radiographic techniques. The licensee stated that encoded PAUT is equivalent or superior to RT for detecting and sizing critical (planar) flaws such as cracks and lack of fusion. Encoded PAUT provides sizing capabilities for both depth and length dimensions of the flaw; however, RT does not have the flaw depth sizing capabilities.

Conclusion:

As set forth above, the NRC staff determined that the licensee's proposed alternative to use encoded PAUT in lieu of RT provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC staff authorizes the use of the proposed alternative for the remainder of the current 10-year ISI intervals at Braidwood Units 1 and 2, Byron Unit Nos. 1 and 2, Calvert Cliffs Units 1 and 2, Clinton, Dresden Units 2 and 3, FitzPatrick, LaSalle Units 1 and 2, Limerick Units 1 and 2, NMP Units 1 and 2, Peach Bottom Units 2 and 3, Quad Cities Units 1 and 2, and Ginna, as specified in the licensee's June 25, 2019, letter. In addition, the NRC staff authorizes the use of the proposed alternative for the duration of the fourth 10-year ISI interval at Clinton and the sixth 10-year ISI interval at Ginna, as specified in the licensee's June 25, 2019, letter.

October 8, 2019 – Letter dated October 8, 2019 from Peter Bamford, Senior Project Manager Beyond-Design-Basis Management Brach Division of Licensing Projects Office of Nuclear Reactor Regulation to Bryan Hanson, Senior Vice President Exelon Generation Company, President and Chief Nuclear Officer Exelon Nuclear with subject of Peach Bottom Atom Power Station units 2 and 3 – correction regarding staff review of seismic probabilistic risk assessment associated with reevaluated seismic hazard implementation of the near-term task force recommendation 2.1:seismic (EPID no. L-2018-JLD-0010)

The purpose of this letter is to provide a correction regarding the staff's evaluation of the Peach Bottom Atomic Power Station, Units 2 and 3 (Peach Bottom), seismic probabilistic risk assessment (SPRA), which was submitted in response to Near~Term Task Force (NTIF) Recommendation 2.1 "Seismic." The correction does not change the U.S. Nuclear Regulatory Commission (NRC) staff's previous conclusion that no further response or regulatory action associated with NTIF Recommendation 2.1 "Seismic" is required for Peach Bottom.

By letter dated March 12, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12053A340), the NRC issued a request for information under Title 10 of the *Code of Federal Regulations* Section 50.54(f) (hereafter referred to as the 50.54(f) letter). The request was issued as part of implementing lessons learned from the accident at the Fukushima Dai-ichi nuclear power plant. Enclosure 1 to the 50.54(f) letter requested that licensees reevaluate seismic hazards at their sites using present-day methodologies and guidance. Enclosure 1, Item (8), of the 50.54(f) letter requested that certain licensees complete an SPRA to determine if plant enhancements are warranted due to the change in the reevaluated seismic hazard compared to the site's design-basis seismic hazard.

By letter dated August 28, 2018 (ADAMS Accession No. ML18240A065), Exelon Generation Company, LLC (Exelon, the licensee), provided its SPRA submittal in response to Enclosure 1, Item (8) of the 50.54(f) letter, for Peach Bottom. The NRC staff reviewed the SPRA submittal and provided its evaluation by letter dated June 10, 2019 (ADAMS Accession No. ML19053A469). This review used the guidance in NRC staff memorandum dated

August 29, 2017, titled, "Guidance for Determination of Appropriate Regulatory Action Based on Seismic Probabilistic Risk Assessment Submittals in Response to Near Term Task Force Recommendation 2.1: Seismic" (ADAMS Accession No. ML17146A200; hereafter referred to as the SPRA Screening Guidance) to develop a recommendation based on its review of the SPRAs submitted by licensees in response to the 50.54{f) letter.

During an internal self-assessment review, the staff recently uncovered an error in the spreadsheet used in the SPRA Screening Guidance to evaluate the Peach Bottom SPRA submittal. The correction of the error resulted in changes to certain numerical values that were documented in the staff's Peach Bottom SPRA evaluation. A description of the error and corrected values for the affected portions of the staff evaluation are provided in the enclosure to this letter. The staff has confirmed that the changes to the numerical values presented in the enclosure to this letter do not impact or change the NRC decision documented by the previously referenced staff evaluation dated June 10, 2019.

The NRC staff regrets any inconvenience this may have caused. If you have any questions, please contact me at (301) 415-2833, or via e-mail at Peter.Bamford@nrc.gov.

By letter dated August 28, 2018 (ADAMS Accession No. ML18240A065), Exelon Generation Company, LLG (Exelon, the licensee), provided its seismic probabilistic risk assessment (SPRA) submittal in response to Enclosure 1, Item (8) of the 50.54(f) letter [Title 10 of the *Code of Federal Regulations* Section 50.54(f), dated March 12, 2012] for Peach Bottom Atomic Power Station, Units 2 and 3 (Peach Bottom). The U.S. Nuclear Regulatory Commission (NRG) staff reviewed the SPRA submittal and provided its evaluation by letter dated June 10, 2019 (ADAMS Accession No. ML19053A469). This review used the guidance in NRG staff memorandum dated August 29, 2017, titled, "Guidance for Determination of Appropriate Regulatory Action Based on Seismic Probabilistic Risk Assessment Submittals in Response to Near Term Task Force Recommendation 2.1: Seismic" (ADAMS Accession No. ML17146A200; hereafter

referred to as SPRA Screening Guidance) to develop a recommendation based on its review of the SPRAs submitted by licensees in response to the 50.54(f) letter.

During an internal self-assessment review, the staff recently uncovered an error in the spreadsheet used to implement the SPRA Screening Guidance for evaluating the Peach Bottom SPRA submittal. The correction of the error resulted in changes to certain numerical values documented in the staff evaluation letter. The staff has confirmed that the changes to the numerical values do not impact or change the NRC decision documented by the previously referenced staff evaluation dated June 10, 2019.

Enclosure 2 to the staff evaluation letter dated June 12, 2019 (page 2 of Enclosure 2) contains a sentence that states the following:

The target RRWs [risk reduction worths] based on the mean and 95th percentile SCDF [seismic core damage frequency] and SLERF [seismic large early release frequency] were also calculated by the NRC staff and ranged between 1.63 and 1.96 for both units.

This sentence should have said (changes in **bold):**
The target RRWs based on the mean and 95th percentile SCDF and SLERF were also

calculated by the NRG staff and ranged between **1.04** and **1.60** for both units.

In addition, the correction of the spreadsheet error impacts certain values presented in Tables 1 and 2 of Enclosure 2 to the staff's evaluation letter dated June 1.0, 2019. The following corrected versions of the impacted portions of Tables 1 and 2 are provided. The numbers that have changed are shown in **bold.**

December 2, 2019 – Letter dated December 2, 2019 from James Danna, Chief Plant Licensing Branch 1 Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to Bryan hanson, Senior Vice President, Exelon Generation Company, President and Chief Nuclear Officer Exelon Generation Company with the subject of Peach Bottom Atomic Power Station Units 2 and 3 – Issuance of relief request 15R-10 Re: Examination of standby liquid control nozzle inside radius section in lier of specific asme code requirements (EPID L-2019-LLR-00760

By application dated August 21, 2019 (Agencywide Documents Access and Management System Accession No. ML19233A133), Exelon Generation Company, LLC (the licensee) submitted Relief Request I5R-10 to the U.S. Nuclear Regulatory Commission (NRC) for a proposed alternative to the requirements of the American Society of Mechanical Engineers Boiler & Pressure Vessel Code (ASME Code), Section XI, for the Peach Bottom Atomic Power Station (Peach Bottom), Units 2 and 3. The proposed alternative would allow the licensee to perform a visual inspection at operating pressure of the reactor pressure vessel head during Class 1 pressure boundary system leakage testing conducted at the end of each outage. The ASME Code requires 100 percent volumetric examination of the subject reactor pressure vessel nozzle inner radius sections. However, the nozzle is inaccessible for examination from inside the vessel due to the location of the nozzle in the reactor pressure vessel lower head area and due to the standby liquid control piping inside the vessel, which is fillet welded into the nozzle socket. These restrictions make the ASME Code-required examinations impractical to perform.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(g)(6)(i), the licensee requested relief and to use an alternative, for inservice inspection items on the basis that the ASME Code requirement is impractical.

The NRC staff has reviewed the subject request and finds that the proposed alternative provides reasonable assurance that the standby liquid control nozzle will maintain its structural integrity, leaktightness, and functionality during the service. Accordingly, the NRC staff concludes, as set forth in the enclosed safety evaluation, that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(g)(6)(i). Therefore, the NRC authorizes the use of this alternative at Peach Bottom, Units 2 and 3, for the fifth 10-year inservice inspection interval.

All other requirements of the ASME Code, Section XI, for which relief has not been specifically requested and authorized by the NRC staff remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

If you have any questions please contact the Peach Bottom Project Manager, Jennifer Tobin, at 301-415-2328 or Jennifer.Tobin@nrc.gov.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
REGARDING ALTERNATIVE REPAIR
FOR HIGH PRESSURE SERVICE WATER SYSTEM PIPING
EXELON GENERATION COMPANY, LLC

PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 DOCKET NOS. 50-277 AND 50-278

INTRODUCTION

By application dated August 21, 2019 (Agencywide Documents Access and Management System Accession No. ML19233A133), Exelon Generation Company, LLC (the licensee) submitted a relief request to the U.S. Nuclear Regulatory Commission (NRC or the Commission) for a proposed alternative to the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, for the Peach Bottom Atomic Power Station (Peach Bottom), Units 2 and 3. The proposed alternative would allow the licensee to perform a visual inspection, at operating pressure, of the reactor pressure vessel (RPV) head during Class 1 pressure boundary system leakage testing conducted at the end of each outage. The ASME Code requires 100 percent volumetric examination of the subject RPV nozzle inner radius sections. However, the nozzle is inaccessible for examination from inside the vessel due to the location of the nozzle in the RPV lower head area and due to the standby liquid control (SLC) piping inside the vessel, which is fillet welded into the nozzle socket. These restrictions make the ASME Code required examinations impractical to perform.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(g)(6)(i), the licensee requested to use the alternative on the basis that complying with the specified requirement would result in hardship or unusual difficulty, without a compensating increase in the level of quality and safety.

2.0 REGULATORY EVALUATION

The licensee’s request proposes an alternative to the requirements of the ASME Code, Section XI, Table IWB-2500-1, Examination Category B-D, Item No. B3.100. Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) must meet the requirements, except design and access provisions and preservice examination requirements, set forth in the ASME Code, Section XI, to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests

conducted during 120-month inspection intervals subsequent to the first inspection interval comply with the latest edition and addenda of the ASME Code, incorporated by reference in 10 CFR 50.55a(a), 12 months before the start of the 120-month inspection interval.

The regulations in 10 CFR 50.55a(g)(5)(iii) state that if a licensee determines that conformance with an ASME Code requirement is impractical for its facility, the licensee must notify the NRC and submit information in support of its determination. Determinations of impracticality must be based on the demonstrated limitations experienced when attempting to comply with the ASME Code requirements during the inservice inspection (ISI) interval for which the request is being submitted. Requests for relief must be submitted to the NRC no later than 12 months after the expiration of the 120-month inspection interval for which relief is sought.

The regulations in 10 CFR 50.55a(g)(6)(i) state that the NRC will evaluate determinations that ASME Code requirements are impractical. The NRC may grant such relief and may impose such alternative requirements as it determines are authorized by law, will not endanger life or property or the common defense and security, and are otherwise in the public interest, giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request the use of an alternative, and the NRC to authorize the proposed alternative for the fifth ISI interval, but not for the life of the plant.

1. 3.0 TECHNICAL EVALUATION
2. 3.1 Background

The relief request addresses the examination of the inside radius section of the SLC nozzle for the fifth ISI interval at Peach Bottom, Units 2 and 3. The examination category and item numbers are addressed in Table IWB-2500-1 of the ASME Code, 2001 Edition through
2003 Addenda.

3.2 ASME Code Requirements

Table IWB-2500-1, Examination Category B-D, Item No. B3.100, requires a volumetric examination to be performed on the inner radius section of all reactor vessel nozzles each inspection interval. Table IWB-2500-1, Examination Category B-D, Item No. B3.100, refers to the nozzle configurations shown in Figure No. IWB-2500-7.

3.3 Applicable ASME Code Edition and Addenda

For the fifth 10-year ISI interval at Peach Bottom, the Code of record for the inspection of ASME Code Class 1, 2, and 3 components is the ASME Code, Section XI, 2001 Edition through the 2003 Addenda.

3.4 Licensee's Proposed Alternative
In its August 21, 2019, submittal, the licensee stated, in part, the following:

The Standby Liquid Control (SLC) nozzle, as shown in Figure 1, is designed with an integral socket to which the boron injection piping is fillet welded. This design is different from the configurations shown in ASME, Section XI,Figure No. IWB-2500-7. The SLC nozzle is located in the bottom head of the vessel in an area that is inaccessible for ultrasonic examinations from the inside of the vessel. Therefore, ultrasonic examinations can only be performed from the outside diameter of the vessel. As shown in Figure 1, the ultrasonic scan would need to travel through the full thickness of the vessel into a complex cladding/socket configuration. These geometric and material reflectors inherent in the design prevent a meaningful examination from being performed on the inner radius of the SLC nozzle. In addition, the inner radius socket attaches to piping that injects boron at locations far removed from the nozzle. Therefore, the SLC nozzle inner radius is not subjected to turbulent mixing conditions that are a concern at other nozzles.

The licensee also stated that conformance with the ASME Code required examinations is impractical, as it would require extensive structural modifications to the component and surrounding structure, which would be cost prohibitive.

3.5 Proposed Alternative and Basis for Use

As an alternative examination, a system leakage test of the Class 1 pressure boundary is conducted at the end of each outage at operating pressure. The RPV bottom head penetrations, including the SLC penetration, are visually inspected during the leakage test with the acceptance criteria being zero leakage.

3.6 Duration of Proposed Alternative

The licensee requested relief for the fifth ISI interval for Peach Bottom, Units 2 and 3, which began on January 1, 2019, and is scheduled to conclude on December 31, 2028, and for the remainder of the plant life. The NRC staff finds that regulatory authority exists for the Commission to grant the relief requested by the licensee for the fifth ISI interval, but not for the life of the plant.

4.0 NRC STAFF EVALUATION

Pursuant to 10 CFR 50.55a(g)(5)(iii), the licensee submitted this request for relief from the examination requirements of the ASME Code, Section XI. The NRC staff's evaluation of the licensee's request for relief focused on (1) whether the ASME Code requirement is impractical, (2) whether the imposition of the ASME Code required inspections would result in a burden to the licensee, and (3) whether the licensee's examination coverage provides reasonable assurance of structural integrity and leaktightness of the subject welds.

The ASME Code requires 100 percent volumetric examination of the subject RPV nozzle inner radius sections. However, as shown in the drawing provided by the licensee in the application, the nozzle configuration and inside geometry prevent obtaining meaningful examination results from the outside of the RPV. The nozzle is inaccessible for examination from inside the vessel due to the location of the nozzle in the RPV lower head area and due to the SLC piping inside the vessel, which is fillet welded into the nozzle socket. These restrictions make the ASME Code-required examinations impractical to perform. To complete the examinations as required by the ASME Code, the licensee would have to redesign and modify the RPV and SLC piping. The NRC staff finds that imposition of the Code required examinations on the subject welds would result in a considerable and unnecessary burden on the licensee and is impractical.

The licensee is not able to obtain coverage of the 2-inch SLC nozzle inner radius section. In addition, because of the design of the nozzle, the SLC nozzle inner radius is not subjected to turbulent mixing conditions that are a concern at other nozzles. However, there are several other inner radius sections on similarly-sized nozzles in the RPV that are examined per ASME Code requirements. Therefore, any significant patterns of degradation should be detected by the other examinations in a timely manner. Therefore, the staff has determined that the licensee's corrective action, trending, and monitoring programs provide reasonable assurance that if any emerging aging degradation were to be detected in the SLC nozzle, the corrective actions would be expected to resolve the issue in a timely manner. The staff noted that previous operating experience to date in the SLC nozzle indicates that there is no active aging degradation mechanism, including intergranular stress corrosion cracking. During each outage, a system leakage test at operating pressure was conducted for the ASME Code Class 1 pressure boundary components and, to date, no leakage was detected in the SLC nozzle.

Based on the licensee's information provided and the staff evaluation as stated above, the staff determined that there is reasonable assurance that the SLC nozzle will maintain its structural integrity, leaktightness, and functionality during the service. The staff's evaluation was based on the following: (1) the SLC nozzle inner radius is not subjected to turbulent mixing conditions, which is validated by the operating experience (no cracking) to date in the SLC nozzle inner radius; (2) there is no active aging degradation mechanism, including intergranular stress corrosion cracking in the SLC nozzle; and (3) the licensee will perform system leakage test and associated VT-2 visual examination every refueling outage.

5.0 CONCLUSION

As set forth above, the NRC staff determines that the licensee has demonstrated that the proposed alternative provides reasonable assurance of structural integrity of the SLC nozzle. The NRC staff determines that granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law, will not endanger life or property or the common defense and security, and is otherwise in the public interest, giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility. Accordingly, the NRC staff concludes that the licensee has adequately addressed all the regulatory requirements set forth in 10 CFR 50.55a(g)(6)(i). Therefore, the NRC staff grants the use of this alternative for the fifth ISI interval for Peach Bottom.

All other requirements of the ASME Code, Section XI, for which relief has not been specifically requested and authorized by NRC staff remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

December 17, 2019 – Letter dated December 17, 2019 from Jennifer Tobin, Project Manager, Plant Licensing Branch 1, Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to Bryan Hanson, Senior Vice President Exelon Generation Company, President and Chief Nuclear Officer with a subject of Peach Bottom Atomic Power Station Units 2 and 3 – Issuance of Amendment numbers 329 and 332 regarding the adoption of TSTF-500, "DC electrical rewrite – update to TSTF-360)EPID L-2019-LLA-0118).

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued the enclosed Amendment Nos. 329 and 332 to Renewed Facility Operating License Nos. DPR-44 and DPR-56 for Peach Bottom Atomic Power Station (Peach Bottom), Units 2 and 3, respectively. These amendments consist of changes to the Technical Specifications (TSs) and Renewed Facility Operating Licenses in response to your application dated June 7, 2019, (Agencywide Documents Access and Management

System (ADAMS) Accession No. ML19158A312) as supplemented by letters dated August 29, 2019, and October 3, 2019 (ADAMS Accession Nos. ML19158A312, ML19241A465, and ML19276F281, respectively).

The amendments revise the requirements related to direct current (DC) electrical systems in TS 3.8.4, "DC Sources - Operating," to add a condition for the opposite unit's battery charger based on the NRG-approved Technical Specifications Task Force (TSTF) Traveler TSTF-500, Revision 2, "DC Electrical Rewrite - Update to TSTF-360." Specifically, the proposed condition allows a 72-hour completion time for an opposite unit battery charger that is required for certain plant configurations.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

EXELON GENERATION COMPANY, LLC
PSEG NUCLEAR, LLC
DOCKET NO. 50-277
PEACH BOTIOM ATOMIC POWER STATION, UNIT 2 AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 329 Renewed License No. DPR-44

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   1. The application for amendment by Exelon Generation Company, LLC (Exelon Generation Company), dated June 7, 2019, as supplemented by letters dated August 29, 2019, and October 3, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
   2. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
   3. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
   4. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
   5. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

   Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-44 is hereby amended to read as follows:

   (2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 329, are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

6. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

EXELON GENERATION COMPANY, LLC
PSEG NUCLEAR LLC
DOCKET NO. 50-278
PEACH BOTIOM ATOMIC POWER STATION, UNIT 3 AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 332 Renewed License No. DPR-56

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

1. The application for amendment by Exelon Generation Company, LLC (Exelon Generation Company), dated June 7, 2019, as supplemented by letters dated August 29, 2019, and October 3, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
2. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
3. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
4. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
5. The issuance of this amendment is in accordance with 1OCFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
6. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-56 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 332, are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

7. This license amendment is effective immediately as of its date of issuance and shall be implemented within 30 days of issuance.

CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

January 9, 2020 – Letter dated January 9, 2020 from Blake Purnell, Project Manager, Plant Licensing Branch III, Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to Bryan Hanson, Senior Vice President, Exelon Generation Company, President and Chief Nuclear Officer, Exelon Nuclear with the subject of: Braidwood Station, Units 1 and 2; Byron Station, Unit 1 and 2; Calvert Cliffs Nuclear Power Plant Units 1 and 2, Clinton Power Station unit 1, Dresden Nuclear Power Station Units 1, 2 and 3; James A Fitzpatrick Nuclear Power Plant; Lasalle County Station Units 1 and 2; Limerick Generating Station Units 1 and 2; Nine Mile Point Nuclear Station units 1 and 2; Peach Bottom Atomic Power Station Units 1 and 2; and R.E. Ginna Nuclear Power Plant – Review of Quality Assurance Program Changes (EPID L-2019-LLQ-0003)

By letter dated December 5, 2019 (Agencywide Documents Access and Management System Accession No. ML19339E544), Exelon Generation Company, LLC (Exelon) requested U.S. Nuclear Regulatory Commission (NRC) approval of changes to its Quality Assurance Topical Report (QATR) in accordance with paragraph 50.54(a)(4) of Title 10 of the *Code of Federal Regulations* (10 CFR). The proposed changes are applicable to the subject plants and their associated independent spent fuel storage installations.

Specifically, Exelon requested NRC approval to increase the internal audit interval for certain topics (listed on page 2 of Attachment 1 of the letter) from 24 months to 36 months. Exelon has determined that the changes in these audit intervals, and an associated deviation from NRC Regulatory Guide 1.189, "Fire Protection for Nuclear Power Plants," are a reduction in commitments requiring prior NRC approval to implement pursuant to 10 CFR 50.54(a). The NRC staff has determined that a review of these changes will take longer than 60 days. Therefore, Exelon shall refrain from implementing these changes until the staff's review is completed. The staff expects to complete this review by December 7, 2020, as requested in Exelon's letter.

In its letter, Exelon also identified additional changes to the QATR that it determined do not require prior NRC approval. The NRC staff is not reviewing these additional changes as part of this request. Exelon may implement these additional changes in accordance with 10 CFR 50.54(a).

February 24, 2020 – Letter dated February 24, 2020 from Mel Gray, Chief Engineering Branch 1 Division of Reactor Safety to Bryan C. Hanson Senior Vice President, Exelon Generation Company, LLC President and Chief Nuclear Officer, Exelon Nuclear Exelon Generation Company with a subject of PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 – REQUEST FOR INFORMATION TO SUPPORT TRIENNIAL BASELINE DESIGN BASES ASSURANCE INSPECTION (TEAM); INSPECTION REPORT 05000277/2020011 AND 05000278/2020011

The purpose of this letter is to notify you that the U.S. Nuclear Regulatory Commission (NRC) Region I staff will conduct a Design Bases Assurance Inspection (DBAI) at Peach Bottom Atomic Power Station, Units 2 and 3. Joe Schoppy, a Senior Reactor Inspector from the NRC's Region I Office, will lead the inspection team. The inspection will be conducted in accordance with Inspection Procedure 71111.21M, "Design Bases Assurance Inspection (Team)," dated December 8, 2016 (ADAMS Accession No. ML16340B000).

The inspection will evaluate the capability of risk-significant/low-margin components to function as designed to support proper system operation. The inspection will also include a review of selected modifications, operating experience, and as applicable, operator actions.

During an onsite conversation on February 18, with Dan Dullum, we confirmed arrangements for an information-gathering site visit and the two-week onsite inspection. The schedule is as follows:

- • Information-gathering visit: Week of April 20
- • Onsite weeks: Weeks of July 13, and July 27

The purpose of the information-gathering visit is to meet with members of your staff to identify risk-significant components, modifications, operator actions, and operating experience items. Information and documentation needed to support the inspection will also be identified.
Frank Arner, a Region I Senior Risk Analyst, will support Joe Schoppy during the information- gathering visit to review probabilistic risk assessment data and identify components to be examined during the inspection.

Experience with previous baseline design/modification inspections of similar depth and length has shown this type of inspection is resource intensive, both for the NRC inspectors and the licensee staff. In order to minimize the inspection impact on the site and to ensure a productive inspection for both parties, we have enclosed a request for information needed for the inspection.

t is important that all of these documents are up-to-date and complete in order to minimize the number of additional documents requested during the preparation and/or the onsite portions of the inspection. Insofar as possible, this information should be provided electronically to the lead inspector.

The information request has been divided into two groups:

- • The first group lists information necessary for our initial inspection scoping activities. This information should be provided to the lead inspector by April 20. By April 27, the lead inspector will communicate the initial selected set of components and modifications.
- • The second group of documents requested are needed to support our in-office preparation activities. This set of documents, specific to the selected components and modifications, should be provided to the lead inspector at the Region I Office no later than July 6. During the in-office preparation activities, the team may

identify additional information needed to support the inspection, and those items will be communicated directly to Dan Dullum.

If there are any questions about the inspection or the material requested in the enclosure, please contact the lead inspector at 610-337-5286 or via e-mail at jgs@nrc.gov.

This letter does not contain new or amended information collection requirements subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). Existing information collection requirements were approved by the Office of Management and Budget, Control Number 3150- 0011. The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid Office of Management and Budget Control Number.

This letter and its enclosure will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with Title 10 of the *Code of Federal Regulations*, Part 2.390, “Public Inspections, Exemptions, Requests for Withholding.”

DOCUMENT REQUEST FOR DESIGN BASES ASSURANCE INSPECTION

Inspection Report: **Onsite Inspection Dates**: **Inspection Procedure**:

Lead Inspector:

05000277/2020011 and 05000278/2020011
July 13, through July 17; and July 27, through July 31

Inspection Procedure 71111.21M, Design Bases Assurance Inspection (Team)

Joe Schoppy, Senior Reactor Inspector Phone: 610-337-5286
Email: jgs@nrc.gov

I. Information Requested for Selection of Components and Modifications

The following information is requested by April 20 to facilitate inspection preparation. Feel free to contact the lead inspector as soon as possible if you have any questions regarding this information request. Please provide the information electronically in “pdf” files, Excel, or other searchable formats, preferably on some portable electronic media (e.g., CD-ROM, DVD). The files should contain descriptive names and be indexed and hyperlinked to facilitate ease of use. Information in “lists” should contain enough information to be easily understood by someone who has knowledge of light water reactor technology.

1. The site probabilistic risk analysis (PRA) “System Notebook” and latest PRA Summary Document.
2. Risk ranking of top 250 basic events sorted by Risk Achievement Worth (>/= 1.3). Include values for Risk Reduction Worth, Birnbaum Importance, and Fussell-Vesely (as applicable). Please provide in an excel spreadsheet or other sortable format and include an understandable definition of the coded basic events.
3. Risk-ranking of top 100 components from site specific PRA sorted by Large Early Release Frequency.
4. If you have an External Events PRA Model, provide the information requested in Item 2 for external events. Provide narrative description of each coded event, including flood zonedescription.
5. List of time-critical and/or risk significant operator actions.
6. List of emergency and abnormal operating procedures.
7. If available, any pre-existing evaluation or list of components and associated calculations with low design margins (e.g., pumps closest to the design limit for flow or pressure, diesel generator close to design required output, heat exchangers close to rated design heat removal).

DOCUMENT REQUEST FOR DESIGN BASES ASSURANCE INSPECTION

8. If applicable, copy of any self-assessments and/or Quality Assurance assessments of low margin structures, systems and components (SSCs) completed since February 10, 2017.
9. List of available design margins in both the open and closed direction for valves in the motor-operated valve and air-operated valve programs (related to GL 96-05, looking for resultant output – matrix of risk vs margin for MOVs and AOVs, as applicable).
10. The age and capacity of the safety-related DC batteries.
11. The In-Service Testing (IST) Program Basis document identifying the in-scope valves and pumps, and the associated IST Program requirements for each component (e.g., IST valve table identifying category, active/passivefunction).
12. Access to IST trend data for the following pumps for both units: RHR pumps, HPCI pumps, RCIC pumps, SLC pumps, HPSW pumps, and ESW pumps. [Note: needed for each discrete component (e.g. for each RHR pump)]
13. Listing of MR (a)(1) systems, date entered into (a)(1) status, and brief description of why in (a)(1) status.
14. List of Maintenance Rule Functional Failure (MRFF) evaluations completed since February 10, 2017 (include those determined not to be a MRFF).
15. A copy of the most recent System Health and/or trending reports for the following systems (as applicable): Safety Related (SR) 4KV, SR 480 Vac, HPCI, RCIC, RHR, HPSW, ESW, SR 125 Vdc, SR 250 Vdc, SLC, and EDGs.
16. A copy of the most recent Program Health and/or trending reports for the following programs, as applicable: GL 89-10 (MOVs), GL 89-13, IST, AOVs, breakers, relays.
17. List of open operability evaluations.
18. List of current “operator work arounds/burdens.”
19. List of “permanent plant modifications” to SSCs that are field work complete since February 10, 2017. For the purpose of this inspection, permanent plant modifications include permanent: plant changes, design changes, set point

changes, equivalency evaluations, suitability analyses, and commercial grade dedications. The list should contain the number of each document, title (sufficient to understand the purpose of the modification), revision/date, and the affected system.

20. List of calculation changes (including new calculations) that have been issued for use since February 10, 2017.
21. Corrective Action Program procedure.

22. Procedures addressing the following: modifications, design changes, set point changes, equivalency evaluations or suitability analyses, commercial grade dedications, and post-modification testing.
23. List of corrective action documents (open and closed) since February 10, 2017, that address permanent plant modifications issues, concerns, or processes.
24. Any internal/external self-assessments and associated corrective action documents generated in preparation for this inspection.
25. Updated Final Safety Analysis Report, Technical Specifications, Technical Specifications Bases, and Technical Requirements Manual.
26. Electrical simple one-line drawings for 4KV, 480V, 500KV, 230KV & 13KV (page size of 11 X 17 preferred).
27. Copy of Exelon's internal response to the following NRC Information Notices: 2017-03, 2017-05 (and 2017-05 Rev. 1), 2018-07, and 2019-02.
28. A list of NRC Part 21 Reports, determined to be applicable to Peach Bottom, since February 10, 2017.
29. An electronic copy of the following Design Basis Documents (DBDs) (if applicable & available): SR 4KV, SR 480 Vac, HPCI, RCIC, RHR, HPSW, ESW, SR 125 Vdc, SR 250 Vdc, SLC, and EDGs.

II. Information Requested to Be Available by July 6

This information should be separated for each selected component and modification, especially if provided electronically (e.g., a folder for each component and modification named after the component or modification that includes the information requested below). Items 1 through 11 are associated with the selected components and Item 12 is for the selected modifications.

1. List of corrective action documents associated with each selected component since February 10, 2017.
2. Maintenance history (e.g., corrective, preventive, and elective) associated with each selected component for the last five years. Identify frequency of preventive maintenance activities.
3. Aging Management Program documents and/or License Renewal committed inspection results applicable to each selected component.
4. List of calculations associated with each selected component, excluding data files. Pipe stress calculations are excluded from this request.
5. System Health Report (last completed) and Design Basis Document associated with each selected component, as applicable.

6. Access to or copy of vendor manual(s) for each selected component.
7. List of open temporary modifications associated with each selected component, if applicable.
8. Trend data/graphs on the selected components' performance since February 10, 2017 (e.g., pump performance including IST, other vibration monitoring, oil sample results).
9. List of normal operating and alarm response procedures associated with each selected component.
10. Last completed tests and surveillances for each selected component performed since February 10, 2017. For those tests and surveillances performed at a periodicity of greater than three years, provide the latest test performed.
11. Schedule of surveillance testing of selected components that occur during the onsite inspection weeks.
12. For each selected modification, copies of associated documents such as modification package, engineering changes, 50.59 screening or evaluation, relevant calculations, post-modification test packages, associated corrective action documents, design drawings, and new/revised preventive maintenance requirements.

February 25, 2020 – Email dated February 25, 2020 from Blake Purnell to David Neff (Exelon Nuclear) and cc to Nancy Salgado, Shannon Rafferty-Czincila (GenCo-Nuc) with subject line of Exelon Generation Company, LLC - Fleet Alternative Request to Use ASME Code Case OMN-26 (EPID L-2020-LLR-0012)

By application dated January 30, 2020 (ADAMS Accession No. ML20034C819), Exelon Generation Company, LLC (the licensee) submitted a request in accordance with Paragraph 50.55a(z)(1) of Title 10 of the Code of Federal Regulations (10 CFR) for a proposed alternative to the requirements of 10 CFR 50.55a and the American Society of Mechanical Engineers (ASME) Code at Braidwood Station, Units 1 and 2; Calvert Cliffs Nuclear Power Plant, Units 1 and 2; Clinton Power Station, Unit No. 1; Limerick Generating Station, Units 1 and 2; Nine Mile Point Nuclear Station, Units 1 and 2; Peach Bottom Atomic Power Station, Units 2 and 3; and R. E. Ginna Nuclear Power Plant. The proposed alternative would allow the licensee to use ASME Code Case OMN-26, "Alternate Risk-Informed and Margin Based Rules for Inservice Testing of Motor Operated Valves," at these facilities.

The purpose of this email is to provide the results of the U.S. Nuclear Regulatory Commission (NRC) staff's acceptance review of this proposed alternative. The acceptance review was performed to determine if there is sufficient technical information in scope and depth to allow the NRC staff to complete its detailed technical review. The acceptance review is also intended to identify whether the application has any readily apparent information insufficiencies in its characterization of the regulatory requirements or the licensing basis of the plant.

The NRC staff has reviewed your application and concluded that it provides technical information in sufficient detail to enable the staff to complete its detailed technical review and make an independent assessment regarding the acceptability of the proposed alternative in terms of regulatory requirements and the protection of public health and safety and the environment. Given the lesser scope and depth of the acceptance review, as compared to the detailed technical review, there may be instances in which issues that impact the staff's ability to complete the detailed technical review are identified despite completion of an adequate acceptance review. You will be advised of any further information needed to support the staff's detailed technical review by separate correspondence.

Based on the information provided in your submittal, the NRC staff estimates that review of this request will take approximately 325 hours to complete. The staff expects to complete its review by February 28, 2021. These estimates are based on the staff's initial review of the application and they could change due to several factors, including requests for additional information. If there are emergent complexities or challenges in our review that would cause changes to the initial forecasted completion date or significant changes to the forecasted hours, I will inform you of the reason for the change and provide the new estimates.

February 27, 2020 – email dated February 27, 2020 from Blake Purnell to David Neff (GenCo-Nuc) and cc to Rafferty-Czincila, Shannon B:(GenCo-Nuc); Salgado, Nancy with a subject of Exelon Generation Company, LLC - Acceptance of Proposed Alternative to Extend Safety Relief Valve Test Frequency (EPIDs L-2020-LLR-0014 through L- 2020-LLR-0018)

By application dated February 4, 2020 (ADAMS Accession No. ML20036D962), Exelon Generation Company, LLC (the licensee) submitted a request in accordance with Paragraph 50.55a(z)(1) of Title 10 of the Code of Federal Regulations (10 CFR) for a proposed alternative to the requirements of 10 CFR 50.55a and the American Society of Mechanical Engineers (ASME) Code at Clinton Power Station, Unit No. 1; Dresden Nuclear Power Station, Units 2 and 3; Nine Mile Point Nuclear Station, Unit 2; Peach Bottom Atomic Power Station, Units 2 and 3; and Quad Cities Nuclear Power Station, Units 1 and 2. The proposed alternative would allow the licensee to extend the safety relief valve test frequency at these facilities.

The purpose of this email is to provide the results of the U.S. Nuclear Regulatory Commission (NRC) staff's acceptance review of this proposed alternative. The acceptance review was performed to determine if there is sufficient technical information in scope and depth to allow the NRC staff to complete its detailed technical review. The acceptance review is also intended to identify whether the application has any readily apparent information insufficiencies in its characterization of the regulatory requirements or the licensing basis of the plant.

The NRC staff has reviewed your application and concluded that it provides technical information in sufficient detail to enable the staff to complete its detailed technical review and make an independent assessment regarding the acceptability of the proposed alternative in terms of regulatory requirements and the protection of public health and safety and the environment. Given the lesser scope and depth of the acceptance review, as compared to the detailed technical review, there may be instances in which issues that impact the staff's ability to complete the detailed technical review are identified despite completion of an adequate acceptance review. You will be advised of any further information needed to support the staff's detailed technical review by separate correspondence.

Based on the information provided in your submittal, the NRC staff estimates that review of this request will take approximately 120 hours per site to complete. The staff expects to complete its review by February 28, 2021. These estimates are based on the staff's initial review of the application and they could change due to several factors, including requests for additional information. If there are emergent complexities or challenges in our review that would cause changes to the initial forecasted completion date or significant changes to the forecasted hours, I will inform you of the reason for the change and provide the new estimates.

March 3, 2020 – Letter from Daniel S. Collins, Director Division of Reactor Projects to Bryan C. Hanson, Exelon Generation Company, LLC with the subject line of ANNUAL ASSESSMENT LETTER FOR PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 (REPORT 05000277/2019006 AND 05000278/2019006)

The NRC has completed its end-of-cycle performance assessment of Peach Bottom Atomic Power Station, Units 2 and 3, reviewing performance indicators (PIs), inspection results, and enforcement actions from January 1, 2019 through December 31, 2019. This letter informs you of the NRC's assessment of your facility during this period and its plans for future inspections at your facility. The NRC concluded that overall performance at your facility preserved public health and safety.

The NRC determined the performance at Peach Bottom Atomic Power Station, Units 2 and 3 during the most recent quarter was within the Licensee Response Column (Column 1) of the NRC's Reactor Oversight Process (ROP) Action Matrix in Inspection Manual Chapter 0305, "Operating Reactor Assessment Program," because all inspection findings had very low safety significance (i.e., Green), and all PIs were within the expected range (i.e., Green). Therefore, the NRC plans to conduct ROP baseline inspections at Peach Bottom Atomic Power Station, Units 2 and 3. Note that Peach Bottom Atomic Power Station, Units 2 and 3, returned to the Licensee Response Column on June 27, 2019 following a supplemental inspection documented in NRC Inspection Report 05000277/2019040 and 05000278/2019040 (ML19178A008).

The enclosed inspection plan lists the inspections scheduled through December 31, 2021. The NRC provides the inspection plan to allow for the resolution of any scheduling conflicts and personnel availability issues. Routine inspections performed by resident inspectors are not included in the inspection plan. You should be aware that the agency is pursuing potential changes to the ROP, including changes to engineering inspections (SECY-18-0113, "Recommendations for Modifying the Reactor Oversight Process Engineering Inspections") and other changes to the baseline inspection program described in SECY-19-0067, "Recommendations for Enhancing the Reactor Oversight Process." Should these changes to the ROP be implemented, the engineering and other region-based inspections are subject to change in scope, as well as schedule. Furthermore, all the inspections listed during the last

twelve months of the inspection plan are tentative and may be revised. The NRC will contact you as soon as possible to discuss changes to the inspection plan should circumstances warrant any changes.

In addition to baseline inspections, the NRC will also conduct Inspection Procedure (IP) 81311, "Physical Security Requirements for Independent Spent Fuel Storage

Installations," in April 2020; IP 60854.1, "Preoperational Testing of Independent Spent Fuel Storage Facility Installation at Operating Plants," in March through May 2020; IP 60855.1, "Operation of an ISFSI at Operating Plant," in June 2020; and Temporary Instruction (TI) 2515/193, "Hardened Containment Vent," in May 2020. Lastly, during this period the NRC will perform inspections per TI 2515/194, "Inspection of the Licensee's Implementation of Industry Initiative Associated with the Open Phase Condition Design Vulnerability in Electrical Power Systems (NRC Bulletin 2012-01)."

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with Title 10 of the *Code of Federal Regulations* 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Please contact Jonathan Greives at (610) 337-5337 with any questions you have regarding this letter.

IP 22 Inspection Activity Plan Report

Unit	Start	End	Activity	CAC	Title	Staff Count
Access Authorization and Fitness for Duty-PB 3						
2, 3 02/03/2020 02/06/2020 IP 71151 001338 Performance Indicator Verification						
ISFSI Dry Run & Operational Inspection - PB 2						
2, 3 03/16/2020 03/20/2020 IP 60854.1 000589 Preoperational Testing of Independent Spent Fuel Storage Facility Installation at Operating Plants						
2, 3 03/30/2020 04/03/2020 IP 60854.1 000589 Preoperational Testing of Independent Spent Fuel Storage Facility Installation at Operating Plants						
2, 3 04/20/2020 04/24/2020 IP 60854.1 000589 Preoperational Testing of Independent Spent Fuel Storage Facility Installation at Operating Plants						
2, 3 05/11/2020 05/15/2020 IP 60854.1 000589 Preoperational Testing of Independent Spent Fuel Storage Facility Installation at Operating Plants						
2, 3 06/01/2020 06/05/2020 IP 60855.1 000590 Operation of an Independent Spent Fuel Storage Installation at Operating Plants						
TI-194 Open Phase Condition Inspection-PB 2						
2, 3 04/13/2020 04/17/2020 TI 2515/194 000512 Inspection of the Licensee's Implementation of Industry Initiative Associated With the Open Phase Condition Design Vulnerabilities In Electric Power Systems (NRC Bulletin 2012-01)						
FY20 PEACH BOTTOM Initial Exam 3						
2, 3 04/19/2020 04/25/2020 OV 000956 VALIDATION OF INITIAL LICENSE EXAMINATION (OV)						
2, 3 05/24/2020 06/06/2020 EXAD 000500 LICENSE EXAM ADMINISTRATION (EXAD)						
EP EXERCISE INSPECTION - PEACH BOTTOM 5						
2, 3 04/20/2020 04/24/2020 IP 71114.01 000716 Exercise Evaluation						
2, 3 04/20/2020 04/24/2020 IP 71114.04 000719 Emergency Action Level and Emergency Plan Changes						
2, 3 04/20/2020 04/24/2020 IP 71151 001397 Performance Indicator Verification						
ISFSI- Security Inspection-PB 2						
2, 3 04/20/2020 04/24/2020 IP 71130.09 001656 Security Plan Changes						

2, 3 04/20/2020 04/24/2020 IP 81311 000831 Physical Security Requirements for Independent Spent Fuel Storage Installations						
Resp Protection & Dose Assessment (71124.03/71124.04) 1						
2, 3 04/27/2020 05/01/2020 IP 71124.03 000727 In-Plant Airborne Radioactivity Control and Mitigation						
2, 3 04/27/2020 05/01/2020 IP 71124.04 000728 Occupational Dose Assessment						

This report does not include INPO and OUTAGE activities.
This report shows only on-site and announced inspection procedures.

3/2/2020 9:48:08 AM

Enclosure

Peach Bottom

01/01/2020 - 12/31/2021

IP 22 Inspection Activity Plan Report

Unit	Start	End	Activity	CAC	Title	Staff Count
TI-193 Hardened Containment Vent 2						
2, 3 05/04/2020 05/08/2020 TI 2515/193 000511 Inspection of the Implementation of EA-13-109: Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation under Severe Accident Conditions						
Rad Monitoring Instrumentation (71124.05) 1						
2, 3 06/08/2020 06/12/2020 IP 71124.05 000729 Radiation Monitoring Instrumentation						
Design Basis Assurance Inspection - Teams - Peach Bottom Units 2 and 3 6						
2, 3 07/13/2020 07/19/2020 IP 71111.21M 000713 Design Bases Assurance Inspection (Teams)						
2, 3 07/27/2020 08/02/2020 IP 71111.21M 000713 Design Bases Assurance Inspection (Teams)						
FORCE-ON-FORCE PLANNING AND EXERCISE WEEKS - PB 6						
2, 3 07/27/2020 07/31/2020 IP 71130.03 000735 Contingency Response - Force-On-Force Testing						
2, 3 08/17/2020 08/21/2020 IP 71130.03 000735 Contingency Response - Force-On-Force Testing						
REMP (71124.07) 1						
2, 3 09/21/2020 09/25/2020 IP 71124.07 000731 Radiological Environmental Monitoring Program						
ISI - UNIT 2 1						
2 10/26/2020 10/30/2020 IP 71111.08G 000701 Inservice Inspection Activities (BWR)						
Rad Hazards (71124.01) & PIs (71151) 1						
2, 3 10/26/2020 10/30/2020 IP 71124.01 000725 Radiological Hazard Assessment and Exposure Controls						
2, 3 10/26/2020 10/30/2020 IP 71151 000746 Performance Indicator Verification						
EP Program Inspection - Peach Bottom 1						
2, 3 02/08/2021 02/12/2021 IP 71114.02 000717 Alert and Notification System Testing						
2, 3 02/08/2021 02/12/2021 IP 71114.03 000718 Emergency Response Organization Staffing and						

Augmentation System
2, 3 02/08/2021 02/12/2021 IP 71114.04 000719 Emergency Action Level and Emergency Plan Changes
2, 3 02/08/2021 02/12/2021 IP 71114.05 000720 Maintenance of Emergency Preparedness
2, 3 02/08/2021 02/12/2021 IP 71151 001397 Performance Indicator Verification
Access Control, Equipment Testing and Maintenance, Training, SPR 3
2, 3 04/05/2021 04/09/2021 IP 71130.09 001656 Security Plan Changes
2, 3 04/05/2021 04/09/2021 IP 71151 001338 Performance Indicator Verification

This report does not include INPO and OUTAGE activities.
This report shows only on-site and announced inspection procedures.

Peach Bottom

01/01/2020 - 12/31/2021

IP 22 Inspection Activity Plan Report

Unit	Start	End	Activity	CAC	Title	Staff Count
FY2021 Peach Bottom Initial Examination 5						
2, 3 05/02/2021 05/08/2021 OV 000956 VALIDATION OF INITIAL LICENSE EXAMINATION (OV)						
2, 3 05/30/2021 06/05/2021 EXAD 000500 LICENSE EXAM ADMINISTRATION (EXAD)						
INSERVICE INSPECTION - UNIT 3 1						
3 10/24/2021 10/30/2021 IP 71111.08G 000701 Inservice Inspection Activities (BWR)						
Rad Hazards (71124.01) and PIs (71151) 1						
2, 3 10/25/2021 10/29/2021 IP 71124.01 000725 Radiological Hazard Assessment and Exposure Controls						
2, 3 10/25/2021 10/29/2021 IP 71151 000746 Performance Indicator Verification						

This report does not include INPO and OUTAGE activities.
This report shows only on-site and announced inspection procedures.

May 6, 2020 – Letter dated May 6, 2020 from Jonathan E. Greives, Chief Reactor Projects Branch 4 Division of Reactor Projects to Bryan C. Hanson, Senior Vice President Exelon Generation Company, LLC with subject of PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 – INTEGRATED INSPECTION REPORT 05000277/2020001 AND 05000278/2020001

On March 31, 2020, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Peach Bottom Atomic Power Station, Units 2 and 3. On April 10, 2020, the NRC inspectors discussed the results of this inspection with Mr. Matthew Herr, Site Vice President, and other members of your staff. The results of this inspection are documented in the enclosed report.

No findings or violations of more than minor significance were identified during this inspection.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with Title 10 of the *Code of Federal Regulations* 2.390, “Public Inspections, Exemptions, Requests for Withholding.”

U.S. NUCLEAR REGULATORY COMMISSION Inspection Report

Docket Numbers: 05000277 and 05000278
License Numbers: DPR-44 and DPR-56
Report Numbers: 05000277/2020001 and 05000278/2020001
Enterprise Identifier: I-2020-001-0057
Licensee: Exelon Generation Company, LLC
Facility: Peach Bottom Atomic Power Station, Units 2 and 3 Delta, PA 17314
Inspection Dates: January 1, 2020 to March 31, 2020
Inspectors: J. Heinly, Senior Resident Inspector P. Boguszewski, Resident Inspector J. Schoppy, Senior Reactor Inspector
Approved by: Jonathan E. Greives, Chief Reactor Projects Branch 4 Division of Reactor Projects

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee’s performance by conducting an integrated inspection at Peach Bottom Atomic Power Station, Units 2 and 3, in accordance with the Reactor Oversight Process. The Reactor Oversight Process is the NRC’s program for overseeing the safe operation of commercial nuclear power reactors. Refer to https://www.nrc.gov/reactors/operating/oversight.html for more information.

List of Findings and Violations

No findings or violations of more than minor significance were identified.

None.

Additional Tracking Items

PLANT STATUS

Unit 2 began the inspection period at rated thermal power. On March 26, 2020, operators reduced power to 55 percent for a routine maintenance load drop. The unit was returned to rated thermal power on March 27, 2020 and remained there for the duration of the inspection period.

Unit 3 began the inspection period at rated thermal power. On March 6, 2020, operators reduced power to 73 percent for a routine maintenance load drop. The unit was returned to rated thermal power on March 9, 2020 and remained there for the duration of the inspection period.

INSPECTION SCOPES

Inspections were conducted using the appropriate portions of the inspection procedures (IPs) in effect at the beginning of the inspection unless otherwise noted. Currently approved IPs with their attached revision histories are located on the public website at http://www.nrc.gov/readingrm/doc-collections/insp-manual/inspection-procedure/index.html. Samples were declared complete when the IP requirements most appropriate to the inspection activity were met consistent with Inspection Manual Chapter (IMC) 2515, "Light-Water Reactor Inspection Program - Operations Phase." From January 1 – March 19, 2020, the inspectors performed plant status activities described in IMC 2515, Appendix D, "Plant Status," and conducted routine reviews using IP 71152, "Problem Identification and Resolution." The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel to assess licensee performance and compliance with Commission rules and regulations, license conditions, site procedures, and standards.

Starting on March 20, 2020, in response to the National Emergency declared by the President of the United States on the public health risks of the coronavirus (COVID-19), resident inspectors were directed to begin telework and to remotely access licensee information using available technology. During this time the resident inspectors performed periodic site visits each week and during that time conducted plant status activities as described in IMC 2515, Appendix D; and observed risk significant activities when warranted. In addition, resident and regional baseline inspections were evaluated to determine if all or portion of the objectives and requirements stated in the IP could be performed remotely. If the inspections could be performed remotely, they were conducted per the applicable IP. In the cases where it was determined the objectives and requirements could not be performed remotely, management elected to postpone and reschedule the inspection to a later date.

REACTOR SAFETY

71111.01 - Adverse Weather Protection
Seasonal Extreme Weather Sample (IP Section 03.01) (1 Sample)

(1) The inspectors evaluated winter readiness for seasonal extreme weather conditions prior to the onset of seasonal cold temperatures, potential deepening extreme drought, and seasonal heavy rains on January 22 and 23, 2020

71111.04 - Equipment Alignment

Partial Walkdown Sample (IP Section 03.01) (5 Samples)

The inspectors evaluated system configurations during partial walkdowns of the following systems/trains:

1. (1) Unit 3 reactor core isolation cooling (RCIC) during Unit 3 high-pressure coolant injection (HPCI) system outage window (SOW) on January 21, 2020
2. (2) Unit 2 standby liquid control system during Unit 2 and Unit 3 E-1 emergency diesel generator (EDG) SOW on February 4, 2020
3. (3) Unit 3 'B' core spray (CS) on February 10, 2020
4. (4) Unit 2 and Unit 3 E-3 EDG while Unit 2 and Unit 3 E-1 EDG was inoperable on

   February 27, 2020

5. (5) Unit 3 HPCI during Unit 3 RCIC SOW on March 10, 2020

71111.05 - Fire Protection
Fire Area Walkdown and Inspection Sample (IP Section 03.01) (5 Samples)

The inspectors evaluated the implementation of the fire protection program by conducting a walkdown and performing a review to verify program compliance, equipment functionality, material condition, and operational readiness of the following fire areas:

1. (1) Inner screen area, PF-145, on January 9, 2020
2. (2) Cable spreading room, PF-78H, on January 15, 2020
3. (3) Unit 3 'A' CS instrument room, PF-13G, on January 23, 2020
4. (4) Unit 3 'B' CS instrument room, PF-13F, on January 23, 2020
5. (5) Switchgear rooms during E-1 SOW, PF 117, on January 31, 2020

71111.06 - Flood Protection Measures
Inspection Activities - Internal Flooding (IP Section 03.01) (1 Sample)

The inspectors evaluated internal flooding mitigation protections in the:

(1) Unit 2 'B' and 'D' CS rooms 71111.07A - Heat Sink Performance
Annual Review (IP Section 03.01) (1 Sample)

The inspectors evaluated readiness and performance of:
(1) Unit 2 'D' residual heat removal (RHR) heat exchanger on January 14, 2020

71111.11Q - Licensed Operator Requalification Program and Licensed Operator Performance

Licensed Operator Performance in the Actual Plant/Main Control Room (IP Section 03.01) (1 Sample)

(1) The inspectors observed and evaluated licensed operator performance in the main control room on February 28, 2020

Licensed Operator Requalification Training/Examinations (IP Section 03.02) (1 Sample)

(1) The inspectors observed and evaluated requalification training that involved multiple equipment failures that resulted in a reactor SCRAM and a steam leak on January 27, 2020

71111.12 - Maintenance Effectiveness
Maintenance Effectiveness (IP Section 03.01) (1 Sample)

The inspectors evaluated the effectiveness of maintenance to ensure the following structures, systems, and components (SSCs) remain capable of performing their intended function:

(1) Unit 2 RHR during the week of March 22, 2020
71111.13 - Maintenance Risk Assessments and Emergent Work Control Risk Assessment and Management Sample (IP Section 03.01) (6 Samples)

The inspectors evaluated the accuracy and completeness of risk assessments for the following planned and emergent work activities to ensure configuration changes and appropriate work controls were addressed:

1. (1) Unit 2 startup electrical feed out of service on January 21, 2020
2. (2) Unit 2 and Unit 3 E-1 SOW during the week of January 27, 2020
3. (3) Unit 3 'B' CS with Unit 3 'A' CS SOW on February 11, 2020
4. (4) Unit 3 HPCI on February 26, 2020
5. (5) Unit 3 RCIC SOW on March 11, 2020
6. (6) Unit 2 HPCI SOW on March 24, 2020

71111.15 - Operability Determinations and Functionality Assessments
Operability Determination or Functionality Assessment (IP Section 03.01) (6 Samples)

The inspectors evaluated the licensee's justifications and actions associated with the following operability determinations and functionality assessments:

1. (1) Unit 2 'D' RHR minimum flow slow stroke on January 3, 2020
2. (2) Unit 2 'D' RHR heat exchanger degraded thermal performance on January 14, 2020
3. (3) Unit 3 HPCI overspeed trip nonfunctional on January 15, 2020
4. (4) Unit 2 and 3 EDG, E-2, and E-4 turbo inlet check valve on January 29, 2020

(5) Unit 2 and Unit 3 E-1 coolant system voiding on February 29, 2020 (6) Bussman KLM-3 fuse defects on March 10, 2020

71111.19 - Post-Maintenance Testing
Post-Maintenance Test Sample (IP Section 03.01) (6 Samples)

The inspectors evaluated the following post-maintenance test activities to verify system operability and functionality:

1. (1) Unit 3 HPCI pressure switch replacement on January 23, 2020
2. (2) Unit 2 and Unit 3 E-1 EDG following SOW on February 13, 2020

3. (3) Unit 3 'A' CS after SOW on February 13, 2020
4. (4) Unit 2 and Unit 3 E-1 EDG following coolant vent replacement on March 5, 2020
5. (5) Unit 3 RCIC after SOW on March 9, 2020
6. (6) Unit 2 HPCI after SOW on March 24, 2020

71111.22 - Surveillance Testing
The inspectors evaluated the following surveillance tests: Surveillance Tests (other) (IP Section 03.01) (4 Samples)

1. (1) Unit 2 and Unit 3 Standby Gas Treatment System 'B' Filter Functional test on January 7, 2020
2. (2) Unit 2 'D' battery charger on February 17 and 18, 2020
3. (3) Unit 2 RCIC Logic System Functional testing on March 3, 2020
4. (4) Unit 3 RCIC on March 12, 2020

Inservice Testing (IP Section 03.01) (1 Sample)
(1) Unit 2 RCIC Pump, Valve and Flow test (IST) on February 28, 2020

OTHER ACTIVITIES – BASELINE

71151 - Performance Indicator Verification
The inspectors verified licensee performance indicators submittals listed below:
IE01: Unplanned Scrams per 7000 Critical Hours Sample (IP Section 02.01) (2 Samples)

1. (1) Unit 2 unplanned scrams from January 2019 to December 2019
2. (2) Unit 3 unplanned scrams from January 2019 to December 2019

IE03: Unplanned Power Changes per 7000 Critical Hours Sample (IP Section 02.02) (2 Samples)

1. (1) Unit 2 unplanned power changes from January 2019 to December 2019
2. (2) Unit 3 unplanned power changes from January 2019 to December 2019

IE04: Unplanned Scrams with Complications (USwC) Sample (IP Section 02.03) (2 Samples)

1. (1) Unit 2 unplanned scrams with complications from January 2019 to December 2019
2. (2) Unit 3 unplanned scrams with complications from January 2019 to December 2019

71152 - Problem Identification and Resolution
Annual Follow-up of Selected Issues (IP Section 02.03) (1 Sample)

The inspectors reviewed the licensee’s implementation of its corrective action program (CAP) related to the following issues:

(1) Unit 3 Control Rod Drive Pressure Low After 220-08 Line Trip

INSPECTION RESULTS

Observation: 71152 - Problem Identification and Resolution (Annual Sample) 71152

The inspectors concluded that Exelon had taken timely and appropriate actions in accordance with Exelon's operating and alarm response procedures; PI-AA-125, "CAP Procedure"; 10 CFR Part 50, Appendix B; and technical specification requirements. The inspectors determined that Exelon's associated engineering evaluations and trending were sufficiently thorough and based on the best available information, sound judgment, and relevant operating experience. Exelon's assigned corrective actions were aligned with engineering evaluations, adequately tracked, appropriately documented, and completed as scheduled. Based on the documents reviewed, plant walkdowns (including the Unit 2 and Unit 3 hydraulic control units, control rod drive systems, and common control room), and discussions with engineering personnel, the inspectors noted that Exelon personnel identified problems and entered them into the CAP at low threshold.

EXIT MEETINGS AND DEBRIEFS

The inspectors verified no proprietary information was retained or documented in this report.

- On April 10, 2020, the inspectors presented the integrated inspection results to Mr. Matthew Herr, Site Vice President, and other members of the licensee staff.
- On February 20, 2020, the inspectors presented the control rod drive system problem identification and resolution sample inspection results to Mr. Pat Navin, former Site Vice President, and other members of the licensee staff.

Email dated May 14, 2020 from Blake Purnell to David Neff (GenCo-Nuc) and cc to Salgado, Nancy; Rafferty-Czincila, Shannon B:(GenCo-Nuc); david.helker@exeloncorp.com with the subject line of Exelon Generation Company, LLC - Request for Additional Information Regarding Request to Extend Safety Relief Valve Test Interval

By application dated February 4, 2020 (Agencywide Documents Access and Management System Accession No. ML20036D962), Exelon Generation Company, LLC (the licensee) submitted a request in accordance with paragraph 50.55a(z)(1) of Title 10 of the Code of Federal Regulations (10 CFR) for a proposed alternative to the requirements of 10 CFR 50.55a and the American Society of Mechanical Engineers Code for Operation and Maintenance of Nuclear Power Plants at Clinton Power Station (Clinton), Unit No. 1; Dresden Nuclear Power Station, Units 2 and 3; Nine Mile Point Nuclear Station, Unit 2; Peach Bottom Atomic Power Station, Units 2 and 3; and Quad Cities Nuclear Power Station, Units 1 and 2. The proposed alternative would allow the licensee to extend the safety relief valve test interval at these facilities. However, the U.S. Nuclear Regulatory Commission (NRC) staff notes that the request for Clinton is only for the remainder of the current inservice testing interval, which ends in June 2020. Thus, the proposed alternative is unlikely to benefit Clinton unless the request is revised.

The NRC staff is reviewing the application and has determined that additional information is needed to complete the review. A response to the attached request for additional information is requested to be provided within 30 days from the date of this email. If you have any questions, please contact me at (301) 415-1380.

REQUEST FOR ADDITIONAL INFORMATION
EXELON GENERATION COMPANY, LLC
PROPOSED ALTERNATIVE TO EXTEND SAFETY RELIEF VALVE TEST INTERVAL
DOCKET NOS. 50-461, 50-237, 50-249, 50-410, 50-277, 50-278, 50-254, AND 50-265

By application dated February 4, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20036D962), Exelon Generation Company, LLC (the licensee) submitted a request in accordance with paragraph 50.55a(z)(1) of Title 10 of the *Code of Federal Regulations* (10 CFR) for a proposed alternative to the requirements of 10 CFR 50.55a and the American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (OM Code) at Clinton Power Station (Clinton), Unit No. 1; Dresden Nuclear Power Station (Dresden), Units 2 and 3; Nine Mile Point Nuclear Station,

Unit 2 (NMP-2); Peach Bottom Atomic Power Station (Peach Bottom), Units 2 and 3; and Quad Cities Nuclear Power Station (Quad Cities), Units 1 and 2. The proposed alternative would allow the licensee to extend the safety relief valve (SRV) test interval at these facilities.

The U.S. Nuclear Regulatory Commission (NRC) has reviewed the application and determined that the information below is needed to complete its review.

Request for Additional Information (RAI) 1

Currently, each of the licensee's facilities is required to test at least 20 percent of the SRVs every 24 months. As an alternative to this requirement, the licensee proposes to test 40 percent of the SRVs at each facility within a 48-month interval. For each facility, the SRV models affected by the proposed alternative are listed in the table below. Under the proposed alternative, it is possible for more than 24 months to elapse between tests of an SRV model.

[redacted]

Facility

Clinton
Dresden Units 2 and 3 NMP-2
Peach Bottom Units 2 and 3 Quad Cities Units 1 and 2

SRV Models

Dikkers Model G-471
Target Rock 3-Stage Model 67F Dikkers Model G-471
T arget Rock Models 73-67F and 74-67F Target Rock 3-Stage Model 74-67F and
Dresser Model 3777Q

Describe any plans to coordinate and share data regarding the SRV testing program at different units and sites that have the same SRV model. Describe any measures to obtain information on the performance of the various model SRVs at intervals more frequent than once every 48 months, such as staggering the testing at different reactor units that have the same SRV model.

RAI 2

For each facility, Exelon is requesting an alternative to the requirements in paragraph I-1320(a) of the ASME OM Code, Mandatory Appendix I. However, the facilities are not all on the same edition and addenda of the ASME OM Code. Currently, the 2004 Edition through 2006

Addenda of the ASME OM Code is applicable to Dresden and Quad Cities, and the 2012 Edition of the ASME OM Code is applicable to Clinton, NMP-2, and Peach Bottom.

Paragraph I-1320(a) of the 2004 Edition of the ASME OM Code, Mandatory Appendix I, states:

Class 1 pressure relief valves shall be tested at least once every 5 years, starting with initial electric power generation. No maximum limit is specified for the number of valves to be tested within each interval; however, a minimum of 20% of the valves from each valve group shall be tested within any 24-month interval. This 20% shall consist of valves that have not been tested during the current 5-year interval, if they exist. The test interval for any individual valve shall not exceed 5 years.

Paragraph I-1320(a) of the 2012 Edition of the ASME OM Code, Mandatory Appendix I, states:

Class 1 pressure relief valves shall be tested at least once every 5 yr [years], starting with initial electric power generation. No maximum limit is specified for the number of valves to be tested within each interval; however, a minimum of 20% of the valves from each valve group shall be tested within any 24-mo [month] interval. This 20% shall consist of valves that have not been tested during the current 5-yr interval, if they exist. The test interval for any installed valve shall not exceed 5 yr. The 5-yr test interval shall begin from the date of the as-left set pressure test for each valve.

Describe any differences in the implementation of the proposed alternative between sites that use the 2004 Edition of the ASME OM Code and sites that use the 2012 Edition of the ASME OM Code.

RAI 3

The proposed alternative relies, in part, on the implementation of the Exelon SRV Best Practices Maintenance program at the facilities. However, the application only provides limited

information about this program. On June 4, 2019, Exelon described its SRV[1] Best Practices

Maintenance program at a public pre-application meeting for the proposed alternative (see ADAMS Accession No. ML19162A027). The Exelon presentation at the meeting identified four pillars of the program: (1) spring testing, which includes physical dimension measurements and compression rate evaluation; (2) SRV lapping techniques and tools; (3) SRV set pressure adjustment methodology precision; and (4) Target Rock SRV average delay time trending performance improvement.

Describe the SRV Best Practices Maintenance program and how it will be implemented to support the proposed alternative. The response should discuss each of the four pillars mentioned in the June 4, 2019, presentation.

[1] SRVs are also referred to as main steam safety valves (MSSVs) in Exelon's presentation.

May 20, 2020 – Letter from Craig G. Erlanger, Director Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to Bryan C. Hanson Senior Vice President Exelon Generation Company, LLC President and Chief Nuclear Officer Exelon Generation Company, LLC with a subject of PEACH BOTTOM ATOMIC POWER STATION, UNITS 2, AND 3 – EXEMPTION REQUEST FROM CERTAIN REQUIREMENTS OF 10 CFR PART 73, APPENDIX B, "GENERAL CRITERIA FOR SECURITY PERSONNEL" (EPID L-2020-LLE-0055)

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has approved the below temporary exemption from specific requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 73, Appendix B, Section VI, "Nuclear Power Reactor Training and Qualification Plan for Personnel Performing Security Program Duties," for the Peach Bottom Atomic Power Station, Units 2 and 3 (Peach Bottom or PBAPS). This action is in response to Exelon Generation Company, LLC's application dated May 8, 2020 (Agencywide Documents Access and Management System Accession No. ML20129K011 (non-public, withheld under

10 CFR 2.390)), that requested an exemption from 10 CFR Part 73, Appendix B, VI.C.3.(l)(1), regarding annual force-on-force (FOF) exercises.

The requirements in 10 CFR Part 73, Appendix B, Section VI, subsection C.3.(l)(1), state, in part:

Each member of each shift who is assigned duties and responsibilities required to implement the safeguards contingency plan and licensee protective strategy participates in at least . . . one (1) force-on-force exercise on an annual basis. Force-on-force exercises conducted to satisfy the NRC triennial evaluation requirement can be used to satisfy the annual force-on-force requirement for the personnel that participate in the capacity of the security response organization.

The purpose of the annual licensee-conducted FOF exercise is to ensure that the site security force maintains its contingency response readiness.

On January 31, 2020, the U.S. Department of Health and Human Services declared a Coronavirus Disease 2019 (COVID-19) public health emergency (PHE) for the United States. Subsequently, the Centers for Disease Control and Prevention issued recommendations (e.g., social distancing, limiting assemblies) to limit the spread of COVID-19.

In your May 8, 2020, application, you stated the following:

- • PBAPS will continue performing the quarterly tactical response drills/exercises and is not requesting an exemption from these requirements at this time.
- • This temporary exemption supports isolation restrictions (e.g., social distancing, group size limitations, self-quarantining, use of personal protective equipment, etc.) necessary to protect required site personnel in response to the 2020 COVID-19 virus. These restrictions are needed to ensure personnel are isolated form the COVID-19 virus and remain capable of maintaining plant security. PBAPS began implementing isolation restrictions after the Governor of the Commonwealth of Pennsylvania issued a disaster declaration on March 6, 2020.
- • PBAPS will maintain a list of the names of the individuals who will not meet the requalification requirements and will include the dates of the last qualification, and will ensure contingency response readiness of security personnel not participating in an annual FOF exercise by conducting the scenario-based evolutions that include the following: a table-top exercise, a communication-based exercise, a lessons-learned review of past exercise, a walkdown of previous exercise route of travel. PBAPS will complete the FOF exercise, within the time period in this request, when isolation restrictions are ended.
- • PBAPS will begin implementing COVID-19 PHE controls for managing personnel performing security program duties upon NRC approval of its request.

  This temporary exemption will apply to Peach Bottom security personnel who have previously been and currently are qualified in accordance with the requirements in 10 CFR Part 73, Appendix B, Section VI. You also stated that given the rigorous nature of the Peach Bottom nuclear security personnel training programs, it is reasonable to conclude that security personnel will continue to maintain their proficiency even though the requalification periodicity is temporarily exceeded. Additionally, you stated that the site-specific COVID-19 PHE controls listed above will be implemented at Peach Bottom to ensure impacted security personnel maintain the knowledge, skills, and abilities required to effectively perform assigned duties and responsibilities. You

  Pursuant to 10 CFR 73.5, “Specific exemptions,” the Commission may, upon application of any interested person or on its own initiative, grant exemptions from 10 CFR Part 73 when the exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest.

  In accordance with 10 CFR 73.5, the Commission may grant an exemption from the regulations in 10 CFR Part 73 that is authorized by law. The NRC staff has reviewed the exemption request and finds that granting the proposed exemption will not result in a violation of the Atomic Energy Act of 1954, as amended, or other laws. Therefore, the NRC staff finds that the exemption is authorized by law.

> In accordance with 10 CFR 73.5, the Commission may grant an exemption from the regulations in 10 CFR Part 73 when the exemption will not endanger life or property or the common defense and security. This exemption will only apply to licensee security personnel who are already satisfactorily qualified on the security requirements in 10 CFR Part 73, Appendix B, Section VI. Based on this fact, and its review of the controls you will implement for the duration of the exemption, including conducting a table-top exercise, a communication-based exercise, a lessons-learned review of past exercise, a walkdown of previous exercise route of travel, and completing the FOF exercise, within the time period in this request, the NRC staff has reasonable assurance that the security force at Peach Bottom will maintain its proficiency and its readiness to implement the licensee's protective strategy and adequately protect the site. Therefore, the NRC staff concludes that the proposed exemption would not endanger life or property or the common defense and security.

In accordance with 10 CFR 73.5, the Commission may grant an exemption from the regulations in 10 CFR Part 73 when the exemption is in the public interest. Participation in tactical drills and FOF exercises places site security personnel in close proximity to one another. Such proximity has the potential to increase the likelihood of security personnel being exposed to the COVID-19 virus. The NRC staff finds that the temporary exemption from the annual FOF requirement in 10 CFR Part 73, Appendix B, Section VI, subsection C.3.(I)(1), would facilitate the licensee's efforts to maintain a healthy workforce capable of operating the plant safely and implementing the site's protective strategy by isolating security personnel from potential exposure to the COVID-19 virus. The NRC staff concludes that granting the temporary exemption is in the public interest because it allows the licensee to maintain the required security posture at Peach Bottom while enabling the facility to continue to provide electrical power to the Nation.

Environmental Considerations

NRC approval of this exemption request is categorically excluded under 10 CFR 51.22(c)(25), and there are no special circumstances present that would preclude reliance on this exclusion. The NRC staff determined, per 10 CFR 51.22(c)(25)(vi)(E), that the requirements from which the exemption is sought involve education, training, experience, qualification, requalification, or other employment suitability requirements. The NRC staff also determined that approval of this exemption request involves no significant hazards consideration because it does not authorize any physical changes to the facility or any of its safety systems, nor does it change any of the assumptions or limits used in the facility licensee's safety analyses or introduce any new failure modes; no significant change in the types or significant increase in the amounts of any effluents that may be released offsite because this exemption does not affect any effluent release limits as provided in the facility licensee's technical specifications or by the regulations in

10 CFR Part 20, "Standards for Protection Against Radiation"; no significant increase in individual or cumulative public or occupational radiation exposure because this exemption does not affect limits on the release of any radioactive material or the limits provided in
10 CFR Part 20 for radiation exposure to workers or members of the public; no significant construction impact because this exemption does not involve any changes to a construction permit; and no significant increase in the potential for or consequences

from radiological accidents because this exemption does not alter any of the assumptions or limits in the facility licensee's safety analysis. In addition, the NRC staff determined that there would be no significant impacts to biota, water resources, historic properties, cultural resources, or socioeconomic conditions in the region. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the approval of this exemption request.

Conclusions

Accordingly, the NRC has determined that pursuant to 10 CFR 73.5, the exemption is authorized by law, will not endanger life or property or the common defense and security, and is otherwise in the public interest. Therefore, the Commission hereby grants the licensee's request to temporarily exempt Peach Bottom from the annual FOF exercise requalification requirement of security personnel in subsection C.3.(I)(1) of 10 CFR Part 73, Appendix B, Section VI. This exemption expires 90 days after the end of the PHE, or December 31, 2020, whichever occurs first.

If you have any questions, please contact the Peach Bottom project manager, Jennifer Tobin, at 301-415-2328 or Jennifer.Tobin@nrc.gov.

June 25, 2020 – email from Jennifer Tobin to David P. Helker and cc to Richard W. Gropp, Jr. with a subject of Acceptance Review for Peach Bottom - License Amendment Request for TSTF-505 (EPID L-2020-LLA-0120)

By letter dated May 29, 2020 (ADAMS Accession Nos. ML20150A007), Exelon Generation Company, LLC submitted a license amendment request to amend the Technical Specifications for Peach Bottom, Units 2 and 3. The proposed amendments would modify Technical Specification requirements to permit the use of risk-informed completion times in accordance with Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF [Risk-Informed TSTF] Initiative 4b," dated July 2, 2018 (ADAMS Accession No. ML18183A493).

The purpose of this e-mail is to provide the results of the Nuclear Regulatory Commission (NRC) staff's acceptance review of this license amendment request. The acceptance review was performed to determine if there is sufficient technical information in scope and depth to allow the NRC staff to complete its detailed technical review for this licensing action. The acceptance review is also intended to identify whether the license amendment request has any readily apparent information insufficiencies in the characterization of the regulatory requirements or the licensing basis of the plant.

The NRC staff has reviewed your license amendment request and concluded that it provides technical information in sufficient detail to enable the NRC staff to complete its detailed technical review and make an independent assessment regarding the acceptability of the license amendment request in terms of protection of public health and safety and the environment. Given the lesser scope and depth of the acceptance review as compared to the detailed technical review, there may be instances in which issues that impact the NRC staff's ability to complete the detailed technical review are identified despite completion of an adequate acceptance review. You will be advised of any further information needed to support the NRC staff's detailed technical review by separate correspondence.

Based on the information provided in your submittal, the NRC staff has estimated that this license amendment request will take a total of approximately 1815 hours to complete. The NRC staff expects to complete this review by June 30, 2021, as you requested. If there are emergent complexities or challenges in our review that would cause changes to the initial forecasted completion date (greater than a month) or significant changes in the forecasted hours (greater than 25%), the reasons for the changes, along with the new estimates, will be communicated during the routine interactions with the assigned project manager. These estimates are based on the NRC staff's initial review of the application and they could change, due to several factors including requests for additional information, unanticipated addition of scope to the review, and review by NRC advisory committees or hearing-related activities. Additional delay may occur if the submittal is provided to the NRC in advance or in parallel with industry program initiatives or pilot applications.

Please contact me if you have any questions. A copy of this email will be made publicly available in ADAMS.

June 30, 2020 – Letter from Blake A. Purnell, Project Manager Plant Licensing Branch III Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to Bryan C. Hanson Senior Vice President Exelon Generation Company, LLC President and Chief Nuclear Officer (CNO) Exelon Nuclear with subject of DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3; LASALLE COUNTY STATION, UNITS 1 AND 2; LIMERICK GENERATING STATION, UNITS 1
AND 2; NINE MILE POINT NUCLEAR STATION, UNIT 2; PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3; AND QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2 - INDIVIDUAL NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENTS TO RENEWED FACILITY OPERATING LICENSES, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY TO REQUEST A HEARING (EPID L-2020-LLA-0096)

The U.S. Nuclear Regulatory Commission has forwarded the enclosed notice of consideration of issuance of amendments to renewed facility operating licenses, proposed no significant hazards consideration determination, and opportunity to request a hearing to the Office of the Federal Register for publication.

This notice relates to your April 30, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20121A274), license amendment request for Dresden Nuclear Power Station, Units 2 and 3; LaSalle County Station, Units 1 and 2; Limerick Generating Station, Units 1 and 2; Nine Mile Point Nuclear Station, Unit 2; Peach Bottom Atomic Power Station, Units 2 and 3; and Quad Cities Nuclear Power Station, Units 1 and 2. The proposed amendments would revise the technical specifications (TSs) for each facility based on Technical Specifications Task Force (TSTF) traveler TSTF-568, Revision 2, "Revise Applicability of BWR/4 TS 3.6.2.5 and TS 3.6.3.2" (ADAMS Accession No. ML19141A122). The proposed amendments would also make other administrative changes to the TSs.

If you have any questions, please contact me at 301-415-1380.

NUCLEAR REGULATORY COMMISSION
[Docket Nos. 50-237, 50-249, 50-373, 50-374, 50-352, 50-353, 50-410, 50-277, 50-278, 50-254, and 50-265; NRC-2020-0151]
Exelon Generation Company, LLC; Dresden Nuclear Power Station, Units 2 and 3; LaSalle County Station, Units 1 and 2; Limerick Generating Station, Units 1 and 2; Nine Mile Point Nuclear Station, Unit 2; Peach Bottom Atomic Power Station, Units 2 and 3; and Quad Cities Nuclear Power Station, Units 1 and 2

AGENCY: Nuclear Regulatory Commission.
ACTION: License amendment application; opportunity to comment, request a hearing, and petition for leave to intervene.
SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) is considering issuance of amendments to the facility operating licenses for the following facilities operated by Exelon Generation Company, LLC: Dresden Nuclear Power Station (Dresden), Units 2 and 3; LaSalle County Station (LaSalle), Units 1 and 2; Limerick Generating Station (Limerick), Units 1 and 2; Nine Mile Point Nuclear Station (Nine Mile Point), Unit 2; Peach Bottom Atomic Power Station (Peach Bottom), Units 2 and 3; and Quad Cities Nuclear Power Station (Quad Cities), Units 1 and 2. The proposed amendments would revise technical specification (TS) requirements for certain physical parameters at each facility.
DATES: Submit comments by **[INSERT DATE 30 DAYS FROM DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**. Requests for a hearing or petitions for leave to intervene must be filed by **[INSERT DATE 60 DAYS FROM DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**.
ADDRESSES: You may submit comments by any of the following methods:

Federal Rulemaking Web Site: Go to https://www.regulations.gov and search for Docket ID **NRC-2020-0151**. Address questions about NRC docket IDs in Regulations.gov to Jennifer Borges-Roman; telephone: 301-287-9127; e-mail: Jennifer.Borges@nrc.gov. For technical questions, contact the individual listed in the **FOR FURTHER INFORMATION CONTACT** section of this document.

Mail comments to: Office of Administration, Mail Stop: TWFN-7-A60M, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, ATTN: Program Management, Announcements and Editing Staff.

For additional direction on obtaining information and submitting comments, see "Obtaining Information and Submitting Comments" in the SUPPLEMENTARY INFORMATION section of this document.
FOR FURTHER INFORMATION CONTACT: Blake A. Purnell, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington DC 20555-0001; telephone: 301-415-1380, e-mail: Blake.Purnell@nrc.gov.

SUPPLEMENTARY INFORMATION:
I. Obtaining Information and Submitting Comments

A. Obtaining Information
Please refer to Docket ID **NRC-2020-0151** when contacting the NRC about the

availability of information for this action. You may obtain publicly-available information related to this action by any of the following methods:

Federal Rulemaking Web Site: Go to https://www.regulations.gov and search for Docket ID **NRC-2020-0151**.

NRC's Agencywide Documents Access and Management System (ADAMS): You may obtain publicly-available documents online in the ADAMS Public Documents collection at https://www.nrc.gov/reading-rm/adams.html. To begin the search, select "Begin Web-based ADAMS Search." For problems with ADAMS, please contact the NRC's Public Document Room (PDR) reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to pdr.resource@nrc.gov. The ADAMS accession number for each document referenced (if it is available in ADAMS) is provided the first time that it is mentioned in this document. The license amendment request from Exelon Generation Company, LLC, dated April 30, 2020, is available in ADAMS under Accession No. ML20121A274.

B. Submitting Comments

Please include Docket ID **NRC-2020-0151** in your comment submission.

The NRC cautions you not to include identifying or contact information that you do not want to be publicly disclosed in your comment submission. The NRC will post all comment submissions at https://www.regulations.gov as well as enter the comment submissions into ADAMS. The NRC does not routinely edit comment submissions to remove identifying or contact information.

If you are requesting or aggregating comments from other persons for submission to the NRC, then you should inform those persons not to include identifying or contact information that they do not want to be publicly disclosed in their comment submission. Your request should state that the NRC does not routinely edit comment submissions to remove such information before making the comment submissions available to the public or entering the comment into ADAMS.

II. Introduction

The NRC is considering issuance of amendments to the facility operating licenses for the following boiling-water reactors (BWRs) operated by Exelon Generation Company, LLC: Dresden, Units 2 and 3, located in Grundy County, Illinois; LaSalle, Units 1 and 2, located in LaSalle County, Illinois; Limerick, Units 1 and 2, located in Montgomery County, Pennsylvania; Nine Mile Point, Unit 2, located in Oswego County, New York; Peach Bottom, Units 2 and 3, located in York and Lancaster Counties, Pennsylvania; and Quad Cities, Units 1 and 2, located in Rock Island County, Illinois.

The proposed amendments would revise certain TS requirements for the following physical parameters: (1) the drywell-to-suppression chamber differential pressure at Dresden and Quad Cities; (2) the primary containment oxygen concentration at Dresden, LaSalle, Nine Mile Point, Peach Bottom, and Quad Cities; and (3) the drywell and suppression chamber oxygen concentration at Limerick. The proposed changes are based, in part, on Technical Specifications Task Force (TSTF) traveler TSTF-568, Revision 2, “Revise Applicability of BWR/4 TS 3.6.2.5 and TS 3.6.3.2” (ADAMS Accession No. ML19141A122).

Before any issuance of the proposed license amendments, the NRC will need to make the findings required by the Atomic Energy Act of 1954, as amended (the Act), and NRC’s regulations.

The NRC has made a proposed determination that the license amendment request involves no significant hazards consideration. Under the NRC’s regulations in § 50.92 of title 10 of the *Code of Federal Regulations* (10 CFR), this means that operation of the facilities in accordance with the proposed amendments would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee’s analysis against the standards of 10 CFR 50.92(c). The NRC staff’s analysis is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change revises certain TS requirements for the following physical parameters: (1) the drywell-to-suppression chamber differential pressure at Dresden and Quad Cities; (2) the primary containment oxygen concentration at Dresden, LaSalle, Nine Mile Point, Peach Bottom, and Quad Cities; and (3) the drywell and suppression chamber oxygen concentration at Limerick. Specifically, the proposed change revises the applicability of the limiting conditions for operation (LCOs) for these parameters and the remedial actions to be taken when these LCOs are not met. The TS limits on these parameters are not affected by the proposed change. These parameters are not initiators to any accident previously evaluated. As a result, the probability of any accident previously evaluated is not affected by the proposed change.

The mitigation of some accidents previously evaluated includes assumptions regarding these physical parameters. The applicability of the LCOs related to oxygen concentration is changed from Mode 1 (Operational Condition 1 for Limerick) when thermal power is greater than 15 percent to Modes 1 and 2 (Operational Conditions 1 and 2 for Limerick). This expands the applicability of the LCOs related to oxygen concentration for each facility and will not affect the consequences of an accident.

The existing exceptions in the applicability of the LCOs for the subject physical parameters are removed. For each subject parameter, if the LCO is not met, then the licensee must either restore the parameter to within the specified limit or be in a mode or condition where the LCO is not applicable. The proposed change includes increasing the completion times for these actions. The consequences of an event that could affect the subject parameters are no different during the proposed completion times than the consequences of the same event during the existing completion times. A note referencing LCO 3.0.4.c is added to the TS actions to permit entering a mode or condition where the LCOs for the subject parameters are applicable but not met. The addition of LCO 3.0.4.c has no effect on the consequences of an accident. The changes to the completion times and addition of LCO 3.0.4.c replace the existing applicability exceptions.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change revises the TS requirements for the following physical parameters: (1) the drywell-to-suppression chamber differential pressure at Dresden and Quad Cities; (2) the primary containment oxygen concentration at Dresden, LaSalle, Nine Mile Point, Peach Bottom, and Quad Cities; and (3) the drywell and suppression chamber oxygen concentration at Limerick. Specifically, the proposed change revises the applicability of the LCOs for these parameters and the actions for when these LCOs are not met. The proposed change does not involve a physical alteration of these plants (i.e., no new or different type of equipment will be installed). No credible new failure mechanisms, malfunctions, or accident initiators that would have been considered a design-basis accident in the Updated Final Safety Analysis Report for these plants are created because hydrogen generation is not risk significant for design-basis accidents.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed change revises the TS requirements for the following physical parameters: (1) the drywell-to-suppression chamber differential pressure at Dresden and Quad Cities; (2) the primary containment oxygen concentration at Dresden, LaSalle, Nine Mile Point, Peach Bottom, and Quad Cities; and (3) the drywell and suppression chamber oxygen concentration at Limerick. Specifically, the proposed change revises the applicability of the LCOs for these parameters and the actions for when these LCOs are not met. No safety limits are affected. No LCOs or physical parameter limits are affected. The TS requirements for these parameters assure sufficient safety margins are maintained, and that the design, operation, surveillance methods, and acceptance criteria specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the licensing basis for each plant. The proposed change does not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. As such, there are no changes being made to safety analysis assumptions, safety limits, or limiting safety system settings that would adversely affect plant safety.

Therefore, the proposed change does not involve a significant reduction in a margin of safety. Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the license amendment request involves no significant hazards consideration.

The NRC is seeking public comments on this proposed determination that the license amendment request involves no significant hazards consideration. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendments until the expiration of 60 days after the date of publication of this notice. The Commission may issue the license amendments before expiration of the 60-day notice period if the Commission concludes the amendments involve no significant hazards consideration. In addition, the Commission may issue the amendments prior to the expiration of the 30-day comment period if circumstances change during the 30-day comment period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility. If the Commission takes action prior to the expiration of either the comment period or the notice period, it will publish in the *Federal Register* a notice of issuance. If the Commission makes a final no significant hazards consideration determination, any hearing will take place after issuance. The Commission expects that the need to take this action will occur very infrequently.

III. Opportunity to Request a Hearing and Petition for Leave to Intervene

Within 60 days after the date of publication of this notice, any persons (petitioner) whose interest may be affected by this action may file a request for a hearing and petition for leave to intervene (petition) with respect to the action. Petitions shall be filed in accordance with the Commission’s “Agency Rules of Practice and Procedure” in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.309. The NRC’s regulations are accessible electronically from the NRC Library on the NRC’s Web site at https://www.nrc.gov/reading-rm/doc-collections/cfr/. If a petition is filed, the Commission or a presiding officer will rule on the petition and, if appropriate, a notice of a hearing will be issued.

As required by 10 CFR 2.309(d) the petition should specifically explain the reasons why intervention should be permitted with particular reference to the following general requirements for standing: (1) the name, address, and telephone number of the petitioner; (2) the nature of the petitioner’s right to be made a party to the proceeding; (3) the nature and extent of the petitioner’s property, financial, or other interest in the proceeding; and (4) the possible effect of any decision or order which may be entered in the proceeding on the petitioner’s interest.

In accordance with 10 CFR 2.309(f), the petition must also set forth the specific contentions which the petitioner seeks to have litigated in the proceeding. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner must provide a brief explanation of the bases for the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to the specific sources and documents on which the petitioner intends to rely to support its position on the issue. The petition must include sufficient information to show that a genuine dispute exists with the applicant or licensee on a material issue of law or fact. Contentions must be limited to matters within the scope of the proceeding. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to satisfy the requirements at 10 CFR 2.309(f) with respect to at least one contention will not be permitted to participate as a party. Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene. Parties have the opportunity to participate fully in the conduct of the hearing with respect to resolution of that party's admitted contentions, including the opportunity to present evidence, consistent with the NRC's regulations, policies, and procedures.

Petitions must be filed no later than 60 days from the date of publication of this notice. Petitions and motions for leave to file new or amended contentions that are filed after the deadline will not be entertained absent a determination by the presiding officer that the filing demonstrates good cause by satisfying the three factors in 10 CFR 2.309(c)(1)(i) through (iii). The petition must be filed in accordance with the filing instructions in the "Electronic Submissions (E-Filing)" section of this document.

If a hearing is requested, and the Commission has not made a final determination on the issue of no significant hazards consideration, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to establish when the hearing is held. If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendments and make them immediately effective, notwithstanding the request for a hearing. Any hearing would take place after issuance of the amendments. If the final determination is that the amendment request involves a significant hazards consideration, then any hearing held would take place before the issuance of the amendments unless the Commission finds an imminent danger to the health or safety of the public, in which case it will issue an appropriate order or rule under 10 CFR part 2.

A State, local governmental body, Federally-recognized Indian Tribe, or agency thereof, may submit a petition to the Commission to participate as a party under 10 CFR 2.309(h)(1). The petition should state the nature and extent of the petitioner's interest in the proceeding. The petition should be submitted to the Commission no later than 60 days from the date of publication of this notice. The petition must be filed in accordance with the filing instructions in the "Electronic Submissions (E-Filing)" section of this document, and should meet the requirements for petitions set forth in this section, except that under 10 CFR 2.309(h)(2) a State, local governmental body, or Federally-recognized Indian Tribe, or agency thereof does not need to address the standing requirements in 10 CFR 2.309(d) if the facility is located within its boundaries. Alternatively, a State, local governmental body, Federally-recognized Indian Tribe, or agency thereof may participate as a non-party under 10 CFR 2.315(c).

If a hearing is granted, any person who is not a party to the proceeding and is not affiliated with or represented by a party may, at the discretion of the presiding officer, be permitted to make a limited appearance pursuant to the provisions of 10 CFR 2.315(a). A person making a limited appearance may make an oral or written statement of his or her position on the issues but may not otherwise participate in the proceeding. A limited appearance may be made at any session of the hearing or at any prehearing conference, subject to the limits and conditions as may be imposed by the presiding officer. Details regarding the opportunity to make a limited appearance will be provided by the presiding officer if such sessions are scheduled.

IV. Electronic Submissions (E-Filing)

All documents filed in NRC adjudicatory proceedings, including a request for hearing and petition for leave to intervene (petition), any motion or other document filed in the proceeding prior to the submission of a request for hearing or petition to intervene, and documents filed by interested governmental entities that request to participate under 10 CFR 2.315(c), must be filed in accordance with the NRC's E-Filing rule (72 FR 49139; August 28, 2007, as amended at 77 FR 46562; August 3, 2012). The E- Filing process requires participants to submit and serve all adjudicatory documents over the internet, or in some cases to mail copies on electronic storage media. Detailed guidance on making electronic submissions may be found in the Guidance for Electronic Submissions to the NRC and on the NRC Web site at https://www.nrc.gov/site-help/e- submittals.html. Participants may not submit paper copies of their filings unless they seek an exemption in accordance with the procedures described below.

To comply with the procedural requirements of E-Filing, at least 10 days prior to the filing deadline, the participant should contact the Office of the Secretary by e-mail at hearing.docket@nrc.gov, or by telephone at 301-415-1677, to (1) request a digital identification (ID) certificate, which allows the participant (or its counsel or representative) to digitally sign submissions and access the E-Filing system for any proceeding in which it is participating; and (2) advise the Secretary that the participant will be submitting a petition or other adjudicatory document (even in instances in which the participant, or its counsel or representative, already holds an NRC-issued digital ID certificate). Based upon this information, the Secretary will establish an electronic docket for the hearing in this proceeding if the Secretary has not already established an electronic docket.

Information about applying for a digital ID certificate is available on the NRC's public Web site at https://www.nrc.gov/site-help/e-submittals/getting-started.html. Once a participant has obtained a digital ID certificate and a docket has been created, the participant can then submit adjudicatory documents. Submissions must be in Portable Document Format (PDF). Additional guidance on PDF submissions is available on the NRC's public Web site at https://www.nrc.gov/site-help/electronic-sub-ref-mat.html. A filing is considered complete at the time the document is submitted through the NRC's E-Filing system. To be timely, an electronic filing must be submitted to the E-Filing system no later than 11:59 p.m. Eastern Time on the due date. Upon receipt of a transmission, the E-Filing system time-stamps the document and sends the submitter an e-mail notice confirming receipt of the document. The E-Filing system also distributes an e-mail notice that provides access to the document to the NRC's Office of the General Counsel and any others who have advised the Office of the Secretary that they wish to participate in the proceeding, so that the filer need not serve the document on those participants separately. Therefore, applicants and other participants (or their counsel or representative) must apply for and receive a digital ID certificate before adjudicatory documents are filed so that they can obtain access to the documents via the E-Filing system.

A person filing electronically using the NRC's adjudicatory E-Filing system may seek assistance by contacting the NRC's Electronic Filing Help Desk through the "Contact Us" link located on the NRC's public Web site at https://www.nrc.gov/site- help/e-submittals.html, by e-mail to MSHD.Resource@nrc.gov, or by a toll-free call at 1- 866-672-7640. The NRC Electronic Filing Help Desk is available between 9 a.m. and 6 p.m., Eastern Time, Monday through Friday, excluding government holidays.

Participants who believe that they have a good cause for not submitting documents electronically must file an exemption request, in accordance with 10 CFR 2.302(g), with their initial paper filing stating why there is good cause for not filing electronically and requesting authorization to continue to submit documents in paper format. Such filings must be submitted by: (1) first class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Rulemaking and Adjudications Staff; or (2) courier, express mail, or expedited delivery service to the Office of the Secretary, 11555 Rockville Pike, Rockville, Maryland 20852, Attention: Rulemaking and Adjudications Staff. Participants filing adjudicatory documents in this manner are responsible for serving the document on all other participants. Filing is considered complete by first-class mail as of the time of deposit in the mail, or by courier, express mail, or expedited delivery service upon depositing the document with the provider of the service. A presiding officer, having granted an exemption request from using E-Filing, may require a participant or party to use E-Filing if the presiding officer subsequently determines that the reason for granting the exemption from use of E-Filing no longer exists.

Documents submitted in adjudicatory proceedings will appear in the NRC's electronic hearing docket which is available to the public at https://adams.nrc.gov/ehd, unless excluded pursuant to an order of the Commission or the presiding officer. If you do not have an NRC-issued digital ID certificate as described above, click "cancel" when the link requests certificates and you will be automatically directed to the NRC's electronic hearing dockets where you will be able to access any publicly available documents in a particular hearing docket. Participants are requested not to include personal privacy information, such as social security numbers, home addresses, or personal phone numbers in their filings, unless an NRC regulation or other law requires submission of such information. For example, in some instances, individuals provide home addresses in order to demonstrate proximity to a facility or site. With respect to copyrighted works, except for limited excerpts that serve the purpose of the adjudicatory filings and would constitute a Fair Use application, participants are requested not to include copyrighted materials in their submission.

Attorney for licensee: Tamra Domeyer, Associate General Counsel, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Nancy L. Salgado.
Dated: June 30, 2020.

For the Nuclear Regulatory Commission.

Blake A. Purnell, Project Manager,
Plant Licensing Branch III,
Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation.

August 31, 2020 – Letter from Jonathan E. Greives, Chief Reactor Projects Branch 4 Division of Reactor Projects to Bryan C. Hanson Senior Vice President Exelon Generation Company, LLC President and Chief Nuclear Officer with subject line of PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 – UPDATED INSPECTION PLAN (INSPECTION REPORTS 05000277/2020005 AND 05000278/2020005)

The enclosed inspection plan lists the inspections scheduled through June 30, 2022, for Peach Bottom Atomic Power Station, Units 2 and 3. The NRC provides the inspection plan to allow for the resolution of any scheduling conflicts and personnel availability issues. Routine inspections performed by resident inspectors are not included in the inspection plan. You should be aware that the agency is pursuing potential changes to the Reactor Oversight Process (ROP), including changes to engineering inspections (SECY-18-0113, “Recommendations for Modifying the Reactor Oversight Process Engineering Inspections”), and other changes to the baseline inspection program described in SECY-19-0067, “Recommendations for Enhancing the Reactor Oversight Process.” Should these changes to the ROP be implemented, the engineering and other region-based inspections are subject to change in scope, as well as schedule. The inspections listed during the last twelve months of the inspection plan are tentative and may be revised. The NRC will contact you as soon as possible to discuss changes to the inspection plan should circumstances warrant any changes.

In response to the COVID-19 public health emergency (PHE), the NRC is adjusting inspection plans and schedules in order to safeguard the health and safety of both NRC and licensee staff while still effectively implementing the ROP. Each planned inspection is being carefully reviewed in order to determine if any portions of the inspection can be performed remotely, determine how best to perform on-site portions to minimize personnel health risks, and adjust inspection schedules if needed. This is done in accordance with guidance contained in the May 28, 2020 memo, “Inspection Guidance During Transition From COVID-19 Mandatory Telework” (ML20141L766). For inspections requiring extensive coordination with offsite organizations, such as evaluated emergency preparedness exercises, NRC guidance and frequently asked questions for security and emergency preparedness can be found here: https://www.nrc.gov/about-nrc/covid-19/security-ep/. Similarly, the NRC has developed guidance if force-on-force inspections cannot be completed as scheduled due to an emergency, such as the COVID-19 PHE. Consequently, the NRC plans to conduct an inspection per Inspection Procedure 92707, “Security Inspection of Facilities Impacted by Local, State, or Federal Emergency Where the NRC’s Ability to Conduct Triennial Force On-Force Exercises is Limited,” at your facility during the period. These changes help ensure the health and safety of both NRC and licensee staff while maintaining the NRC’s important safety and security mission during the COVID-19 PHE.

The attached inspection plan is accurate on the date of issuance but remains subject to change based on approval of potential exemption requests or other changes needed due to changing conditions in the COVID-19 PHE. NRC staff will contact your appropriate regulatory affairs staff in order to coordinate inspection planning and scheduling.

NRC Region I plans to conduct an inspection per Inspection Procedure 81311, "Physical Security Requirements for Independent Spent Fuel Storage Installations," at your facility during the period.

Additionally, during this period the NRC will schedule an additional inspection per a revised version Temporary Instruction (TI) 2515/194, "Inspection of the Licensee's Implementation of Industry Initiative Associated with the Open Phase Condition Design Vulnerability in Electrical Power Systems (NRC Bulletin 2012-01,)" for any sites who elect to implement the guidance of the Industry Initiative on Open Phase Condition, Revision 3 (ML19163A176), which included an option for relying on annunciation and operator manual actions instead of automatic protective features to isolate a power supply affected by an open phase condition. Licensees will be individually notified when the NRC schedules these inspections

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390 of the NRC's "Rules of Practice," a copy of this letter will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Please contact me at 610-337-5337 with any questions you have regarding this letter.

IP 22 Inspection Activity Plan Report

Unit	Start	End	Activity	CAC	Title	Staff Count
Design Basis Assurance Inspection - Teams - Peach Bottom Units 2 and 3 6						
2, 3 07/13/2020 07/19/2020 IP 71111.21M 000713 Design Bases Assurance Inspection (Teams)						
2, 3 07/27/2020 08/02/2020 IP 71111.21M 000713 Design Bases Assurance Inspection (Teams)						
TI-193 Hardened Containment Vent 2						
2, 3 08/17/2020 08/21/2020 TI 2515/193 000511 Inspection of the Implementation of EA-13-109: Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation under Severe Accident Conditions						
Resp Protection & Dose Assessment (71124.03/71124.04) 1						
2, 3 08/24/2020 08/28/2020 IP 71124.03 000727 In-Plant Airborne Radioactivity Control and Mitigation						
2, 3 08/24/2020 08/28/2020 IP 71124.04 000728 Occupational Dose Assessment						
LIMITED SCOPE TACTICAL RESPONSE DRILLS - PB 6						
2, 3 08/31/2020 09/04/2020 IP 92707 001711 Security Inspection of Facilities Impacted by a Local, State, or Federal Emergency Where the NRC's Ability to Conduct Triennial Force On-Force Exercises is Limited						
REMP (71124.07) 1						
2, 3 09/21/2020 09/25/2020 IP 71124.07 000731 Radiological Environmental Monitoring Program						
ISI - UNIT 2 1						
2 10/26/2020 10/30/2020 IP 71111.08G 000701 Inservice Inspection Activities (BWR)						
Rad Hazards (71124.01) & PIs (71151) 1						
2, 3 10/26/2020 10/30/2020 IP 71124.01 000725 Radiological Hazard Assessment and Exposure Controls						

2, 3	10/26/2020	10/30/2020	IP 71151	000746	Performance Indicator Verification	
ISFSI- Security Inspection-PB						2
2, 3	11/16/2020	11/20/2020	IP 71130.09	001656	Security Plan Changes	
2, 3	11/16/2020	11/20/2020	IP 81311	000831	Physical Security Requirements for Independent Spent Fuel Storage Installations	
EP EXERCISE INSPECTION - PEACH BOTTOM						5
2, 3	12/07/2020	12/11/2020	IP 71114.01	000716	Exercise Evaluation	
2, 3	12/07/2020	12/11/2020	IP 71114.04	000719	Emergency Action Level and Emergency Plan Changes	
2, 3	12/07/2020	12/11/2020	IP 71151	001397	Performance Indicator Verification	

This report does not include INPO and OUTAGE activities.
This report shows only on-site and announced inspection procedures.

Peach Bottom

07/01/2020 - 06/30/2022

IP 22 Inspection Activity Plan Report

Unit	Start	End	Activity	CAC	Title	Staff Count
EP Program Inspection - Peach Bottom						1
2, 3	02/08/2021	02/12/2021	IP 71114.02	000717	Alert and Notification System Testing	
2, 3	02/08/2021	02/12/2021	IP 71114.03	000718	Emergency Response Organization Staffing and Augmentation System	
2, 3	02/08/2021	02/12/2021	IP 71114.04	000719	Emergency Action Level and Emergency Plan Changes	
2, 3	02/08/2021	02/12/2021	IP 71114.05	000720	Maintenance of Emergency Preparedness	
2, 3	02/08/2021	02/12/2021	IP 71151	001397	Performance Indicator Verification	
PB Requal Inspection with P/F Results						2
2	03/08/2021	03/12/2021	IP 71111.11A	000703	Licensed Operator Requalification Program and Licensed Operator Performance	
2	03/08/2021	03/12/2021	IP 71111.11B	000704	Licensed Operator Requalification Program and Licensed Operator Performance	
Access Control, Equipment Testing and Maintenance, Training, SPR						3
2, 3	04/05/2021	04/09/2021	IP 71130.02	000734	Access Control	
2, 3	04/05/2021	04/09/2021	IP 71130.04	000736	Equipment Performance, Testing, and Maintenance	
2, 3	04/05/2021	04/09/2021	IP 71130.07	000739	Security Training	
2, 3	04/05/2021	04/09/2021	IP 71130.09	001656	Security Plan Changes	
2, 3	04/05/2021	04/09/2021	IP 71151	001338	Performance Indicator Verification	
RETS (71124.06)						1
2, 3	04/12/2021	04/16/2021	IP 71124.06	000730	Radioactive Gaseous and Liquid Effluent Treatment	
FY2021 Peach Bottom Initial Examination						4
2, 3	05/02/2021	05/07/2021	OV	000956	VALIDATION OF INITIAL LICENSE EXAMINATION (OV)	
2, 3	05/30/2021	06/11/2021	EXAD	000500	LICENSE EXAM ADMINISTRATION (EXAD)	

ALARA (71124.02) 1						
2, 3	05/17/2021	05/21/2021	IP 71124.02	000726	Occupational ALARA Planning and Controls	
Fire Protection - Peach Bottom 3						
2, 3	06/07/2021	06/11/2021	IP 71111.21N.05	001646	DO NOT USE: Fire Protection Team Inspection (FPTI)	
2, 3	06/21/2021	06/25/2021	IP 71111.21N.05	001646	DO NOT USE: Fire Protection Team Inspection (FPTI)	
Radwaste (71124.08) 1						
2, 3	07/26/2021	07/30/2021	IP 71124.08	000732	Radioactive Solid Waste Processing & Radioactive Material Handling, Storage, & Transportation	

This report does not include INPO and OUTAGE activities.
This report shows only on-site and announced inspection procedures.

Peach Bottom

07/01/2020 - 06/30/2022

IP 22 Inspection Activity Plan Report

Unit	Start	End	Activity	CAC	Title	Staff Count
PI&R BIENNIAL - PB 4						
2, 3	07/26/2021	07/30/2021	IP 71152B	000747	Problem Identification and Resolution	
2, 3	08/09/2021	08/13/2021	IP 71152B	000747	Problem Identification and Resolution	
HP Instrumentation (71124.05) 1						
2, 3	09/13/2021	09/17/2021	IP 71124.05	000729	Radiation Monitoring Instrumentation	
INSERVICE INSPECTION - UNIT 3 1						
3	10/31/2021	11/06/2021	IP 71111.08G	000701	Inservice Inspection Activities (BWR)	
Rad Hazards (71124.01) and PIs (71151) 1						
2, 3	11/01/2021	11/05/2021	IP 71124.01	000725	Radiological Hazard Assessment and Exposure Controls	
2, 3	11/01/2021	11/05/2021	IP 71151	000746	Performance Indicator Verification	
Design Basis Assurance Inspection - Power Operated Valves - Peach Bottom Units 1 and 2 3						
2, 3	03/21/2022	04/08/2022	IP 71111.21N.02	001645	Design-Basis Capability of Power-Operated Valves Under 10 CFR 50.55a Requirements	
Resp Protection & Dose Assessment (71124.03/71124.04) 1						
2, 3	04/18/2022	04/22/2022	IP 71124.03	000727	In-Plant Airborne Radioactivity Control and Mitigation	
2, 3	04/18/2022	04/22/2022	IP 71124.04	000728	Occupational Dose Assessment	
FY22 Peach Bottom Initial License Examination 4						
2, 3	04/24/2022	04/29/2022	OV	000956	VALIDATION OF INITIAL LICENSE EXAMINATION (OV)	
2, 3	05/15/2022	05/27/2022	EXAD	000500	LICENSE EXAM ADMINISTRATION (EXAD)	
EP EXERCISE INSPECTION - PEACH BOTTOM 4						
2, 3	04/25/2022	04/29/2022	IP 71114.04	000719	Emergency Action Level and Emergency Plan Changes	
2, 3	04/25/2022	04/29/2022	IP 71114.07	000722	Exercise Evaluation - Hostile Action (HA) Event	
2, 3	04/25/2022	04/29/2022	IP 71151	001397	Performance Indicator Verification	
Access Control, Protective Strat, TSR, SPR, PI-PB 4						

2, 3	05/16/2022	05/20/2022	IP 71130.02	000734	Access Control
2, 3	05/16/2022	05/20/2022	IP 71130.05	000737	Protective Strategy Evaluation
2, 3	05/16/2022	05/20/2022	IP 71130.09	001656	Security Plan Changes
2, 3	05/16/2022	05/20/2022	IP 71130.14	000743	Review of Power Reactor Target Sets
2, 3	05/16/2022	05/20/2022	IP 71151	001338	Performance Indicator Verification

This report does not include INPO and OUTAGE activities.
This report shows only on-site and announced inspection procedures.

Peach Bottom

07/01/2020 - 06/30/2022

IP 22 Inspection Activity Plan Report

Unit	Start	End	Activity	CAC	Title	Staff Count
59.59 Inspection - Peach Bottom Units 1 and 2 3						
2, 3	06/13/2022	06/17/2022	IP 71111.17T	000709	Evaluations of Changes, Tests, and Experiments	

This report does not include INPO and OUTAGE activities.
This report shows only on-site and announced inspection procedures.

September 9, 2020 – Letter from Josephine A. Ambrosini, Team Leader Technical Support and Administrative Team Division of Reactor Projects to Bryan C. Hanson Senior Vice President Exelon Generating Company, LLC President and Chief Nuclear Officer Exelon Nuclear with the subject of PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 – TEMPORARY INSTRUCTION 2515/193 INSPECTION REPORT 05000277/2020012 AND 05000278/2020012

On August 20, 2020, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Peach Bottom Atomic Power Station, Units 2 and 3 and discussed the results of this inspection with Mr. Dave Henry, Plant Manager and other members of your staff. The results of this inspection are documented in the enclosed report.

No findings or violations of more than minor significance were identified during this inspection.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with Title 10 of the *Code of Federal Regulations* 2.390, “Public Inspections, Exemptions, Requests for Withholding.”

U.S. NUCLEAR REGULATORY COMMISSION Inspection Report

Docket numbers: 05000277 and 05000278

License numbers: DPR-44 and DPR-56

Report numbers: 05000277/2020012 and 05000278/2020012

Enterprise identifier: I-2020-012-0007

Licensee: Exelon Nuclear

Facility: Peach Bottom Atomic Power Station, Units 2 and 3

Location: Delta, PA 17314

Inspection dates: August 17, 2020 to August 20, 2020

Inspectors: F. Arner, Senior Reactor Analyst
P. Boguszewski, Resident Inspector

Approved by: Josephine A. Ambrosini, Team Leader Technical Support and Administrative Team Division of Reactor Projects

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting Temporary Instruction (TI) 2515/193 inspection at Peach Bottom Atomic Power Station, Units 2 and 3, in accordance with the Reactor Oversight Process. The Reactor Oversight Process is the NRC's program for overseeing the safe operation of commercial nuclear power reactors. Refer to https://www.nrc.gov/reactors/operating/oversight.html for more information.

List of Findings and Violations

No findings or violations of more than minor significance were identified.

Additional Tracking Items

None.

INSPECTION SCOPES

Inspections were conducted using the appropriate portions of the inspection procedures (IPs) in effect at the beginning of the inspection unless otherwise noted. Currently approved IPs with their attached revision histories are located on the public website at http://www.nrc.gov/reading- rm/doc-collections/insp-manual/inspection-procedure/index.html. Samples were declared complete when the IP requirements most appropriate to the inspection activity were met consistent with Inspection Manual Chapter (IMC) 2515, "Light-Water Reactor Inspection Program - Operations Phase." The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel to assess licensee performance and compliance with Commission rules and regulations, license conditions, site procedures, and standards.

Starting on March 20, 2020, in response to the National Emergency declared by the President of the United States on the public health risks of the coronavirus (COVID-19), regional inspectors were directed to begin telework. Regional based inspections were evaluated to determine if all or a portion of the objectives and requirements stated in the IP could be performed remotely. For the inspection documented below portions of the IP were completed remotely as well as on site and all the objectives and requirements for completion of the IP were met.

OTHER ACTIVITIES – TEMPORARY INSTRUCTIONS, INFREQUENT AND ABNORMAL

2515/193 - Inspection of the Implementation of EA-13-109: Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation under Severe Accident Conditions

Inspection of the Implementation of EA-13-109: Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation under Severe Accident Conditions (1 Sample)

(1) PHASE 1
The inspectors evaluated licensee implementation of the appropriate elements of the reliable hardened containment wetwell vent as described in the plant specific submittal(s) and the associated NRC safety evaluation (ADAMS Accession No. ML19015A422). Specifically, the inspectors evaluated licensee implementation of the hardened containment vent system (HCVS) functional requirements, design features, maintenance and testing, quality standards, and programmatic requirements as described in Appendix A of TI 2515/193, Revision 1.

PHASE 2
The inspectors evaluated licensee implementation of the appropriate elements of the reliable wetwell venting strategy as described in the plant specific submittal(s) and the associated NRC safety evaluation (ADAMS Accession No. ML19015A422). Specifically, the inspectors evaluated licensee implementation of severe accident water addition/severe accident water management (SAWA/SAWM) functional requirements, installed instrumentation, maintenance and testing, and programmatic requirements as described in Appendix B of TI 2515/193, Revision 1.

INSPECTION RESULTS

No findings were identified.

EXIT MEETINGS AND DEBRIEFS

The inspectors verified no proprietary information was retained or documented in this report.

• On August 20, 2020, the inspectors presented the Temporary Instruction 2515/193 inspection results to Mr. Dave Henry, Plant Manager and other members of the licensee staff.

September 10, 2020 – Letter from Craig G. Erlanger, Director Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to Bryan C. Hanson Senior Vice President Exelon Generation Company, LLC President and Chief Nuclear Officer Exelon Nuclear with a subject line of PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 – EXEMPTION FROM SELECT REQUIREMENTS OF 10 CFR PART 26 (EPID L-2020-LLE-0138 [COVID-19])

The U.S. Nuclear Regulatory Commission (NRC) has approved the requested exemption from specific requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 26, "Fitness for Duty Programs," Section 26.205, "Work hours," for Peach Bottom Atomic Power Station, Units 2 and 3 (Peach Bottom). This action is in response to the Exelon Generation Company, LLC (Exelon, the licensee) application dated August 28, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20241A078), which cited the March 28, 2020, letter (ADAMS Accession No. ML20087P237) from

Mr. Ho Nieh describing a process to request expedited review of certain exemptions from 10 CFR Part 26 during the Coronavirus Disease 2019 (COVID-19) public health emergency (PHE).

The application provided the following information:

- • A statement that Peach Bottom can no longer meet the work-hour controls of 10 CFR 26.205(d) for certain positions;
- • A list of positions for which Peach Bottom will implement alternative work-hour controls upon the NRC granting the requested exemption. (From this, the NRC has determined the positions for which Peach Bottom will maintain current work-hour controls under 10 CFR 26.205(d)(1)-(d)(7) and the positions for which Exelon is requesting the exemption);
- • The date and time when Peach Bottom will begin implementing site-specific COVID-19 PHE fatigue-management controls for personnel specified in 10 CFR 26.4(a);
- • A statement that Peach Bottom's site-specific COVID-19 PHE fatigue-management controls are consistent with the constraints outlined in the March 28, 2020, letter; and

• A statement that Peach Bottom will establish alternative controls for the management of fatigue during the period of the exemption and that, at a minimum, the controls ensure that for individuals subject to these alternative controls:

○ Individuals will not work more than 16 work hours in any 24-hour period and not more than 86 work hours in any 7-day period, excluding shift turnover;

- A minimum 10-hour break is provided between successive work periods; ○ 12-hour shifts are limited to not more than 14 consecutive days;
- A minimum of 6 days off is provided in any 30-day period; and
- Requirements have been established for behavioral observation and

self-declaration during the period of the exemption.

Therefore, the NRC staff finds that the technical basis for an exemption described in the March 28, 2020, letter is applicable to the Peach Bottom application.

Section 26.9, “Specific exemptions,” of 10 CFR allows the NRC to grant exemptions from the requirements of 10 CFR Part 26 as it determines are authorized by law, will not endanger life or property or the common defense and security, and are otherwise in the public interest.

The NRC determined that the requested exemption is permissible under the Atomic Energy Act of 1954, as amended, and other regulatory requirements. Therefore, the NRC finds that the requested exemption is authorized by law.

The underlying purpose of 10 CFR 26.205(d) is to prevent impairment from fatigue due to duration, frequency, or sequestering of successive shifts. Based on the evaluation provided in the NRC’s March 28, 2020, letter and the criteria discussed above, no new accident precursors are created by utilizing whatever licensee staff resources may be necessary or available during the term of this exemption to respond to a plant emergency and to ensure that the plant maintains a safe and secure status. Therefore, the probability of postulated accidents is not increased. Also, the consequences of postulated accidents are not increased because there is no change in the types of accidents previously evaluated. The requested exemption would allow the utilization of licensee staff resources as may be necessary to maintain safe operation of the plant and to respond to a plant emergency. Therefore, the NRC finds that the requested exemption will not endanger life or property.

The requested exemption would allow the utilization of licensee security staff resources as may be necessary to ensure the common defense and security. Therefore, the NRC finds that the requested exemption will not endanger the common defense and security.

Due to the impacts that the COVID-19 PHE has had on the licensee’s ability to comply with the work-hour controls of 10 CFR 26.205(d), the importance of maintaining the operations of Peach Bottom, and the controls that the licensee has established, the NRC finds that the requested exemption is in the public interest.

Granting the requested exemption is categorically excluded under 10 CFR 51.22(c)(25) and there are no extraordinary circumstances present that would preclude reliance on this exclusion. The NRC staff determined, per 10 CFR 51.22(c)(25)(vi)(I), that the requirements from which the exemption is sought involve other requirements of an administrative, managerial, or organizational nature. The NRC staff also determined that approval of this exemption involves no significant hazards consideration because it does not authorize any physical changes to the facility or any of its safety systems, does not authorize changes to any of the assumptions or limits used in the licensee’s safety analyses, and does not introduce any new failure modes. There is no significant change

in the types or significant increase in the amounts of any effluents that may be released offsite because this exemption does not affect any effluent release limits as provided in the licensee's technical specifications or by the regulations in 10 CFR Part 20, "Standards for Protection Against Radiation." There is no significant increase in individual or cumulative public or occupational radiation exposure because this exemption does not affect limits on the release of any radioactive material or the limits provided in 10 CFR Part 20 for radiation exposure to workers or members of the public. There is no significant construction impact because this exemption does not involve any changes to a construction permit. There is no significant increase in the potential for or consequences from radiological accidents because the exemption does not alter any of the assumptions or limits in the licensee's safety analysis.
In addition, the NRC staff determined that there would be no significant impacts to biota, water resources, historic properties, cultural resources, or socioeconomic conditions in the region. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the approval of this exemption request.

Based on the above, the NRC staff finds that (1) the exemption is authorized by law, (2) the exemption will not endanger life or property or the common defense and security, and (3) the exemption is otherwise in the public interest.

This exemption is effective from October 12, 2020, through December 11, 2020.

September 22, 2022 – Email from Blake Purnell to Thomas Loomis (GenCo-Nuc) and cc to Gudger, David T:(GenCo-Nuc); Salgado, Nancy with a subject of Exelon Generation Company, LLC - Fleet Request to Use Paragraph IWA-5120 of the 2017 Edition of the ASME B&PV Code, Section XI (EPID: L-2020-LLR- 0118)

By application dated September 2, 2020 (ADAMS Accession No. ML20246G739), Exelon Generation Company, LLC (Exelon) submitted a request in accordance with Paragraph 50.55a(g)(4)(iv) of Title 10 of the Code of Federal Regulations (10 CFR) to use a specific provision in a later edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code at Braidwood Station, Units 1 and 2; Byron Station, Unit Nos. 1 and 2; Calvert Cliffs Nuclear Power Plant, Units 1 and 2; Clinton Power Station, Unit No. 1; Dresden Nuclear Power Station, Units 2 and 3; James A. FitzPatrick Nuclear Power Plant; LaSalle County Station, Units 1 and 2; Limerick Generating Station, Units 1 and 2; Nine Mile Point Nuclear Station, Units 1 and 2; Peach Bottom Atomic Power Station, Units 2 and 3; Quad Cities Nuclear Power Station, Units 1 and 2; and R. E. Ginna Nuclear Power Plant. Specifically, Exelon requested approval to use paragraph IWA-5120 of the 2017 Edition of the ASME B&PV Code, Section XI, for periodic system pressure test exemptions at each facility.

The purpose of this email is to provide the results of the U.S. Nuclear Regulatory Commission (NRC) staff's acceptance review of this request. The acceptance review was performed to determine if there is sufficient technical information in scope and depth to allow the NRC staff to complete its detailed technical review. The acceptance review is also intended to identify whether the application has any readily apparent information insufficiencies in its characterization of the regulatory requirements or the licensing basis of the plant.

The NRC staff has reviewed your application and concluded that it provides technical information in sufficient detail to enable the staff to complete its detailed technical review and make an independent assessment regarding the acceptability of the request in terms of regulatory requirements and the protection of public health and safety and the environment. Given the lesser scope and depth of the acceptance review, as compared to the detailed technical review, there may be instances in which issues that impact the staff's ability to complete the detailed technical review are identified despite completion of an adequate acceptance review. You will be advised of any further information needed to support the staff's detailed technical review by separate correspondence.

Based on the information provided in your submittal, the NRC staff estimates that review of this request will take approximately 150 hours to complete. The staff expects to complete its review by September 30, 2021. These estimates are based on the staff's initial review of the application and they could change due to several factors, including requests for additional information. If there are emergent complexities or challenges in our review that would cause changes to the initial forecasted completion date or significant changes to the forecasted hours, I will inform you of the reason for the change and provide the new estimates.

October 21, 2020 – Letter from Jennifer C. Tobin, Project Manager Plant Licensing Branch I Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to Bryan C. Hanson Senior Vice President Exelon Generation Company, LLC President and Chief Nuclear Officer Exelon Nuclear with subject of PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 − REGULATORY VIRTUAL AUDIT PLAN REGARDING LICENSE AMENDMENT REQUEST TO ADOPT TSTF-505, REVISION 2 (EPID L-2020-LLA-0120)

By letter dated May 29, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20150A007), Exelon Generation Company, LLC submitted a license amendment request to adopt Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b," dated July 2, 2018 (ADAMS Accession No. ML18183A493), for the Peach Bottom Atomic Power Station, Units 2 and 3.

The proposed amendments would revise technical specification requirements to permit the use of risk-informed completion times for actions to be taken when limiting conditions for operation are not met.

The U.S. Nuclear Regulatory Commission staff will be conducting a virtual audit from November 9, 2020, to November 13, 2020 (excluding November 11, 2020, which is a Federal holiday), with Exelon Generation Company, LLC staff and associated contractors. The regulatory virtual audit plan is enclosed with this letter.

If you have any questions regarding this matter, please contact me at 301-415-2328 or by e-mail to Jennifer.Tobin@nrc.gov.

REGULATORY VIRTUAL AUDIT PLAN
LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT TSTF-505, REVISION 2
EXELON GENERATION COMPANY, LLC

PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3
DOCKET NOS. 50-277 AND 50-278

1.0 BACKGROUND

By application dated May 29, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20150A007), Exelon Generation Company, LLC (the licensee) submitted a license amendment request (LAR) for Peach Bottom Atomic Power Station, Units 2 and 3 (Peach Bottom). The amendments would revise technical specification (TS) requirements to permit the use of risk-informed completion times (RICTs) for actions to be taken when limiting conditions for operation are not met. The proposed changes are based on Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF [Risk Informed Technical Specification Task Force] Initiative 4b," dated July 2, 2018 (ADAMS Accession No. ML18183A493). The U.S. Nuclear Regulatory Commission (NRC) issued a final model safety evaluation approving TSTF-505, Revision 2, on November 21, 2018 (ADAMS Package Accession No. ML18269A041).

2.0 REGULATORY AUDIT BASES

A regulatory audit is a planned license or regulation-related activity that includes the examination and evaluation of primarily non-docketed information. The audit is conducted with the intent to gain understanding, to verify information, and to identify information that will require docketing to support the basis of a licensing or regulatory decision. Performing a regulatory audit is expected to assist the NRC staff in efficiently conducting its review of the LAR and to gain insights of the licensee's processes and procedures. Information that the NRC staff relies upon to make the safety determination must be submitted on the docket.

The basis of this audit is the Peach Bottom LAR to revise TS requirements to permit the use of RICTs and NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" (SRP), Chapter 19, Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance" (ADAMS Accession No. ML071700658).

The audit will be performed consistent with NRC Office of Nuclear Reactor Regulation Office Instruction LIC-111, Revision 1, "Regulatory Audits," dated October 31, 2019 (ADAMS Accession No. ML19226A274). An audit was determined to be the most efficient approach toward a timely resolution of issues associated with this LAR review, since the NRC staff will have an opportunity to minimize the potential for multiple rounds of requests for additional information and ensure no unnecessary burden will be imposed by requiring the licensee to address issues that are no longer necessary to make a safety determination.

3.0 PURPOSE AND SCOPE

The purpose of this audit is to identify information that the licensee should submit on the docket for NRC staff to make a safety determination and to gain a better understanding of the following areas related to the LAR:

- • calculations, analyses, and bases underlying the LAR;
- • approach for developing and implementing the plant's risk-managed TS program;
- • extent that the LAR is consistent with TSTF-505, Revision 2; Nuclear Energy Institute

  (NEI) Topical Report 06-09, Revision 0-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines, Industry Guidance Document," dated November 6, 2006 (ADAMS Package Accession
  No. ML122860402); and the NRC's Final Safety Evaluation for NEI 06-09, dated May 17, 2007 (ADAMS Accession No. ML071200238);

- • whether the proposed configurations introduce any adverse effects on the ability or capacity of plant equipment to perform its design-basis function(s) when the plant is operated in the proposed TS allowable configuration;
- • technical acceptability of the probabilistic risk assessment (PRA) for use in the application and how plant design features are modeled in the PRA used to support the LAR; and
- • use of the Configuration Risk Management Program tool (i.e., PARAGON) to support RICT program implementation.

  The areas of focus for the regulatory audit are the information contained in the LAR, the audit information needs listed in the following section of this audit plan, and all associated and relevant supporting documentation (e.g., methodology, process information, calculations, etc.). The relevant supporting documents are identified below.

4.0 INFORMATION AND OTHER MATERIAL NECESSARY FOR THE REGULATORY AUDIT

The following documentation should be available to the audit team:

1. the documentation specified in Section 4 of the portal audit plan dated August 4, 2020 (ADAMS Accession No. ML20217L346),
2. PRA notebook regarding component data calculations that address SSC mission times, including the emergency diesel generator split mission times,
3. calculation notebook regarding the tornado missile hazard risk value determinations, and
4. any additional supporting documentation that the licensee may determine is responsive to the NRC staff's above information requests.

5.0 AUDIT TEAM
The members of the audit team are anticipated to be:

- • Jennifer Tobin, Project Manager, NRC/DORL (Jennifer.Tobin@nrc.gov)
- • Todd Hilsmeier, Team Leader, NRC/APLA (Todd.Hilsmeier@nrc.gov)
- • Jeff Circle, NRC/APLA (Jeff.Circle@nrc.gov)
- • Robert Pascarelli, Branch Chief, NRC/APLA (Robert.Pascarelli@nrc.gov)

- • Milton Valentin-Olmeda, NRC/APLC (Milton.Valentin-Olmeda@nrc.gov)
- • Wesley Wu, NRC/APLC (De.Wu@nrc.gov)
- • Robert Vettori, NRC/APLB, (Robert.Vettori@nrc.gov)
- • Bernard Grenier, NRC/APLB (Bernard.Grenier@nrc.gov)
- • Henry Marchlewski, NRC/APLB (Henry.Marchlewski@nrc.gov)
- • Khoi Nguyen, NRC/EEOB (Khoi.Nguyen@nrc.gov)
- • Stephen Wyman, NRC/EEOB (Stephen.Wyman@nrc.gov)
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- • Derek Scully, NRC/SCPB (Derek.Scully@nrc.gov)
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- • Gurjendra Bedi, NRC/EMIB (Gurjendra.Bedi@nrc.gov)
- • Yuken Wong, NRC/EMIB (Yuken.Wong@nrc.gov)
- • Shie-Jeng Peng, NRC/SNSB (Shie-Jeng.Peng@nrc.gov)
- • Mark Wilk, NRC Contractor, Pacific Northwest National Laboratory

  (mark.wilk@pnnl.gov)

6.0 LOGISTICS

The audit will be conducted remotely from November 9, 2020, to November 13, 2020 (excluding November 11, 2020), between 8:30 a.m. and 4:00 p.m. each day. An entrance briefing will be held at the beginning of the audit, and an exit briefing will be held at the end of the audit. Attachment A of the audit plan provides the proposed agenda for the remote audit. Attachment B contains the audit questions that the NRC staff would like to have prepared dialogue. The NRC project manager will coordinate with the licensee any identified changes to the audit schedule and logistics.

7.0 SPECIAL REQUESTS

The NRC staff would like access to the documents listed in Section 4.0 above through an online portal that allows the NRC staff and contractors to access documents via the internet. The following conditions associated with the online portal must be maintained throughout the duration that the NRC staff and contractors have access to the online portal:

- • The online portal will be password-protected, and separate passwords will be assigned to the NRC staff and contractors who are participating in the audit.
- • The online portal will be sufficiently secure to prevent the NRC staff and contractors from printing, saving, downloading, or collecting any information on the online portal.

8.0

Conditions of use of the online portal will be displayed on the login screen and will require acknowledgement by each user.

Username and password information should be provided directly to the NRC staff and contractors. The NRC project manager will provide Exelon the names and contact information of the NRC staff and contractors who will be participating in the audit. All other communications should be coordinated through the NRC project manager.

DELIVERABLES

An audit summary, which may be public, will be prepared within 90 days of the completion of the audit. If the NRC staff identifies information during the audit that is needed to support its regulatory decision, the staff will issue requests for additional information to the licensee after the audit.

ATTACHMENT A
Proposed Audit Agenda (Revision 0)
Peach Bottom Atomic Power Station, Units 2 and 3, License Amendment Request to Adopt TSTF-505, Revision 2

Day 1 – Monday, November 9, 2020 (8:30 am to 4:00 pm)*

- Entrance briefing
    - Opening comments by NRC and Exelon Generation Company, LLC (Exelon)
    - Introductions and logistics
- Real-time risk (RTR) model demonstration by Exelon
- Discuss RTR model and calculation of RICT estimates
    - RTR model (including benchmarking, updating, and how seasonal variations are accounted) **(APLA Questions 07, 08, and 09)**
    - PRA functional determination and RICT estimates
    - Treatment of common cause failures for planned and emergent conditions
    - Identification of risk-management actions **(EEOB Question 05)**
- Discuss Key Principle 5, Maintenance Rule and monitoring **(APLA Question 11)**
- Summary of the day[1]
- NRC staff internal meeting

Day 2 – Tuesday, November 10, 2020 (8:30 am to 4:00 pm)*

- Summary of previous day and review open items

- Discuss internal events PRA technical acceptability

    - I&C diversity and modeling in PRA (**EICB Question 01; APLA Question 06)**
    - EDGs, RCIC, HPSW, and vacuum breakers **(APLA Questions 01 to 04)**
    - Credit for FLEX equipment and actions **(APLA Question 05)**

[1] If discussion topics are completed early, additional discussions for Day 1 may include seismic hazard from Day 4 and/or design-success criteria from Day 4.

- Discuss key assumptions and uncertainties - process **(APLA Question 10)**

- - Summary of the day[2]
- - NRC staff internal meeting
  Wednesday, November 11, 2020 – Veterans Day observed (no audit)

  Day 3 – Thursday, November 12, 2020 (8:30 am to 4:00 pm)*

- • Summary of previous day and review open items
- • Discuss fire PRA technical acceptability **(APLB Questions 01 to 12)**
  - - Summary of the day
  - - NRC staff internal meeting

  Day 4 – Friday, November 13, 2020 (8:30 am to 4:00 pm)*

- • Summary of previous day and review open items
- • Discuss seismic hazard **(APLC Questions 01 to 03)**
- • Discuss design-success criteria **(STSB Questions 01 and 02; EEEB Questions 01 to 04)**
- • Follow-up on any remaining open items
- • Summary of audit and exit meeting (tentatively scheduled for 3:30 pm)

* Lunch will be tentatively scheduled from 12:00 pm – 1:00 pm Acronyms:

28705. APLA NRC/NRR/PRA Licensing Branch A
28706. APLB NRC/NRR/PRA Licensing Branch B
28707. APLC NRC/NRR/PRA Licensing Branch C

EDG Emergency Diesel Generator

EEEB NRC/NRR/Electrical Engineering Branch EICB NRC/NRR/Instrumentation & Controls Branch FLEX Flexible Mitigation Strategies
HPSW High-Pressure Service Water
I&C Instrumentation and Control
NRC U.S. Nuclear Regulatory Commission

[2] If discussion topics are completed early, additional discussions for Day 2 may include seismic hazard from Day 4, design-success criteria from Day 4, and/or fire PRA from Day 3, if not discussed earlier.

NRR Office of Nuclear Reactor Regulation PRA Probabilistic Risk Assessment
RCIC Reactor Core Isolation Cooling System RICT Risk-informed Completion Time

RTR Real-time Risk
STSB NRC/NRR/Technical Specifications Branch

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ATTACHMENT B

AUDIT QUESTIONS
LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT TSTF-505, REVISION 2
EXELON GENERATION COMPANY, LLC
PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3
DOCKET NOS. 50-277 AND 50-278

By application dated May 29, 2020, Exelon Generation Company, LLC (the licensee) submitted a license amendment request (LAR) for Peach Bottom Atomic Power Station, Units 2 and 3 (Peach Bottom) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20150A007). The amendment would revise technical specification (TS) requirements to permit the use of risk-informed completion times (RICTs) for actions to be taken when limiting conditions for operation (LCOs) are not met. The proposed changes are based on Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, “Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b,” dated July 2, 2018 (ADAMS Accession No. ML18183A493). The U.S. Nuclear Regulatory Commission (NRC) issued a final model safety evaluation (SE) approving TSTF 505, Revision 2, on November 21, 2018 (ADAMS Accession No. ML18269A041). The NRC staff has determined that the following information is needed in order to complete its review.

Probabilistic Risk Assessment Licensing Branch A (APLA) Audit Questions

Regulatory Guide (RG) 1.174, Revision 3, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis” (ADAMS Accession No. ML17317A256), states that the scope, level of detail, and technical adequacy of the probabilistic risk assessment (PRA) are to be commensurate with the application for which it is intended and the role the PRA results play in the integrated decision process. The NRC’s SE for Nuclear Energy Institute (NEI) Topical Report NEI 06-09, Revision 0-A, “Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines, Industry Guidance Document,” dated November 6, 2006 (ADAMS Package Accession No. ML122860402) (hereafter NEI 06-09), and the NRC’s Final Safety Evaluation for NEI 06-09, dated May 17, 2007 (ADAMS Accession No. ML071200238), state that the PRA models should conform to the guidance in RG 1.200, Revision 1, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities.” The current version is RG 1.200, Revision 2 (ADAMS Accession No. ML090410014), which clarifies the current applicable American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA standard is ASME/ANS RA-Sa-2009, “Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications.” In RG 1.200, the quality of the PRA must be compatible with the safety implications of the proposed TS change and the role the PRA plays in justifying the change. RG 1.200

describes a peer review process using ASME/ANS RA-Sa-2009 as one acceptable approach

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for determining the technical acceptability of the PRA. The primary results of a peer review are the facts and observations (F&Os) recorded by the peer review team and the subsequent resolution of these F&Os. A process to close finding-level F&Os is documented in
Appendix X to the NEI guidance documents NEI 05-04, NEI 07-12, and NEI 12-13, titled "NEI 05-04/07-12/12-[13] Appendix X: Close-out of Facts and Observations (F&Os)" (ADAMS Package Accession No. ML17086A431), which was accepted by the NRC in a letter dated May 3, 2017 (ADAMS Accession No. ML17079A427). NEI 06-09 states that the PRA shall meet Capability Category (CC)-II for the supporting requirements of the PRA standard, and any deviations from these capability categories relative to the RMTS program shall be justified.

APLA QUESTION 01 – Probabilistic Risk Assessment Modeling of Emergency Diesel Generators

As part of its audit (ADAMS Accession No. ML20217L346), the NRC staff noted that the analysis in Section 4.5.2 of PRA Notebook PB-PRA-013 documented the impact of using a "split" mission time of 4 and 8.2 hours for the emergency diesel generators (EDGs). The results of a sensitivity study in Section 4.5.2, which used a PRA model mission time of 24 hours for the EDGs, demonstrated a 3 percent increase in overall core damage frequency (CDF). It is unclear to the NRC staff how the EDGs would only be required for a specific portion of the PRA analysis window of 24 hours. The NRC staff notes that this source of uncertainty does not appear to have been addressed in PRA Notebook PB-MISC-043, which addresses the impact of PRA assumptions on RICT calculations, especially conditions related to alternating current (AC) and direct current (DC) power (e.g., TSs 3.8.1 and 3.8.4). In light of these observations, provide the following information:

1. a) Provide justification for the use of split mission times for the EDGs in the Peach Bottom PRA models. Include in this discussion the reasoning for not using the standard 24-hour mission time used in PRA models.
2. b) Provide the results of RICT sensitivity studies for AC and DC power-related TS LCOs submitted in the LAR that demonstrate the impact of not implementing the 24-hour PRA mission time. Include a discussion of the impact of the split mission times for the EDGs on the RICT calculations.

APLA QUESTION 02 – Probabilistic Risk Assessment Modeling of RCIC Black Start

As part of its audit (ADAMS Accession No. ML20217L346), the NRC staff noted that Table 2-1 of PRA Notebook PB-MISC-043 states, "Systems that normally require DC [power] for operation are not credited for continued operation upon battery depletion"; however, the reactor core isolation cooling system (RCIC) is credited after battery depletion, referred to as "RCIC black start." The analysis in Table 2-1 states that the initial operation of RCIC or high-pressure coolant injection (HPCI) for 2 hours will provide sufficient reactor pressure vessel level to perform the RCIC black start prior to core damage. The analysis assessment states the RCIC black start credit represents “a slight conservative bias.” It is unclear to the NRC staff whether this action is feasible, since the operators have no indication of vessel level or injection flow, and this is a conservative assumption. Provide the following information:

a) Identify which RICT TS LCOs are affected by the credit for RCIC black start.

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2. b) Provide the basis for the feasibility of crediting RCIC after battery depletion (i.e., RCIC black start). Include in this discussion what licensee program directs this action (e.g., emergency operating procedures, severe accident management guidelines, mitigating FLEX strategies).
3. c) Provide the results of RICT sensitivity studies of the associated TS LCOs identified in part (a) that remove credit for RCIC black start. Include a discussion of the impact of this assumption on the RICT calculations.

APLA QUESTION 03 – Probabilistic Risk Assessment Modeling of High-Pressure Service Water

Table E1-1 of LAR Enclosure 1 regarding TS LCO 3.7.1.A (one high-pressure service water (HPSW), subsystem inoperable) states in Note 4 of the table that the HPSW consists of two independent subsystems. Each subsystem contains two HPSW pumps that discharge to both residual heat removal (RHR) heat exchangers. The design-success criteria (DSC) for this TS LCO in Table E1-1 is one of two subsystems; however, the PRA success criteria is one pump and one heat exchanger. It is unclear to the NRC staff whether the PRA success criteria is equivalent to a single subsystem as described in Note 4.

Provide a description of the HPSW system modeling in the Peach Bottom PRAs, and describe the analysis performed to support the PRA success criteria for HPSW.

APLA QUESTION 04 – Probabilistic Risk Assessment Modeling of Vacuum Breakers (Implementation Items)

LAR Attachment 6 lists the following implementation items that must be completed prior to implementation of the RICT program to satisfy the guidance in NEI 06-09 that the PRA reflect the as-built, as-operated plant and that the PRA technical adequacy is acceptable:

- • Exelon will ensure that the reactor building-to-suppression chamber vacuum breakers are modeled in the Peach Bottom PRA with sufficient detail to accurately calculate the RICT.
- • Exelon will ensure that the suppression chamber-to-drywell vacuum breakers are modeled in the Peach Bottom PRA with sufficient detail to accurately calculate the RICT.

LAR Attachment 6 also states that if implementation of any of these changes constitutes a PRA upgrade as defined in the PRA standard, as endorsed by RG 1.200, then a focused-scope peer review will be performed on these changes, and any findings will be resolved and incorporated in the PRA prior to the implementation of the RICT program. However, it is unclear to the NRC staff how the addition of these system models will meet CC-II of the PRA standard, as endorsed by RG 1.200. In light of these observations, provide the following information:

Regarding the implementation items identified above, describe how the associated systems will be adequately modeled in the PRA to CC-II. Include in this discussion:

i. How mechanical components, instrument channels, logic components, and other relevant system components will be modeled.

ii. Provide details of the success criteria for these systems. If the PRA success criteria do not match the DSC, then provide a justification for the PRA success criteria.
iii. ConfirmwhethertheseimplementationitemsapplytoboththeinternaleventsPRA (IEPRA) and the fire PRA (FPRA). Accordingly, adjust the wording for each of the affected implementation items in LAR Attachment 6. If any of these implementation items will not be applied to the FPRA, then justify the position that the FPRA model will be sufficient to support the RICT program.

APLA QUESTION 05 – Probabilistic Risk Assessment Modeling and Uncertainty of FLEX Equipment and Actions

The NRC memorandum dated May 30, 2017, “Assessment of the Nuclear Energy Institute 16-06, ‘Crediting Mitigating Strategies in Risk-Informed Decision Making,’ Guidance for Risk-Informed Changes to Plants Licensing Basis” (ADAMS Accession No. ML17031A269), provides the NRC’s staff assessment of challenges to incorporating FLEX equipment and strategies into a PRA model in support of risk-informed decisionmaking in accordance with the guidance of RG 1.200.

Regarding equipment failure probability in the May 30, 2017, memorandum, the NRC staff concludes (Conclusion 8):

The uncertainty associated with failure rates of portable equipment should be considered in the PRA models consistent with the ASME/ANS PRA Standard as endorsed by RG 1.200. Risk-informed applications should address whether and how these uncertainties are evaluated.

Regarding human reliability analysis (HRA), NEI 16-06, Section 7.5, recognizes that the current HRA methods do not translate directly to human actions required for implementing mitigating strategies. Sections 7.5.4 and 7.5.5 of NEI 16-06 describe such actions to which the current HRA methods cannot be directly applied, such as debris removal, transportation of portable equipment, installation of equipment at a staging location, routing of cables and hoses, and those complex actions that require many steps over an extended period, multiple personnel and locations, evolving command and control, and extended time delays. In the May 30, 2017, memorandum, the NRC staff concludes (Conclusion 11):

Until gaps in the human reliability analysis methodologies are addressed by improved industry guidance, [human error probabilities] HEPs associated with actions for which the existing approaches are not explicitly applicable, such as actions described in Sections 7.5.4 and 7.5.5 of NEI 16-06, along with assumptions and assessments, should be submitted to NRC for review.

Regarding uncertainty, Section 2.3.4 of NEI 06-09 states that PRA modeling uncertainties shall be considered in the application of the PRA base model results to the RICT program and that sensitivity studies should be performed on the base model prior to initial implementation of the RICT program on uncertainties that could potentially impact the results of an RICT calculation. NEI 06-09 also states that the insights from the sensitivity studies should be used to develop appropriate risk management actions (RMAs), including highlighting risk-significant operator actions, confirming availability and operability of important standby equipment, and assessing the presence of severe or unusual environmental conditions. Uncertainty exists in PRA modeling of FLEX related to the equipment failure probabilities for FLEX equipment used in the model, the corresponding operator actions, and pre-initiator failure probabilities. Therefore, FLEX modeling assumptions can be key assumptions and sources of uncertainty for the RICTs proposed in this application.

LAR Enclosure 9, Table E9-1, indicates that FLEX equipment and actions have been credited in the IEPRA. The LAR states that a sensitivity study was performed for the IEPRA to address this issue. The LAR stated that the sensitivity did not significantly impact the RICT values. As part of its audit (ADAMS Accession No. ML20217L346), the NRC staff noted that Section 8 of PRA Notebook PB-MISC-043 provided results of a sensitivity study where the failure probability of the FLEX injection pump and diesel generator was significantly increased. However, the NRC staff notes the significant challenges of modeling FLEX equipment and actions without sufficient industry data and without a consensus HRA approach to address unique aspects of FLEX actions.

The NRC staff also notes that the difference between failure rates associated with permanently installed safety-related diesel generators and portable non-safety-related diesel generators could be greater than a factor of 10 without consideration of further uncertainty. It is unclear to the NRC staff whether the stated sensitivity study addressed the uncertainties associated with estimating HEP values for FLEX actions, especially for non-operator trained actions. Given the observations above, it is not clear whether the sensitivity study performed to assess the impact of crediting FLEX equipment and actions is sufficient to conclude that the impact to the RICT program of the uncertainties associated with modeling FLEX is negligible. For this reason, and to understand the credit that will be taken for FLEX equipment and actions in the RICT program, address the following separately for the IEPRA, internal flooding PRA, and FPRA:

1. a) Provide results of LCO-specific sensitivity studies that assess the removal of FLEX credit on RICT calculations.
2. b) Regarding HRA, address the following items:

i. Discuss whether any credited operator actions related to FLEX equipment contain actions described in Sections 7.5.4 and Sections 7.5.5 of NEI 16-06.

   If any credited operator actions related to FLEX equipment contain actions described in Sections 7.5.4 and Sections 7.5.5 of NEI 16-06, answer either item (ii) or (iii) below.

ii. Justify and provide results of LCO-specific sensitivity studies that assess impact from the FLEX-independent and FLEX-dependent HEPs associated with deploying and staging FLEX portable equipment on the RICTs proposed in this application. As part of the response, include the following information:
   1. Justify independent and joint HEP values selected for the sensitivity studies, including justification of why the chosen values constitute bounding realistic estimates.
   2. Provide numerical results on specific selected RICTs and discussion of the results.
   3. Discuss composite sensitivity studies of the RICT results to the operator action HEPs and the FLEX equipment reliability uncertainty sensitivity study.
   4. Describe how the source of uncertainty due to the uncertainty in FLEX operator action HEPs will be addressed in the RICT program. Describe specific RMAs being proposed and how these RMAs are expected to reduce the risk associated with this source of uncertainty.

iii. Alternatively to item (b)ii above, provide information associated with the following items listed in supporting requirements (SR) HR-G3 and HR-G7 of the PRA standard to support the NRC staff's detailed review of the LAR:

1. the level and frequency of training that the operators and non-operators receive for deployment of the FLEX equipment (performance shaping factor (a) in SR HR-G3),

2. performance shaping factor (f) in SR HR-G3 regarding estimates of time available and time required to execute the response,

3. performance shaping factor (g) in SR HR-G3 regarding complexity of detection, diagnosis, and decisionmaking and executing the required response,

4. performance shaping factor (h) in SR HR-G3 regarding consideration of environmental conditions, and

5. human action dependencies as listed in SR HR-G7 of the PRA standard.

c) The PRA
new methodology or significant changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences. Section 1-5 of Part 1 of the PRA standard states that upgrades of a PRA shall receive a peer review in

accordance with the requirements specified in the peer review section of each respective part of this standard.

i. Provide an evaluation of the model changes associated with incorporating FLEX mitigating strategies that demonstrates that none of the following criteria are satisfied: (1) use of new methodology, (2) change in scope that impacts the significant accident sequences or the significant accident progression sequences, and (3) change in capability that impacts the significant accident sequences or the significant accident progression sequences.
ii. Alternatively to item (c)i above, confirm that the modeling of FLEX equipment and FLEX actions in the PRA has been peer reviewed in accordance with NRC-accepted methods. Provide the findings of the peer review performed on the FLEX modeling and the disposition of the findings as they pertain to the impact on this LAR.

APLA QUESTION 06 – Probabilistic Risk Assessment Modeling and Uncertainty of Digital Instrumentation and Controls

Section 2.3.4 of NEI 06-09 states that PRA modeling uncertainties be considered in application of the PRA base model results to the RICT program. The NRC SE for NEI 06-09 states that this consideration is consistent with Section 2.3.5 of RG 1.177, Revision 1, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications" (ADAMS Accession No. ML100910008). NEI 06-09 further states that sensitivity studies should be performed on the base model prior to initial implementation of the RICT program on uncertainties that could potentially impact the results of an RICT calculation and that sensitivity studies should be used to develop appropriate compensatory RMAs.

1. a) A TS LCO condition listed in LAR Table E1-1 indicates that instrumentation and control (I&C) modeling in the PRA is insufficient to model the condition, and therefore, the inoperability of the associated equipment (e.g., channel) will be modeled using a surrogate event. Furthermore, based on documentation in the LAR for other TS LCO conditions in the RICT program, it is not clear to NRC staff whether I&C is modeled in sufficient detail to support implementation of TSTF-505, Revision 2.

   Describe how I&C equipment that is applicable or that impacts the RICT calculations is modeled/considered in the PRA. Include in this discussion: (1) the scope of the I&C equipment that is explicitly modeled (e.g., bistables, relays, sensors, integrated circuit cards), (2) description of the level of detail that the PRA model supports (e.g., are all channels of an actuation circuit modeled), (3) discussion of the generic data and plant-specific data used, and (4) discussion of the associated TS functions for which an RICT can be applied.

2. b) Regarding digital I&C, the NRC staff notes the lack of consensus industry guidance for modeling these systems in plant PRAs to be used to support risk-informed regulatory applications. In addition, known modeling challenges exist such as lack of industry data for digital I&C components, differences between digital and analog system failure modes, and the complexities associated with modeling software failures, including common cause software failures. Also,

although reliability data from vendor tests may be available, this source of data is not a substitute for in-the-field operational data. Given these challenges, the uncertainty associated with modeling a digital I&C system could impact the RICT program.

Attachment 4 of the LAR identifies digital feedwater control system is employed at the plant. However, the modeling of this digital system is not identified in Enclosure 9 as a source of uncertainty. Therefore, it is not clear to the NRC staff whether the digital feedwater control system is the only digital system credited in the PRA and whether there are other digital systems credited in the PRA that could potentially impact RICT calculations. In light of these observations, provide the following information:

i. Describe and provide the results of a sensitivity study performed for each digital system modeled in the PRA demonstrating that the uncertainty associated with PRA modeling the digital I&C systems has inconsequential impact on the RICT calculations.
ii. As an alternative to item (b)i above, identify which LCOs are determined to be impacted by digital I&C system modeling for which RMAs will be applied during an RICT. Explain and justify the criteria used to determine what level of impact to the RICT calculation requires additional RMAs.

APLA QUESTION 07 – PRA Update Process

Section 2.3.4 of NEI 06-09 specifies, “criteria shall exist in PRA configuration risk management to require PRA model updates concurrent with implementation of facility changes that significantly impact RICT calculations.”

LAR Enclosure 7 states that if a plant change or a discovered condition is identified and can have significant impact on the RICT calculations, then an unscheduled update of the PRA models will be implemented. More specifically, the LAR states that if the plant changes meet specific criteria defined in the plant PRA and update procedures, including criteria associated with consideration of the cumulative risk impact, then the change will be incorporated into applicable PRA models without waiting for the next periodic PRA update. The LAR does not explain under what conditions an unscheduled update of the PRA model will be performed or the criteria defined in the plant procedures that will be used to initiate the update.

In light of these observations, describe the conditions under which an unscheduled PRA update (i.e., more than once every two refueling cycles) would be performed and the criteria that would be used to require a PRA update. In the response, define what is meant by “significant impact to the RICT Program calculations.”

APLA QUESTION 08 – Real-Time Risk Model

Regulatory Position 2.3.3 of RG 1.174 states that the level of detail in the PRA should be sufficient to model the impact of the proposed licensing basis change. The characterization of the problem should include establishing a cause-effect relationship to identify portions of the PRA affected by the issue being evaluated. Full-scale applications of the PRA should reflect this cause-effect relationship in a quantification of the impact of the proposed licensing basis change on the PRA elements.

Section 4.2 of NEI 06-09 describes attributes of the tool used for configuration risk management (CRM). Some of these attributes are listed below.

- • Initiating events accurately model external conditions and effects of out-of-service equipment.
- • Model translation from the PRA to a separate CRM tool is appropriate; CRM fault trees are traceable to the PRA. Appropriate benchmarking of the CRM tool against the PRA model shall be performed to demonstrate consistency.
- • Each CRM application tool is verified to adequately reflect the as-built, as-operated plant, including risk contributors that vary by time of year or time in fuel cycle or otherwise demonstrate to be conservative or bounding.
- • Application-specific risk important uncertainties contained in the CRM model (that are identified via PRA model to CRM tool benchmarking) are identified and evaluated prior to use of the CRM tool for RMTS applications.

- • CRM application tools and software are accepted and maintained by an appropriate quality program.
- • The CRM tool shall be maintained and updated in accordance with approved station procedures to ensure it accurately reflects the as-built, as-operated plant.

Enclosure 8 of the LAR describes the attributes of the real-time-risk (RTR) model (i.e., Peach Bottom's CRM tool) for use in RICT calculations. The LAR explains that the internal events, internal flooding events, and fire events PRA models are maintained as separate models. The LAR also describes several changes made to the PRA models to support calculation of configuration-specific risk and mentions approaches for ensuring the fidelity of the RTR to the PRAs, including RTR maintenance, documentation of changes, and testing. Regarding development and application of the RTR model, provide the following information:

1. a) Describe the process that will be used to maintain the accuracy of any presolved cutsets with changes in plant configuration.
2. b) Describe the benchmarking activities performed to confirm consistency of the RTR model results to the results of each PRA model of record, including periodicity of RTR updates compared to the model of record updates. Address each model of record (i.e., internal events, internal flooding events, and internal fire events) in the response.

APLA QUESTION 09 – Impact of Seasonal Variations on the Real-Time Risk Model

Regulatory Position 2.3.3 of RG 1.174 states that the level of detail in the PRA should be sufficient to model the impact of the proposed licensing basis change. The characterization of the problem should include establishing a cause-effect relationship to identify portions of the PRA affected by the issue being evaluated. Full-scale applications of the PRA should reflect this cause-effect relationship in a quantification of the impact of the proposed licensing basis change on the PRA elements. Additionally, NEI 06-09 states the following:

If the PRA model is constructed using data points or basic events that change as a result of time of year or time of cycle (examples include moderator temperature coefficient, summer versus winter alignments for HVAC, seasonal alignments for service water), then the RICT calculation shall either 1) use the more conservative assumption at all time, or 2) be adjusted appropriately to reflect the current (e.g., seasonal or time of cycle) configuration for the feature as modeled in the PRA.

Section 2 of LAR Enclosure 8 states, "The impact of outside temperatures on system requirements are addressed in the RTR model." As part of its audit (ADAMS Accession No. ML20217L346), the NRC staff noted that PRA Notebook PB-MISC-043 states that two EDG fans are required when ambient temperature is above 80 degrees Fahrenheit (°F); however, the PRA model uses a split fraction to represent the percentage of the year assumed to be over
80 °F in modifying the success criteria. The analysis states that RICT will necessitate identifying specific time periods when two fans are required.

Provide further explanation supporting the statement above by summarizing the plant equipment subject to seasonal variations and how it is modeled in the PRA to remove the seasonal dependency.

APLA QUESTION 10 – Probabilistic Risk Assessment Model Uncertainty Analysis Process

The NRC staff SE to NEI 06-09 specifies that the LAR should identify key assumptions and sources of uncertainty and assess and disposition each as to its impact on the RMTS application. Section 5.3 of NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking, Main Report," dated March 2017 (ADAMS Accession No. ML17062A466), presents guidance on the process of identifying, characterizing, and qualitatively screening model uncertainties.

LAR Enclosure 9 states that the process for identifying key assumptions and sources of uncertainty for the IEPRA (includes internal floods) and FPRA was performed using the guidance in NUREG-1855, Revision 1. The LAR indicates that in addition to reviewing generic industry sources of uncertainty for applicability, the IEPRA and FPRA models and notebooks were reviewed for plant-specific assumptions and sources of uncertainty.

However, for the IEPRA (includes internal floods), it is not clear to the NRC staff what specific process and criteria were used to screen uncertainties from an initial comprehensive list of assumptions and sources of PRA modeling uncertainty (including those associated with plant-specific features, modeling choices, and generic industry concerns) in order to conclude that no uncertainty issues could impact the RICT calculations. The NRC staff notes from review of Enclosure 9 of the LAR that the dispositions to many identified sources of uncertainty highlight the phrase "not significantly impact the RICT values." It is not clear to the NRC staff what this phrase means in all cases. Also, for certain sources of uncertainty, the disposition states that a sensitivity study was performed to evaluate the impact of the uncertainty, but it is not clear what criteria was used to determine when a sensitivity study was performed or when additional RMAs should be considered.

Therefore, address the following regarding the IEPRA (includes internal floods) uncertainties:

1. a) Describe the process used to screen uncertainties from the initial comprehensive lists of PRA uncertainties (including those associated with plant-specific features, modeling choices, and generic industry concerns) in order to eventually conclude that the uncertainty issues could not impact the RICT calculations. Include a description of the criteria that was used to screen down from a comprehensive listing of sources of uncertainty to a smaller set of key candidate assumptions and sources of uncertainty. Also, describe the criteria used to justify that none of the key candidate assumptions and sources of uncertainty could have an impact on the RICT calculations. As part of this description, explain the criteria used to determine when the results of sensitivity studies do not significantly impact RICT values.
2. b) Concerning the evaluation criteria used to evaluate and screen uncertainties addressed in item (a) above:
    i. Discuss the criteria used to consistently determine when a sensitivity study was used to address the identified source of uncertainty.
    ii. Discuss the criteria used to consistently determine when additional RMAs should be implemented because of modeling uncertainty.

APLA QUESTION 11 – Performance Monitoring and Feedback

Section 2.3 of LAR Attachment 1 states that the application of an RICT will be evaluated using the guidance provided in NEI 06-09, which was approved by the NRC on May 17, 2007 (ADAMS Accession No. ML071200238). The NRC SE for NEI 06-09, states, "The impact of the proposed change should be monitored using performance measurement strategies." NEI 06-09 considers the use of NUMARC 93-01, Revision 4F, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants (ADAMS Accession No. ML18120A069), as endorsed by RG 1.160, Revision 4 (ADAMS Accession No. ML18220B281), for the implementation of the Maintenance Rule. NUMARC 93-01, Section 9.0, contains guidance for the establishment of performance criteria.

Furthermore, Section 2.3 of LAR Attachment 1 states:

In addition, the NEI 06-09-A, Revision 0 methodology satisfies the five key safety principles specified in Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decision-making: Technical Specifications," dated August 1998 (ADAMS Accession No. ML003740176), relative to the risk impact due to the application of a RICT.

NRC staff position C.3.2 provided in RG 1.177 for meeting the fifth key safety principle acknowledges the use of performance criteria to assess degradation of operational safety over a period of time. It is unclear to the NRC staff how the licensee's process for the risk-informed application captures performance monitoring for the structures, systems, and components (SSCs) within-scope of the application. In light of these observations, address either (a) or (b) below.

1. a) Confirm that the Peach Bottom Maintenance Rule program incorporates the use of performance criteria to evaluate SSC performance as described in the NRC-endorsed guidance in NUMARC 93-01.

   OR

2. b) Describe the approach/method used by Peach Bottom for SSC performance monitoring as described in Regulatory Position C.3.2 referenced in RG 1.177 for meeting the fifth key safety principle. In the description, include criteria (e.g., qualitative or quantitative), along with the appropriate risk metrics, and explain how the approach and criteria demonstrate the intent to monitor the potential degradation of SSCs in accordance with the NRC SE for NEI 06-09.

Probabilistic Risk Assessment Licensing Branch B (APLB) Audit Questions

RG 1.200 states that "NRC reviewers... [will] focus their review on key assumptions and areas identified by peer reviewers as being of concern and relevant to the application." Relatively extensive and detailed reviews of FPRAs were undertaken in support of each LAR to transition to National Fire Protection Association Standard 805 (NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants"). These reviews determined that implementation of some of the complex FPRA methods often used nonconservative and oversimplified assumptions to apply the method to specific plant configurations. Some of these issues were not always identified in F&Os by the peer review teams, but are considered potential key assumptions by the NRC staff. Using more defensible and less simplified assumptions could substantively affect the fire risk and fire risk profile of the plant.

The NRC staff evaluates the acceptability of the PRA for each new risk-informed application and, as discussed in RG 1.174, recognizes that the technical acceptability of risk analyses necessary to support regulatory decisionmaking may vary with the relative weight given to the risk assessment element of the decision-making process. The NRC staff notes that the calculated results of the PRA are used to calculate an RICT, which subsequently determines how long SSCs (both individual SSCs and multiple, unrelated SSCs) controlled by TSs can remain inoperable. Therefore, the PRA results are given a very high weight in a TSTF-505 application, and the NRC staff requests additional information on the following issues that have been previously identified as potentially key FPRA assumptions.

APLB QUESTION 01 – Reduced Transient Heat Release Rates

The key factors used to justify using transient fire-reduced heat release rates (HRRs) below those prescribed in NUREG/CR-6850, "EPRI/NRC Fire PRA Methodology for Nuclear Power Facilities" (ADAMS Accession No. ML052580075), are discussed in a letter from the NRC to NEI, dated June 21, 2012 (ADAMS Package Accession No. ML12172A406).

If any reduced transient HRRs below the bounding 98th percentile HRR of 317 kilowatts (kW) from NUREG/CR-6850 were used, discuss the key factors used to justify the reduced HRRs. In this discussion, also provide the following information:

1. a) Identification of the fire areas where a reduced transient fire HRR is credited and what reduced HRR value was applied.
2. b) A description for each location where a reduced HRR is credited, and a description of the administrative controls that justify the reduced HRR, including how location-specific attributes and considerations are addressed. Include a discussion of the required controls for ignition sources in these locations and the types and quantities of combustible materials needed to perform maintenance. Also, include discussion of the personnel traffic that would be expected through each location.
3. c) The results of a review of records related to compliance with the transient combustible and hot work controls.

APLB QUESTION 02 – Joint Human Error Probability Floor

NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines" (ADAMS Accession No. ML12216A104), discusses the need to consider a minimum value for the joint probability of multiple human failure events (HFEs) in HRAs. NUREG-1921 refers to Table 2-1 of NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA)" (ADAMS Accession No. ML051160213), which recommends that joint human error probability (HEP) values should not be below 1E-5. Table 4-4 of Electric Power Research Institute
(EPRI) 1021081, "Establishing Minimum Acceptable Values for Probabilities of Human Failure Events," provides a lower limiting value of 1E-6 for sequences with a very low level of dependence. Therefore, the guidance in NUREG-1921 allows for assigning joint HEPs that are less than 1E-5, but only through assigning proper levels of dependency. TSTF-505 evaluations use the FPRA and IEPRA. The LAR does not provide information about whether and, if so, what minimum joint HEP value(s) is currently assumed in the FPRA. Also, even if the assumed minimum joint HEP value(s) is shown to have no impact on the current FPRA risk estimates, it is not clear to the NRC staff how it will be ensured that the impact remains minimal for future PRA revisions. In light of these observations, provide the following information:

1. a) Explain what minimum joint HEP value(s) was assumed in the FPRA.
2. b) If a minimum joint HEP value less than 1E-05 was used in the FPRA, then provide a description of the sensitivity study that was performed and the quantitative results that justify that the minimum joint HEP value(s) has no impact on the RICT application.
3. c) If, in response to item (b) above, it cannot be justified that the minimum joint HEP value(s) has no impact on the application, confirm that each joint HEP value used in the FPRA below 1E-5 includes its own separate justification that demonstrates the inapplicability of the NUREG-1792 lower value guideline (i.e., using such criteria as the dependency factors identified in NUREG-1921 to assess level of dependence). Provide an estimate of the number of these joint HEP values below the guideline value of 1E-5 for the FPRA, discuss the range of values, and provide at least two different examples where this justification is applied.

APLB QUESTION 03 – Obstructed Plume Model

NUREG-2178, Volume 1, "Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire (RACHELLE-FIRE)" (ADAMS Accession No. ML16110A140), contains refined peak HRRs, compared to those presented in NUREG/CR-6850, and guidance on modeling the effect of plume obstruction. Additionally, NUREG-2178 provides guidance that indicates that the obstructed plume model is not applicable to cabinets in which the fire is assumed to be located at elevations of less than one-half of the cabinet height.

1. a) If obstructed plume modeling was used, then indicate whether the base of the fire was assumed to be located at an elevation of less than one-half of the cabinet height.
2. b) Justify any modeling in which the base of an obstructed plume is located at less than one-half of the cabinet's height.

APLB QUESTION 04 – Systems Not Credited in the Fire PRA

As part of its audit (ADAMS Accession No. ML20217L346), the NRC staff reviewed PRA Notebook PB-PRA-021.62, which noted that several systems were identified as not being modeled in the FPRA. The NRC staff notes that some conservative PRA modeling assumptions could have a nonconservative impact on the RICT calculations. If an SSC is part of a system not credited in the FPRA or is supported by a system that is assumed to always fail, then the risk increases due to taking that SSC out of service are masked. Therefore, provide the following information:

a) Identify the systems or components that are assumed to be always failed in the FPRA or not included in the FPRA (e.g., due to lack of cable tracing or other reasons). Justify that these assumptions have an inconsequential impact on the RICT calculations and no RMAs are required to address these items.

b) As an alternative to item (a) above, propose a mechanism to ensure that a sensitivity study is performed for the RICT calculations for applicable SSCs that accounts for the impact on the RICT of the 1) conservative FPRA assumption of failed SSCs or 2) SSCs not included in the FPRA model. The proposed mechanism should also ensure that any additional risk from correcting these assumptions is either accounted for in the RICT calculations or is compensated for by applying additional RMAs during the RICT.

APLB QUESTION 05 - Implementation Item for Cable Data for Standby Liquid Control

LAR Attachment 6 lists the following implementation item that must be completed prior to implementation of the RICT program to satisfy the guidance in NEI 06-09 that the PRA reflect the as-built, as-operated plant and that the PRA technical adequacy is acceptable:

- Exelon will ensure that the updated standby liquid control cable data will be incorporated in the Peach Bottom PRA with sufficient detail to accurately calculate the RICT.

LAR Attachment 6 also states that if implementation of this change constitutes a PRA upgrade as defined in the PRA standard, as endorsed by RG 1.200, then a focused-scope peer review will be performed on this change, and any findings will be resolved and incorporated in the PRA prior to the implementation of the RICT program. However, it is unclear to the NRC staff how the addition of this system model will meet CC-II of the PRA standard, as endorsed by

RG 1.200.

In light of this observation, describe how this system will be adequately modeled in the FPRA and in accordance with the PRA standard's CC-II.

APLB QUESTION 06 – Well-Sealed Motor Control Center Cabinets

Guidance in Frequently Asked Question (FAQ) 08-0042, "Fire Propagation from Electrical Cabinets" (from Supplement 1 of NUREG/CR-6850), applies to electrical cabinets below
440 volts (V). With respect to Bin 15 as discussed in Chapter 6 of NUREG/CR-6850, it clarifies the meaning of "robustly or well-sealed." Thus, for cabinets of 440 V or less, fires from well-sealed cabinets do not propagate outside the cabinet. For cabinets of 440 V and higher, the original guidance in Chapter 6 remains and requires that Bin 15 panels, which house circuit voltages of 440 V or greater, are counted, because an arcing fault could compromise panel integrity (an arcing fault could burn through the panel sides, but this should not be confused with the high energy arcing fault type fires).

FAQ 14-0009, "Treatment of Well-Sealed MCC Electrical Panels Greater than 440V" (ADAMS Accession No. ML15119A176), provides the technique for evaluating fire damage from MCC cabinets having a voltage greater than or equal to 440 V. Therefore, propagation of fire outside the ignition source panel must be evaluated for all MCC cabinets that house circuits of 440 V or greater.

1. a) Describe how fire propagation outside of well-sealed MCC cabinets greater than or equal to 440 V is evaluated.
2. b) If well-sealed cabinets less than 440 V are included in the Bin 15 count of ignition sources, provide justification for using this approach, as this is contrary to the guidance.

APLB QUESTION 07 – Fire Probabilistic Risk Assessment Method for Very Early Warning Fire Detection Systems

LAR Enclosure 9, Section 4, states that the Peach Bottom FPRA was developed using consensus methods outlined in NUREG/CR-6850 and interpretations of technical approaches as required by the NRC. Part (e) of TS 5.5.16 states that the approaches and methods used in the RICT program shall be acceptable to the NRC. Methods to assess risk must be those used to support the LAR or other methods approved by the NRC for generic use.

There have been some changes to the FPRA methodology since the development of the Peach Bottom FPRA that was peer reviewed. The integration of the NRC-accepted FPRA method described in NUREG-2180, "Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities (DELORES-VEWFIRE)" (ADAMS Accession No. ML16343A058), may be relevant to this submittal and could potentially impact the TSTF-505 results, CDF, or large early release frequency (LERF).

Section 2.5.5 of RG 1.174 provides guidance that indicates additional analysis is necessary to ensure that contributions from the above influence would not change the conclusions of the LAR.

1. a) If the above guidance has not been implemented in the Peach Bottom FPRA, provide a detailed justification for why the integration of this guidance would not change the conclusions of the LAR and subsequently not impact the TSTF-505 RICT calculations and risk metrics for total CDF and total LERF. As part of this justification, identify any FPRA methodologies used in the FPRA that are no longer accepted by the NRC staff (e.g., guidance provided in FAQ 08-0046, "Closure of National Fire Protection Association 805 Frequently Asked Question 08-0046 Incipient Fire Detection Systems" (ADAMS Accession No. ML093220426), has been retired by letter dated July 1, 2016 (ADAMS Accession No. ML16167A444)). Provide technical justification for its use in TSTF-505 RICT calculations and evaluate the significance of its use on the RICT estimates.

   OR

2. b) Alternatively, if the above guidance has been implemented in the FPRA, provide the following information:
   i. Indicate whether the changes to the FPRA are PRA maintenance or a PRA upgrade as defined in the PRA standard, Section 1-5.4, as qualified by
   RG 1.200, along with justification for this determination.
   ii. Discuss any focused- or full-scope peer reviews performed to evaluate these changes that were determined in item (b)(i) above to constitute a PRA upgrade, including when the peer review was performed and when the peer review report that evaluated the upgrade was approved.

APLB QUESTION 08 – Treatment of Sensitive Electronics

FAQ 13-0004, "Clarifications on Treatment of Sensitive Electronics" (ADAMS Accession No. ML13322A085), provides supplemental guidance for application of the damage criteria provided in Sections 8.5.1.2 and H.2 of NUREG/CR-6850, Volume 2, for solid-state and sensitive electronics.

1. a) Describe the treatment of sensitive electronics for the FPRA and explain whether it is consistent with the guidance in Frequently Asked Question (FAQ) 13-0004, including the caveats about configurations that can invalidate the approach (i.e., sensitive electronics mounted on the surface of cabinets and the presence of louver or vents).
2. b) If the approach cannot be justified to be consistent with FAQ 13-0004, then justify that the treatment of sensitive electronics has no impact on the RICT calculations.

APLB QUESTION 09 – Probabilistic Risk Assessment Treatment of Dependencies Between Units 2 and 3

Many plants have multiple units adjoined and thus have common areas. For these plants, the risk contribution from fires originating in one unit must be addressed for impacts to the other unit, given the physical proximity of the other unit, common areas, and the existence of shared systems. Therefore, address the following if Units 2 and 3 have common areas and shared systems:

1. a) Explain how the risk contribution of fires originating in one unit is addressed for the other unit, given impacts due to the physical proximity of equipment and cables in one unit to equipment and cables in the other unit. Include identification of locations where a fire in one unit can affect components in the other unit, and explain how the risk contributions of such scenarios are allocated for an RICT calculation.
2. b) Explain how the contributions of fires in common areas are addressed, including the risk contribution of fires that can impact components in both units.
3. c) Explain the extent to which systems are shared by both units and whether shared systems are credited in the PRA models (IEPRA and FPRA) for both units. If shared systems are credited in the PRA models for each unit, then explain how the PRAs address the possibility that a shared system is demanded in both units in response to a single internal events initiating event or fire initiator.

APLB QUESTION 10 - Probabilistic Risk Assessment Model Uncertainty Analysis Results

The NRC staff SE to NEI 06-09 specifies that the LAR should identify key assumptions and sources of uncertainty and should assess/disposition each as to its impact on the RMTS application. LAR Enclosure 9, Table E9-3, identifies the key assumptions and sources of uncertainty for the FPRA and provides dispositions for each source of uncertainty for this TSTF-505 application. The NRC staff reviewed the dispositions provided in LAR Table E9-3 to the key assumptions and sources of modeling uncertainty and noted that not all uncertainties that appeared to have the potential to impact the RICT calculations seemed fully resolved.

LAR Enclosure 9, Table E9-3, identifies post-fire HRA as a source of FPRA modeling uncertainty because fire HEPs must be adjusted to consider the additional challenges present given a fire. The LAR states that industry consensus modeling approaches are used and concludes that this source of uncertainty impact “is expected to be small” with apparently no sensitivities being performed. To address this source of uncertainty, the LAR states that appropriate RMAs would be required – for example, pre-job briefs. It is unclear to the NRC staff how the RMAs will adequately address the impact on RICT values. Therefore, address the following items:

1. a) Justify that the uncertainty associated with post-fire HRA modeling does not have a consequential impact on calculated RICTs for components supporting TS LCO conditions in the RICT program.

   OR

2. b) Explain what RMAs will be considered to compensate for this uncertainty.

APLB QUESTION 11 – Fire Modeling

The LAR referred to risk evaluation and the application of fire modeling technology. The NRC staff was unable to fully evaluate the fire modeling performed as part of the FPRA.

Regarding the acceptability of the FPRA approach, methods, and data, describe the fire modeling calculational model or numerical methods (e.g., fire modeling tools and techniques) used in support of the FPRA.

APLB QUESTION 12 – Damage Thresholds

Part 4 of ASME/ANS RA-Sa-2009 indicates that damage thresholds be established to support the FPRA. The PRA standard further indicates that thermal impact(s) must be considered in determining the potential for thermal damage of SSCs, and appropriate temperature and critical heat flux criteria must be used in the analysis. Therefore, provide the following information:

1. a) Describe how the installed cabling in the fire areas was characterized, specifically regarding the critical damage threshold temperatures and critical heat fluxes for thermoset and thermoplastic cables.
2. b) An IEEE-383 (Institute of Electrical and Electronics Engineers Standard 383, "IEEE Standard for Type Test of Class 1 E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations") qualified cable may or may not meet the criteria for a 'thermoset cable." It is also possible that a non-IEEE-383 qualified cable actually meets the criteria for a "thermoset" cable. Provide clarification on the assumptions that were made in terms of damage thresholds of cables.
3. c) For those areas that are assumed to have thermoset damage criteria, confirm that the cables are actually thermoset and that the potential confusion about IEEE-383/thermoset is not applicable.
4. d) Describe how raceways with a mixture of thermoplastic and thermoset cables are treated in terms of damage thresholds.
5. e) In each fire area where they are credited, explain how cable tray covers, fire-resistant coatings, and fire wraps were credited in terms of delaying or preventing damage of cables. In addition, explain how holes in cable tray covers were treated regarding the fire modeling damage criteria.

   f) Explain how the damage thresholds for non-cable components (i.e., pumps, valves, electrical cabinets, etc.) were determined. Identify any non-cable components that were assigned damage thresholds different from those for thermoset and thermoplastic cables, and provide a technical justification for these damage thresholds.

Probabilistic Risk Assessment Licensing Branch C (APLC) Audit Questions

APLC QUESTION 01 – Impacts from Seismic Hazard Frequencies

Section 2.3.1, Item 7 of NEI 06-09, states that the "impact of other external events risk shall be addressed in the RMTS program" and explains that one method to do this is by "performing a reasonable bounding analysis and applying it along with the internal events risk contribution in calculating the configuration risk and the associated RICT." The NRC staff's SE for NEI 06-09 states that "Where PRA models are not available, conservative or bounding analyses may be performed to quantify the risk impact and support the calculation of the RICT."

In Section 3 of Enclosure 4 to the LAR, the licensee stated that the site-specific seismic PRA (SPRA) completed in response to the 10 CFR 50.54(f) request for information associated with the Fukushima Near-Term Task Force (NTTF) activities is not directly used in the RICT program but provides input into the calculation for seismic core damage frequency (SCDF) and seismic large early release frequency (SLERF). The licensee selected the seismic hazard curve that was used in the development of NTTF SPRA model, which is based on the peak ground acceleration (PGA). In the same section of the LAR, the licensee mentioned its seismic hazard and screening report (ADAMS Accession No. ML14090A247), which provided seismic hazard curves at various frequencies at 100 (PGA), 25, 10, 5, 2.5, 1, and 0.5 hertz (Hz). The NRC staff compared the seismic hazard curves between these two documents and found that the PGA hazard curve used in the LAR is different than that in the seismic hazard and screening report.

1. a) Explain the difference between the two PGA hazard curves cited above and justify the selection of the PGA hazard curve for use in the estimation of the SCDF penalty in the LAR.
2. b) Justify that the consideration of seismic hazard curves at frequencies other than the PGA does not significantly change the SCDF penalty proposed in the LAR.

APLC QUESTION 02 – Representativeness of Discretization of Seismic Hazard Curve

The licensee provided the PGA seismic hazard curve data from 0.005 gram (g) to 7.5 g in
Table E4-1 of Enclosure 4 to the LAR. The seismic hazard interval frequencies are represented by discretizing the hazard curve into eight 'bins' as shown in Table E4-2 of Enclosure 4 to the LAR. The representative PGA for the last 'bin' is selected to be 0.99 g for representing the entire hazard from 0.9 g to 7.5 g. This approach results in a mean fragility probability of 0.95 instead of 1.0 as shown in Table E4-3 of Enclosure 4 to the LAR. As explained in Enclosure 4 to the LAR, this change has a minor impact on the estimated SCDF value. However, the NRC staff notes that sensitivity analysis 1d in the licensee's SPRA report (ADAMS Accession
No. ML18240A065) shows a 17 percent increase in SLERF due to refinement in the discretization of the last 'bin.' This is likely to increase the seismic conditional large early release probability (SCLERP) estimate, and therefore, the SLERF penalty estimate. The LAR does not discuss the impact of the refinement of the discretization for the last 'bin' on the estimated SLERF penalty.

Justify that the selected representative PGA of 0.99 g for the last 'bin' is reasonable and conservative for the estimated SLERF penalty or provide an updated SLERF penalty.

APLC QUESTION 03 – Seismic Core Damage Frequency and Large Early Release Frequency Penalty Estimate

Section 2.3.1, Item 7 of NEI 06-09, states that the “impact of other external events risk shall be addressed in the RMTS program” and explains that one method to do this is by “performing a reasonable bounding analysis and applying it along with the internal events risk contribution in calculating the configuration risk and the associated RICT.” The NRC staff’s SE for NEI 06-09 states that “Where PRA models are not available, conservative or bounding analyses may be performed to quantify the risk impact and support the calculation of the RICT.”

The seismic penalty approach is used to quantify the risk impact and to support the RICT evaluation. The staff notes that there is a site-specific seismic PRA that could be used for this analysis. Section 3 of Enclosure 4 to the LAR states that the site-specific SPRA was not directly used in the RICT program but provided input into the calculation for SCDF and SLERF. The licensee compared the estimated SCDF penalty for the proposed RICT calculations against the point-estimate SCDF from the site-specific SPRA. In addition, the licensee used the SLERF to SCDF ratio from the site-specific SPRA to determine the SLERF penalty for use in the proposed RICT calculations.

The comparison of the estimated SCDF and SLERF penalties against the corresponding point-estimate mean values from the site-specific SPRA does not provide justification that the SCDF and SLERF penalty estimates are conservative, as stated in the NEI 06-09 guidance. There is no upper bound on the change-in-risk calculation, and the change in risk can exceed the base SCDF and SLERF. However, it appears to the NRC staff that the SPRA could provide the means to justify that the proposed SCDF and SLERF penalty estimates are conservative, and therefore, consistent with the staff’s SE for NEI 06-09.

Justify that the SCDF and SLERF penalty estimates are conservative based on the results and insights from change-in-risk calculations for the proposed RICTs using the recent site-specific SPRA.

Technical Specifications Branch (STSB) Audit Questions

STSB QUESTION 01 – Technical Specification 3.5.1.E, One ADS [Automatic Depressurization System] Valve Inoperable

LAR Enclosure 1, Table E1-1 lists in the column of “TS 3.5.1.E” a condition with one ADS valve inoperable. The corresponding column of the “SSCs Covered by TS LCO Condition” indicates that ADS (five safety relief valves) are required to be operable, and the column of “Design Success Criteria” indicates that five ADS valves are available.

Clarify for TS 3.5.1.E condition with one of five required ADS valves inoperable, that the Design Success Criteria need 3 or 4 available ADS valves. Discuss the Analyses of Record (AOR) that demonstrated adequacy of 3 or 4 ADS valves for reactor pressure vessel rapid depressurization to mitigate the loss-of-coolant accident consequences and reference the NRC documents approving the AOR of the concern or address the acceptability of the AOR if it was not previously approved by the NRC.

STSB QUESTION 02 – Technical Specification 3.5.1.F, One Automatic Depressurization System valve inoperable and One Low Pressure Emergency Core Cooling System Subsystem Inoperable

LAR Enclosure 1, Table E1-1 lists in the column of “TS 3.5.1.F” a condition with one ADS valve inoperable and one low pressure Emergency Core Cooling System (ECCS) injection/spray subsystem inoperable. Clarify the same for 3.5.1.F. The corresponding column of the “SSCs Covered by TS LCO Condition” states, “See LCO Condition 3.5.1.A and 3.5.1.E,” which indicates that ADS (five safety relief valves) are required to be operable, and the column of “Design Success Criteria” indicates that five ADS valves are available.

Clarify for TS 3.5.1.F Condition with one of 5 required ADS valves inoperable, that the DSC need three or four available ADS valves. Discuss the AOR that demonstrated adequacy of three or four ADS valves for reactor pressure vessel rapid depressurization to mitigate the loss-of-coolant accident consequences and reference the NRC documents approving the AOR of the concern, or address the acceptability of the AOR if it was not previously approved by the NRC.

Electrical Engineering Branch (EEEB) Audit Questions

EEEB QUESTION 01 – Technical Specification 3.8.1.D, Two or More Offsite Alternating Current Power Circuits Inoperable

Peach Bottom’s DSC is derived from the current licensing basis of the plant, as documented in the Updated Safety Analysis Report, and should include a minimum set of required equipment that has the capacity and capability to safely shut down the reactor in case of an accident and maintain it in a safe condition. In Table E1-1 of Enclosure 1 of the LAR, the DSC for TS

LCO 3.8.1.D (two or more offsite AC power circuits inoperable) is “one of two offsite AC power sources.” The NRC staff notes that if both offsite circuits are inoperable, one offsite AC power source as listed in the DSC is not available to provide the necessary power to safely shut down the reactor and maintain it in safe condition. Therefore, it is not clear how one offsite circuit can be the DSC for TS 3.8.1.D during the RICT program entry.

Explain this apparent discrepancy in Table E1-1 of Enclosure 1 of the LAR. Additionally, describe any effect the discrepancy may have on the PRA success criteria for TS LCO 3.8.1.D.

EEEB QUESTION 02 – Technical Specification 3.8.1.B, One Diesel Generator Inoperable

Table E1-1 in Enclosure 1 of the LAR states that the DSC for TS 3.8.1 Condition B is “three of four diesel generators.” Explain the basis for this DSC. Include in the explanation, as necessary to clarify the basis, a description of the onsite AC power system’s design configuration, including each diesel generator’s capacity and loading.

EEEB QUESTION 03 – Technical Specification 3.8.4 Conditions A, B, C, D, and E

Table E1-1 in Enclosure 1 of the LAR states that the DSC for TS 3.8.4 Conditions A, B, C, D, and E is “three of four DC divisions.” Explain the basis for the DSC for these TS conditions. Include in the explanation for each applicable TS condition, as necessary to clarify the basis, a description of each unit’s 125 VDC and 250 VDC system’s configuration, including number and type of batteries and chargers with associated capacities and loading, and use of any cross ties, as applicable.

EEEB QUESTION 04 – Technical Specification 3.8.7 Conditions A, B, C, and D

Table E1-1 in Enclosure 1 of the LAR states that the DSC for TS 3.8.7 Conditions A, B, C, and D is “three of four divisions.” Explain the basis for the DSC for these TS conditions. Include in the explanation for each TS condition, as necessary to clarify the basis, a description of the associated system configuration.

EEEB QUESTION 05 – RMA Examples

As part of its evaluation, the NRC staff reviews the proposed RMA examples for reasonable assurance that the RMAs are considered to monitor and control risk and to ensure adequate defense in depth. Enclosure 12 of the LAR describes the RMAs examples for TS 3.8.1.A,
TS 3.8.1.B, TS 3.8.1.D, and TS 3.8.4.A. However, the LAR does not include the RMA examples for TS 3.8.7 conditions related to the power distribution system. Provide the RMA example(s) for TS 3.8.7.

Instrumentation & Controls Branch B (EICB) Audit Questions

EICB QUESTION 01 – Instrumentation & Controls Redundancy and Diversity

RG 1.174, Revision 3, states the licensee should assess whether the proposed licensing basis change meets the defense-in-depth principle by not overrelying on programmatic activities as compensatory measures associated with the change in the licensing basis. RG 1.174 further elaborates that human actions (e.g., manual system actuation) are considered as one type of compensatory measure.

Therefore, in LAR Attachment 5, if the diverse means identified are the manual actuations, demonstrate by one example that these “manual actuations” identified as the diverse means are modeled in the plant PRA defined in plant operation procedures to which operators are trained, and confirm the completion times associated with these actions are evaluated as adequate.

October 21, 2020 – Letter from Jennifer C. Tobin, Project Manager Plant Licensing Branch I Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to Bryan C. Hanson Senior Vice President Exelon Generation Company, LLC President and Chief Nuclear Officer Exelon Nuclear with subject of PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 − REGULATORY VIRTUAL AUDIT PLAN REGARDING LICENSE AMENDMENT REQUEST TO ADOPT TSTF-505, REVISION 2 (EPID L-2020-LLA-0120)

By letter dated May 29, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20150A007), Exelon Generation Company, LLC submitted a license amendment request to adopt Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b," dated July 2, 2018 (ADAMS Accession No. ML18183A493), for the Peach Bottom Atomic Power Station, Units 2 and 3.

The proposed amendments would revise technical specification requirements to permit the use of risk-informed completion times for actions to be taken when limiting conditions for operation are not met.

The U.S. Nuclear Regulatory Commission staff will be conducting a virtual audit from November 9, 2020, to November 13, 2020 (excluding November 11, 2020, which is a Federal holiday), with Exelon Generation Company, LLC staff and associated contractors. The regulatory virtual audit plan is enclosed with this letter.

If you have any questions regarding this matter, please contact me at 301-415-2328 or by e-mail to Jennifer.Tobin@nrc.gov.

REGULATORY VIRTUAL AUDIT PLAN
LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT TSTF-505, REVISION 2
EXELON GENERATION COMPANY, LLC
PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3
DOCKET NOS. 50-277 AND 50-278

1.0 BACKGROUND

By application dated May 29, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20150A007), Exelon Generation Company, LLC (the licensee) submitted a license amendment request (LAR) for Peach Bottom Atomic Power Station, Units 2 and 3 (Peach Bottom). The amendments would revise technical specification (TS) requirements to permit the use of risk-informed completion times (RICTs) for actions to be taken when limiting conditions for operation are not met. The proposed changes are based on Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF [Risk Informed Technical Specification Task Force] Initiative 4b," dated July 2, 2018 (ADAMS Accession No. ML18183A493). The U.S. Nuclear Regulatory Commission (NRC) issued a final model safety evaluation approving TSTF-505, Revision 2, on November 21, 2018 (ADAMS Package Accession No. ML18269A041).

2.0 REGULATORY AUDIT BASES

A regulatory audit is a planned license or regulation-related activity that includes the examination and evaluation of primarily non-docketed information. The audit is conducted with the intent to gain understanding, to verify information, and to identify information that will require docketing to support the basis of a licensing or regulatory decision. Performing a regulatory audit is expected to assist the NRC staff in efficiently conducting its review of the LAR and to gain insights of the licensee's processes and procedures. Information that the NRC staff relies upon to make the safety determination must be submitted on the docket.

The basis of this audit is the Peach Bottom LAR to revise TS requirements to permit the use of RICTs and NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition” (SRP), Chapter 19, Section 19.2, “Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance” (ADAMS Accession No. ML071700658).

The audit will be performed consistent with NRC Office of Nuclear Reactor Regulation Office Instruction LIC-111, Revision 1, “Regulatory Audits,” dated October 31, 2019 (ADAMS Accession No. ML19226A274). An audit was determined to be the most efficient approach toward a timely resolution of issues associated with this LAR review, since the NRC staff will have an opportunity to minimize the potential for multiple rounds of requests for additional information and ensure no unnecessary burden will be imposed by requiring the licensee to address issues that are no longer necessary to make a safety determination.

3.0 PURPOSE AND SCOPE

The purpose of this audit is to identify information that the licensee should submit on the docket for NRC staff to make a safety determination and to gain a better understanding of the following areas related to the LAR:

- • calculations, analyses, and bases underlying the LAR;
- • approach for developing and implementing the plant’s risk-managed TS program;
- • extent that the LAR is consistent with TSTF-505, Revision 2; Nuclear Energy Institute

  (NEI) Topical Report 06-09, Revision 0-A, “Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines, Industry Guidance Document,” dated November 6, 2006 (ADAMS Package Accession
  No. ML122860402); and the NRC’s Final Safety Evaluation for NEI 06-09, dated May 17, 2007 (ADAMS Accession No. ML071200238);

- • whether the proposed configurations introduce any adverse effects on the ability or capacity of plant equipment to perform its design-basis function(s) when the plant is operated in the proposed TS allowable configuration;
- • technical acceptability of the probabilistic risk assessment (PRA) for use in the application and how plant design features are modeled in the PRA used to support the LAR; and
- • use of the Configuration Risk Management Program tool (i.e., PARAGON) to support RICT program implementation.

  The areas of focus for the regulatory audit are the information contained in the LAR, the audit information needs listed in the following section of this audit plan, and all associated and relevant supporting documentation (e.g., methodology, process information, calculations, etc.). The relevant supporting documents are identified below.

4.0 INFORMATION AND OTHER MATERIAL NECESSARY FOR THE REGULATORY AUDIT

The following documentation should be available to the audit team:

1. the documentation specified in Section 4 of the portal audit plan dated August 4, 2020 (ADAMS Accession No. ML20217L346),
2. PRA notebook regarding component data calculations that address SSC mission times, including the emergency diesel generator split mission times,
3. calculation notebook regarding the tornado missile hazard risk value determinations, and
4. any additional supporting documentation that the licensee may determine is responsive to the NRC staff's above information requests.

5.0 AUDIT TEAM

The members of the audit team are anticipated to be:

- • Jennifer Tobin, Project Manager, NRC/DORL (Jennifer.Tobin@nrc.gov)
- • Todd Hilsmeier, Team Leader, NRC/APLA (Todd.Hilsmeier@nrc.gov)
- • Jeff Circle, NRC/APLA (Jeff.Circle@nrc.gov)
- • Robert Pascarelli, Branch Chief, NRC/APLA (Robert.Pascarelli@nrc.gov)
- • Milton Valentin-Olmeda, NRC/APLC (Milton.Valentin-Olmeda@nrc.gov)
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- • Yuken Wong, NRC/EMIB (Yuken.Wong@nrc.gov)
- • Shie-Jeng Peng, NRC/SNSB (Shie-Jeng.Peng@nrc.gov)
- • Mark Wilk, NRC Contractor, Pacific Northwest National Laboratory

  (mark.wilk@pnnl.gov)

6.0 LOGISTICS

The audit will be conducted remotely from November 9, 2020, to November 13, 2020 (excluding November 11, 2020), between 8:30 a.m. and 4:00 p.m. each day. An entrance briefing will be held at the beginning of the audit, and an exit briefing will be held at the end of the audit. Attachment A of the audit plan provides the proposed agenda for the remote audit. Attachment B contains the audit questions that the NRC staff would like to have prepared dialogue. The NRC project manager will coordinate with the licensee any identified changes to the audit schedule and logistics.

7.0 SPECIAL REQUESTS

The NRC staff would like access to the documents listed in Section 4.0 above through an online portal that allows the NRC staff and contractors to access documents via the internet. The following conditions associated with the online portal must be maintained throughout the duration that the NRC staff and contractors have access to the online portal:

- • The online portal will be password-protected, and separate passwords will be assigned to the NRC staff and contractors who are participating in the audit.
- • The online portal will be sufficiently secure to prevent the NRC staff and contractors from printing, saving, downloading, or collecting any information on the online portal.

8.0

Conditions of use of the online portal will be displayed on the login screen and will require acknowledgement by each user.

Username and password information should be provided directly to the NRC staff and contractors. The NRC project manager will provide Exelon the names and contact information of the NRC staff and contractors who will be participating in the audit. All other communications should be coordinated through the NRC project manager.

DELIVERABLES

An audit summary, which may be public, will be prepared within 90 days of the completion of the audit. If the NRC staff identifies information during the audit that is needed to support its regulatory decision, the staff will issue requests for additional information to the licensee after the audit.

ATTACHMENT A
Proposed Audit Agenda (Revision 0)
Peach Bottom Atomic Power Station, Units 2 and 3, License Amendment Request to Adopt TSTF-505, Revision 2

Day 1 – Monday, November 9, 2020 (8:30 am to 4:00 pm)*

- • Entrance briefing
  - - Opening comments by NRC and Exelon Generation Company, LLC (Exelon)
  - - Introductions and logistics
- • Real-time risk (RTR) model demonstration by Exelon
- • Discuss RTR model and calculation of RICT estimates
  - - RTR model (including benchmarking, updating, and how seasonal variations are accounted) **(APLA Questions 07, 08, and 09)**

- - PRA functional determination and RICT estimates
- - Treatment of common cause failures for planned and emergent conditions
- - Identification of risk-management actions **(EEOB Question 05)**

- • Discuss Key Principle 5, Maintenance Rule and monitoring **(APLA Question 11)**
- • Summary of the day[1]
- • NRC staff internal meeting

Day 2 – Tuesday, November 10, 2020 (8:30 am to 4:00 pm)*

• Summary of previous day and review open items

• Discuss internal events PRA technical acceptability

- - I&C diversity and modeling in PRA (**EICB Question 01; APLA Question 06)**
- - EDGs, RCIC, HPSW, and vacuum breakers **(APLA Questions 01 to 04)**
- - Credit for FLEX equipment and actions **(APLA Question 05)**

[1] If discussion topics are completed early, additional discussions for Day 1 may include seismic hazard from Day 4 and/or design-success criteria from Day 4.

• Discuss key assumptions and uncertainties - process **(APLA Question 10)**

- - Summary of the day[2]
- - NRC staff internal meeting

Wednesday, November 11, 2020 – Veterans Day observed (no audit)

Day 3 – Thursday, November 12, 2020 (8:30 am to 4:00 pm)*

- • Summary of previous day and review open items
- • Discuss fire PRA technical acceptability **(APLB Questions 01 to 12)**
  - - Summary of the day
  - - NRC staff internal meeting

Day 4 – Friday, November 13, 2020 (8:30 am to 4:00 pm)*

- • Summary of previous day and review open items
- • Discuss seismic hazard **(APLC Questions 01 to 03)**
- • Discuss design-success criteria **(STSB Questions 01 and 02; EEEB Questions 01 to 04)**
- • Follow-up on any remaining open items

- • Summary of audit and exit meeting (tentatively scheduled for 3:30 pm)

* Lunch will be tentatively scheduled from 12:00 pm – 1:00 pm Acronyms:

28705. APLA NRC/NRR/PRA Licensing Branch A
28706. APLB NRC/NRR/PRA Licensing Branch B
28707. APLC NRC/NRR/PRA Licensing Branch C

EDG Emergency Diesel Generator

EEEB NRC/NRR/Electrical Engineering Branch EICB NRC/NRR/Instrumentation & Controls Branch FLEX Flexible Mitigation Strategies
HPSW High-Pressure Service Water
I&C Instrumentation and Control
NRC U.S. Nuclear Regulatory Commission

[2] If discussion topics are completed early, additional discussions for Day 2 may include seismic hazard from Day 4, design-success criteria from Day 4, and/or fire PRA from Day 3, if not discussed earlier.

NRR Office of Nuclear Reactor Regulation PRA Probabilistic Risk Assessment
RCIC Reactor Core Isolation Cooling System RICT Risk-informed Completion Time

RTR Real-time Risk
STSB NRC/NRR/Technical Specifications Branch

ATTACHMENT B

AUDIT QUESTIONS
LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT TSTF-505, REVISION 2
EXELON GENERATION COMPANY, LLC
PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3
DOCKET NOS. 50-277 AND 50-278

By application dated May 29, 2020, Exelon Generation Company, LLC (the licensee) submitted a license amendment request (LAR) for Peach Bottom Atomic Power Station, Units 2 and 3 (Peach Bottom) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20150A007). The amendment would revise technical specification (TS) requirements to permit the use of risk-informed completion times (RICTs) for actions to be taken when limiting conditions for operation (LCOs) are not met. The proposed changes are based on Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, “Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b,” dated July 2, 2018 (ADAMS Accession No. ML18183A493). The U.S. Nuclear Regulatory Commission (NRC) issued a final model safety evaluation (SE) approving TSTF 505, Revision 2, on November 21, 2018 (ADAMS Accession No. ML18269A041). The NRC staff has determined that the following information is needed in order to complete its review.

Probabilistic Risk Assessment Licensing Branch A (APLA) Audit Questions

Regulatory Guide (RG) 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (ADAMS Accession No. ML17317A256), states that the scope, level of detail, and technical adequacy of the probabilistic risk assessment (PRA) are to be commensurate with the application for which it is intended and the role the PRA results play in the integrated decision process. The NRC's SE for Nuclear Energy Institute (NEI) Topical Report NEI 06-09, Revision 0-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines, Industry Guidance Document," dated November 6, 2006 (ADAMS Package Accession No. ML122860402) (hereafter NEI 06-09), and the NRC's Final Safety Evaluation for NEI 06-09, dated May 17, 2007 (ADAMS Accession No. ML071200238), state that the PRA models should conform to the guidance in RG 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." The current version is RG 1.200, Revision 2 (ADAMS Accession No. ML090410014), which clarifies the current applicable American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA standard is ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications." In RG 1.200, the quality of the PRA must be compatible with the safety implications of the proposed TS change and the role the PRA plays in justifying the change. RG 1.200 describes a peer review process using ASME/ANS RA-Sa-2009 as one acceptable approach

for determining the technical acceptability of the PRA. The primary results of a peer review are the facts and observations (F&Os) recorded by the peer review team and the subsequent resolution of these F&Os. A process to close finding-level F&Os is documented in
Appendix X to the NEI guidance documents NEI 05-04, NEI 07-12, and NEI 12-13, titled "NEI 05-04/07-12/12-[13] Appendix X: Close-out of Facts and Observations (F&Os)" (ADAMS Package Accession No. ML17086A431), which was accepted by the NRC in a letter dated May 3, 2017 (ADAMS Accession No. ML17079A427). NEI 06-09 states that the PRA shall meet Capability Category (CC)-II for the supporting requirements of the PRA standard, and any deviations from these capability categories relative to the RMTS program shall be justified.

APLA QUESTION 01 – Probabilistic Risk Assessment Modeling of Emergency Diesel Generators

As part of its audit (ADAMS Accession No. ML20217L346), the NRC staff noted that the analysis in Section 4.5.2 of PRA Notebook PB-PRA-013 documented the impact of using a "split" mission time of 4 and 8.2 hours for the emergency diesel generators (EDGs). The results of a sensitivity study in Section 4.5.2, which used a PRA model mission time of 24 hours for the EDGs, demonstrated a 3 percent increase in overall core damage frequency (CDF). It is unclear to the NRC staff how the EDGs would only be required for a specific portion of the PRA analysis window of 24 hours. The NRC staff notes that this source of uncertainty does not appear to have been addressed in PRA Notebook PB-MISC-043, which addresses the impact of PRA assumptions on RICT calculations, especially conditions related to alternating current (AC) and direct current (DC) power (e.g., TSs 3.8.1 and 3.8.4). In light of these observations, provide the following information:

1. a) Provide justification for the use of split mission times for the EDGs in the Peach Bottom PRA models. Include in this discussion the reasoning for not using the standard 24-hour mission time used in PRA models.
2. b) Provide the results of RICT sensitivity studies for AC and DC power-related TS LCOs submitted in the LAR that demonstrate the impact of not implementing the 24-hour PRA mission time. Include a discussion of the impact of the split mission times for the EDGs on the RICT calculations.

APLA QUESTION 02 – Probabilistic Risk Assessment Modeling of RCIC Black Start

As part of its audit (ADAMS Accession No. ML20217L346), the NRC staff noted that Table 2-1 of PRA Notebook PB-MISC-043 states, "Systems that normally require DC [power] for operation are not credited for continued operation upon battery depletion"; however, the reactor core isolation cooling system (RCIC) is credited after battery depletion, referred to as "RCIC black start." The analysis in Table 2-1 states that the initial operation of RCIC or high-pressure coolant injection (HPCI) for 2 hours will provide sufficient reactor pressure vessel level to perform the RCIC black start prior to core damage. The analysis assessment states the RCIC black start credit represents "a slight conservative bias." It is unclear to the NRC staff whether this action is feasible, since the operators have no indication of vessel level or injection flow, and this is a conservative assumption. Provide the following information:

a) Identify which RICT TS LCOs are affected by the credit for RCIC black start.

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2. b) Provide the basis for the feasibility of crediting RCIC after battery depletion (i.e., RCIC black start). Include in this discussion what licensee program directs this action (e.g., emergency operating procedures, severe accident management guidelines, mitigating FLEX strategies).
3. c) Provide the results of RICT sensitivity studies of the associated TS LCOs identified in part (a) that remove credit for RCIC black start. Include a discussion of the impact of this assumption on the RICT calculations.

APLA QUESTION 03 – Probabilistic Risk Assessment Modeling of High-Pressure Service Water

Table E1-1 of LAR Enclosure 1 regarding TS LCO 3.7.1.A (one high-pressure service water (HPSW), subsystem inoperable) states in Note 4 of the table that the HPSW consists of two independent subsystems. Each subsystem contains two HPSW pumps that discharge to both residual heat removal (RHR) heat exchangers. The design-success criteria (DSC) for this TS LCO in Table E1-1 is one of two subsystems; however, the PRA success criteria is one pump and one heat exchanger. It is unclear to the NRC staff whether the PRA success criteria is equivalent to a single subsystem as described in Note 4.

Provide a description of the HPSW system modeling in the Peach Bottom PRAs, and describe the analysis performed to support the PRA success criteria for HPSW.

APLA QUESTION 04 – Probabilistic Risk Assessment Modeling of Vacuum Breakers (Implementation Items)

LAR Attachment 6 lists the following implementation items that must be completed prior to implementation of the RICT program to satisfy the guidance in NEI 06-09 that the PRA reflect the as-built, as-operated plant and that the PRA technical adequacy is acceptable:

- • Exelon will ensure that the reactor building-to-suppression chamber vacuum breakers are modeled in the Peach Bottom PRA with sufficient detail to accurately calculate the RICT.
- • Exelon will ensure that the suppression chamber-to-drywell vacuum breakers are modeled in the Peach Bottom PRA with sufficient detail to accurately calculate the RICT.

  LAR Attachment 6 also states that if implementation of any of these changes constitutes a PRA upgrade as defined in the PRA standard, as endorsed by RG 1.200, then a focused-scope peer review will be performed on these changes, and any findings will be resolved and incorporated in the PRA prior to the implementation of the RICT program. However, it is unclear to the NRC staff how the addition of these system models will meet CC-II of the PRA standard, as endorsed by RG 1.200. In light of these observations, provide the following information:

  Regarding the implementation items identified above, describe how the associated systems will be adequately modeled in the PRA to CC-II. Include in this discussion:

i. How mechanical components, instrument channels, logic components, and other relevant system components will be modeled.

ii. Provide details of the success criteria for these systems. If the PRA success criteria do not match the DSC, then provide a justification for the PRA success criteria.

iii. ConfirmwhethertheseimplementationitemsapplytoboththeinternaleventsPRA (IEPRA) and the fire PRA (FPRA). Accordingly, adjust the wording for each of the affected implementation items in LAR Attachment 6. If any of these implementation items will not be applied to the FPRA, then justify the position that the FPRA model will be sufficient to support the RICT program.

APLA QUESTION 05 – Probabilistic Risk Assessment Modeling and Uncertainty of FLEX Equipment and Actions

The NRC memorandum dated May 30, 2017, "Assessment of the Nuclear Energy Institute 16-06, 'Crediting Mitigating Strategies in Risk-Informed Decision Making,' Guidance for Risk-Informed Changes to Plants Licensing Basis" (ADAMS Accession No. ML17031A269), provides the NRC's staff assessment of challenges to incorporating FLEX equipment and strategies into a PRA model in support of risk-informed decisionmaking in accordance with the guidance of RG 1.200.

Regarding equipment failure probability in the May 30, 2017, memorandum, the NRC staff concludes (Conclusion 8):

The uncertainty associated with failure rates of portable equipment should be considered in the PRA models consistent with the ASME/ANS PRA Standard as endorsed by RG 1.200. Risk-informed applications should address whether and how these uncertainties are evaluated.

Regarding human reliability analysis (HRA), NEI 16-06, Section 7.5, recognizes that the current HRA methods do not translate directly to human actions required for implementing mitigating strategies. Sections 7.5.4 and 7.5.5 of NEI 16-06 describe such actions to which the current HRA methods cannot be directly applied, such as debris removal, transportation of portable equipment, installation of equipment at a staging location, routing of cables and hoses, and those complex actions that require many steps over an extended period, multiple personnel and locations, evolving command and control, and extended time delays. In the May 30, 2017, memorandum, the NRC staff concludes (Conclusion 11):

Until gaps in the human reliability analysis methodologies are addressed by improved industry guidance, [human error probabilities] HEPs associated with actions for which the existing approaches are not explicitly applicable, such as actions described in Sections 7.5.4 and 7.5.5 of NEI 16-06, along with assumptions and assessments, should be submitted to NRC for review.

Regarding uncertainty, Section 2.3.4 of NEI 06-09 states that PRA modeling uncertainties shall be considered in the application of the PRA base model results to the RICT program and that sensitivity studies should be performed on the base model prior to initial implementation of the RICT program on uncertainties that could potentially impact the results of an RICT calculation. NEI 06-09 also states that the insights from the sensitivity studies should be used to develop appropriate risk management actions (RMAs), including highlighting risk-significant operator actions, confirming availability and operability of important standby equipment, and assessing the presence of severe or unusual environmental conditions. Uncertainty exists in PRA modeling of FLEX related to the equipment failure probabilities for FLEX equipment used in

the model, the corresponding operator actions, and pre-initiator failure probabilities. Therefore, FLEX modeling assumptions can be key assumptions and sources of uncertainty for the RICTs proposed in this application.

LAR Enclosure 9, Table E9-1, indicates that FLEX equipment and actions have been credited in the IEPRA. The LAR states that a sensitivity study was performed for the IEPRA to address this issue. The LAR stated that the sensitivity did not significantly impact the RICT values. As part of its audit (ADAMS Accession No. ML20217L346), the NRC staff noted that Section 8 of PRA Notebook PB-MISC-043 provided results of a sensitivity study where the failure probability of the FLEX injection pump and diesel generator was significantly increased. However, the NRC staff notes the significant challenges of modeling FLEX equipment and actions without sufficient industry data and without a consensus HRA approach to address unique aspects of FLEX actions.

The NRC staff also notes that the difference between failure rates associated with permanently installed safety-related diesel generators and portable non-safety-related diesel generators could be greater than a factor of 10 without consideration of further uncertainty. It is unclear to the NRC staff whether the stated sensitivity study addressed the uncertainties associated with estimating HEP values for FLEX actions, especially for non-operator trained actions. Given the observations above, it is not clear whether the sensitivity study performed to assess the impact of crediting FLEX equipment and actions is sufficient to conclude that the impact to the RICT program of the uncertainties associated with modeling FLEX is negligible. For this reason, and to understand the credit that will be taken for FLEX equipment and actions in the RICT program, address the following separately for the IEPRA, internal flooding PRA, and FPRA:

1. a) Provide results of LCO-specific sensitivity studies that assess the removal of FLEX credit on RICT calculations.
2. b) Regarding HRA, address the following items:
   i. Discuss whether any credited operator actions related to FLEX equipment contain actions described in Sections 7.5.4 and Sections 7.5.5 of NEI 16-06.

      If any credited operator actions related to FLEX equipment contain actions described in Sections 7.5.4 and Sections 7.5.5 of NEI 16-06, answer either item (ii) or (iii) below.

   ii. Justify and provide results of LCO-specific sensitivity studies that assess impact from the FLEX-independent and FLEX-dependent HEPs associated with deploying and staging FLEX portable equipment on the RICTs proposed in this application. As part of the response, include the following information:
      1. Justify independent and joint HEP values selected for the sensitivity studies, including justification of why the chosen values constitute bounding realistic estimates.
      2. Provide numerical results on specific selected RICTs and discussion of the results.

3. Discuss composite sensitivity studies of the RICT results to the operator action HEPs and the FLEX equipment reliability uncertainty sensitivity study.
4. Describe how the source of uncertainty due to the uncertainty in FLEX operator action HEPs will be addressed in the RICT program. Describe specific RMAs

being proposed and how these RMAs are expected to reduce the risk associated with this source of uncertainty.

iii. Alternatively to item (b)ii above, provide information associated with the following items listed in supporting requirements (SR) HR-G3 and HR-G7 of the PRA standard to support the NRC staff's detailed review of the LAR:

1. the level and frequency of training that the operators and non-operators receive for deployment of the FLEX equipment (performance shaping factor (a) in SR HR-G3),
2. performance shaping factor (f) in SR HR-G3 regarding estimates of time available and time required to execute the response,
3. performance shaping factor (g) in SR HR-G3 regarding complexity of detection, diagnosis, and decisionmaking and executing the required response,
4. performance shaping factor (h) in SR HR-G3 regarding consideration of environmental conditions, and

5. human action dependencies as listed in SR HR-G7 of the PRA standard.

c) The PRA
new methodology or significant changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences. Section 1-5 of Part 1 of the PRA standard states that upgrades of a PRA shall receive a peer review in accordance with the requirements specified in the peer review section of each respective part of this standard.

i. Provide an evaluation of the model changes associated with incorporating FLEX mitigating strategies that demonstrates that none of the following criteria are satisfied: (1) use of new methodology, (2) change in scope that impacts the significant accident sequences or the significant accident progression sequences, and (3) change in capability that impacts the significant accident sequences or the significant accident progression sequences.
ii. Alternatively to item (c)i above, confirm that the modeling of FLEX equipment and FLEX actions in the PRA has been peer reviewed in accordance with NRC-accepted methods. Provide the findings of the peer review performed on the FLEX modeling and the disposition of the findings as they pertain to the impact on this LAR.

standard defines PRA upgrade as the incorporation into a PRA model of a

APLA QUESTION 06 – Probabilistic Risk Assessment Modeling and Uncertainty of Digital Instrumentation and Controls

Section 2.3.4 of NEI 06-09 states that PRA modeling uncertainties be considered in application of the PRA base model results to the RICT program. The NRC SE for NEI 06-09 states that this consideration is consistent with Section 2.3.5 of RG 1.177, Revision 1, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications" (ADAMS Accession No. ML100910008). NEI 06-09 further states that sensitivity studies should be performed on the base model prior to initial implementation of the RICT program on uncertainties that could potentially impact the results of an RICT calculation and that sensitivity studies should be used to develop appropriate compensatory RMAs.

1. a) A TS LCO condition listed in LAR Table E1-1 indicates that instrumentation and control (I&C) modeling in the PRA is insufficient to model the condition, and therefore, the inoperability of the associated equipment (e.g., channel) will be modeled using a surrogate event. Furthermore, based on documentation in the LAR for other TS LCO conditions in the RICT program, it is not clear to NRC staff whether I&C is modeled in sufficient detail to support implementation of TSTF-505, Revision 2.

   Describe how I&C equipment that is applicable or that impacts the RICT calculations is modeled/considered in the PRA. Include in this discussion: (1) the scope of the I&C equipment that is explicitly modeled (e.g., bistables, relays, sensors, integrated circuit cards), (2) description of the level of detail that the PRA model supports (e.g., are all channels of an actuation circuit modeled), (3) discussion of the generic data and plant-specific data used, and (4) discussion of the associated TS functions for which an RICT can be applied.

2. b) Regarding digital I&C, the NRC staff notes the lack of consensus industry guidance for modeling these systems in plant PRAs to be used to support risk-informed regulatory applications. In addition, known modeling challenges exist such as lack of industry data for digital I&C components, differences between digital and analog system failure modes, and the complexities associated with modeling software failures, including common cause software failures. Also, although reliability data from vendor tests may be available, this source of data is not a substitute for in-the-field operational data. Given these challenges, the uncertainty associated with modeling a digital I&C system could impact the RICT program.

   Attachment 4 of the LAR identifies digital feedwater control system is employed at the plant. However, the modeling of this digital system is not identified in Enclosure 9 as a source of uncertainty. Therefore, it is not clear to the NRC staff whether the digital feedwater control system is the only digital system credited in the PRA and whether there are other digital systems credited in the PRA that could potentially impact RICT calculations. In light of these observations, provide the following information:

   i. Describe and provide the results of a sensitivity study performed for each digital system modeled in the PRA demonstrating that the uncertainty associated with PRA modeling the digital I&C systems has inconsequential impact on the RICT calculations.
   ii. As an alternative to item (b)i above, identify which LCOs are determined to be impacted by digital I&C system modeling for which RMAs will be applied during an

RICT. Explain and justify the criteria used to determine what level of impact to the RICT calculation requires additional RMAs.

APLA QUESTION 07 – PRA Update Process

Section 2.3.4 of NEI 06-09 specifies, "criteria shall exist in PRA configuration risk management to require PRA model updates concurrent with implementation of facility changes that significantly impact RICT calculations."

LAR Enclosure 7 states that if a plant change or a discovered condition is identified and can have significant impact on the RICT calculations, then an unscheduled update of the PRA models will be implemented. More specifically, the LAR states that if the plant changes meet specific criteria defined in the plant PRA and update procedures, including criteria associated with consideration of the cumulative risk impact, then the change will be incorporated into applicable PRA models without waiting for the next periodic PRA update. The LAR does not explain under what conditions an unscheduled update of the PRA model will be performed or the criteria defined in the plant procedures that will be used to initiate the update.

In light of these observations, describe the conditions under which an unscheduled PRA update (i.e., more than once every two refueling cycles) would be performed and the criteria that would be used to require a PRA update. In the response, define what is meant by "significant impact to the RICT Program calculations."

APLA QUESTION 08 – Real-Time Risk Model

Regulatory Position 2.3.3 of RG 1.174 states that the level of detail in the PRA should be sufficient to model the impact of the proposed licensing basis change. The characterization of the problem should include establishing a cause-effect relationship to identify portions of the PRA affected by the issue being evaluated. Full-scale applications of the PRA should reflect this cause-effect relationship in a quantification of the impact of the proposed licensing basis change on the PRA elements.

Section 4.2 of NEI 06-09 describes attributes of the tool used for configuration risk management (CRM). Some of these attributes are listed below.

- • Initiating events accurately model external conditions and effects of out-of-service equipment.
- • Model translation from the PRA to a separate CRM tool is appropriate; CRM fault trees are traceable to the PRA. Appropriate benchmarking of the CRM tool against the PRA model shall be performed to demonstrate consistency.
- • Each CRM application tool is verified to adequately reflect the as-built, as-operated plant, including risk contributors that vary by time of year or time in fuel cycle or otherwise demonstrate to be conservative or bounding.
- • Application-specific risk important uncertainties contained in the CRM model (that are identified via PRA model to CRM tool benchmarking) are identified and evaluated prior to use of the CRM tool for RMTS applications.

- • CRM application tools and software are accepted and maintained by an appropriate quality program.
- • The CRM tool shall be maintained and updated in accordance with approved station procedures to ensure it accurately reflects the as-built, as-operated plant.

Enclosure 8 of the LAR describes the attributes of the real-time-risk (RTR) model (i.e., Peach Bottom's CRM tool) for use in RICT calculations. The LAR explains that the internal events, internal flooding events, and fire events PRA models are maintained as separate models. The LAR also describes several changes made to the PRA models to support calculation of configuration-specific risk and mentions approaches for ensuring the fidelity of the RTR to the PRAs, including RTR maintenance, documentation of changes, and testing. Regarding development and application of the RTR model, provide the following information:

1. a) Describe the process that will be used to maintain the accuracy of any presolved cutsets with changes in plant configuration.
2. b) Describe the benchmarking activities performed to confirm consistency of the RTR model results to the results of each PRA model of record, including periodicity of RTR updates compared to the model of record updates. Address each model of record (i.e., internal events, internal flooding events, and internal fire events) in the response.

APLA QUESTION 09 – Impact of Seasonal Variations on the Real-Time Risk Model

Regulatory Position 2.3.3 of RG 1.174 states that the level of detail in the PRA should be sufficient to model the impact of the proposed licensing basis change. The characterization of the problem should include establishing a cause-effect relationship to identify portions of the PRA affected by the issue being evaluated. Full-scale applications of the PRA should reflect this cause-effect relationship in a quantification of the impact of the proposed licensing basis change on the PRA elements. Additionally, NEI 06-09 states the following:

If the PRA model is constructed using data points or basic events that change as a result of time of year or time of cycle (examples include moderator temperature coefficient, summer versus winter alignments for HVAC, seasonal alignments for service water), then the RICT calculation shall either 1) use the more conservative assumption at all time, or 2) be adjusted appropriately to reflect the current (e.g., seasonal or time of cycle) configuration for the feature as modeled in the PRA.

Section 2 of LAR Enclosure 8 states, “The impact of outside temperatures on system requirements are addressed in the RTR model.” As part of its audit (ADAMS Accession No. ML20217L346), the NRC staff noted that PRA Notebook PB-MISC-043 states that two EDG fans are required when ambient temperature is above 80 degrees Fahrenheit (°F); however, the PRA model uses a split fraction to represent the percentage of the year assumed to be over
80 °F in modifying the success criteria. The analysis states that RICT will necessitate identifying specific time periods when two fans are required.

Provide further explanation supporting the statement above by summarizing the plant equipment subject to seasonal variations and how it is modeled in the PRA to remove the seasonal dependency.

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APLA QUESTION 10 – Probabilistic Risk Assessment Model Uncertainty Analysis Process

The NRC staff SE to NEI 06-09 specifies that the LAR should identify key assumptions and sources of uncertainty and assess and disposition each as to its impact on the RMTS application. Section 5.3 of NUREG-1855, Revision 1, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking, Main Report,” dated March 2017 (ADAMS Accession No. ML17062A466), presents guidance on the process of identifying, characterizing, and qualitatively screening model uncertainties.

LAR Enclosure 9 states that the process for identifying key assumptions and sources of uncertainty for the IEPRA (includes internal floods) and FPRA was performed using the guidance in NUREG-1855, Revision 1. The LAR indicates that in addition to reviewing generic industry sources of uncertainty for applicability, the IEPRA and FPRA models and notebooks were reviewed for plant-specific assumptions and sources of uncertainty.

However, for the IEPRA (includes internal floods), it is not clear to the NRC staff what specific process and criteria were used to screen uncertainties from an initial comprehensive list of assumptions and sources of PRA modeling uncertainty (including those associated with plant-specific features, modeling choices, and generic industry concerns) in order to conclude that no uncertainty issues could impact the RICT calculations. The NRC staff notes from review of Enclosure 9 of the LAR that the dispositions to many identified sources of uncertainty highlight the phrase “not significantly impact the RICT values.” It is not clear to the NRC staff what this phrase means in all cases. Also, for certain sources of uncertainty, the disposition states that a sensitivity study was performed to evaluate the impact of the uncertainty, but it is not clear what criteria was used to determine when a sensitivity study was performed or when additional RMAs should be considered.

Therefore, address the following regarding the IEPRA (includes internal floods) uncertainties:

1. a) Describe the process used to screen uncertainties from the initial comprehensive lists of PRA uncertainties (including those associated with plant-specific features, modeling choices, and generic industry concerns) in order to eventually conclude that the uncertainty issues could not impact the RICT calculations. Include a description of the criteria that was used to screen down from a comprehensive listing of sources of uncertainty to a smaller set of key candidate assumptions and sources of uncertainty. Also, describe the criteria used to justify that none of the key candidate assumptions and sources of uncertainty could have an impact on the RICT calculations. As part of this description, explain the criteria used to determine when the results of sensitivity studies do not significantly impact RICT values.

2. b) Concerning the evaluation criteria used to evaluate and screen uncertainties addressed in item (a) above:
    i. Discuss the criteria used to consistently determine when a sensitivity study was used to address the identified source of uncertainty.
    ii. Discuss the criteria used to consistently determine when additional RMAs should be implemented because of modeling uncertainty.

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APLA QUESTION 11 – Performance Monitoring and Feedback

Section 2.3 of LAR Attachment 1 states that the application of an RICT will be evaluated using the guidance provided in NEI 06-09, which was approved by the NRC on May 17, 2007 (ADAMS Accession No. ML071200238). The NRC SE for NEI 06-09, states, "The impact of the proposed change should be monitored using performance measurement strategies." NEI 06-09 considers the use of NUMARC 93-01, Revision 4F, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants (ADAMS Accession No. ML18120A069), as endorsed by RG 1.160, Revision 4 (ADAMS Accession No. ML18220B281), for the implementation of the Maintenance Rule. NUMARC 93-01, Section 9.0, contains guidance for the establishment of performance criteria.

Furthermore, Section 2.3 of LAR Attachment 1 states:

In addition, the NEI 06-09-A, Revision 0 methodology satisfies the five key safety principles specified in Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decision-making: Technical Specifications," dated August 1998 (ADAMS Accession No. ML003740176), relative to the risk impact due to the application of a RICT.

NRC staff position C.3.2 provided in RG 1.177 for meeting the fifth key safety principle acknowledges the use of performance criteria to assess degradation of operational safety over a period of time. It is unclear to the NRC staff how the licensee's process for the risk-informed application captures performance monitoring for the structures, systems, and components (SSCs) within-scope of the application. In light of these observations, address either (a) or (b) below.

1. a) Confirm that the Peach Bottom Maintenance Rule program incorporates the use of performance criteria to evaluate SSC performance as described in the NRC-endorsed guidance in NUMARC 93-01.

    OR

2. b) Describe the approach/method used by Peach Bottom for SSC performance monitoring as described in Regulatory Position C.3.2 referenced in RG 1.177 for meeting the fifth key safety principle. In the description, include criteria (e.g., qualitative or quantitative), along with the appropriate risk metrics, and explain how the approach and criteria demonstrate the intent to monitor the potential degradation of SSCs in accordance with the NRC SE for NEI 06-09.

Probabilistic Risk Assessment Licensing Branch B (APLB) Audit Questions

RG 1.200 states that “NRC reviewers... [will] focus their review on key assumptions and areas identified by peer reviewers as being of concern and relevant to the application.” Relatively extensive and detailed reviews of FPRAs were undertaken in support of each LAR to transition to National Fire Protection Association Standard 805 (NFPA 805, “Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants”). These reviews determined that implementation of some of the complex FPRA methods often used nonconservative and oversimplified assumptions to apply the method to specific plant configurations. Some of these issues were not always identified in F&Os by the peer review teams, but are considered potential key assumptions by the NRC staff. Using more defensible

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and less simplified assumptions could substantively affect the fire risk and fire risk profile of the plant.

The NRC staff evaluates the acceptability of the PRA for each new risk-informed application and, as discussed in RG 1.174, recognizes that the technical acceptability of risk analyses necessary to support regulatory decisionmaking may vary with the relative weight given to the risk assessment element of the decision-making process. The NRC staff notes that the calculated results of the PRA are used to calculate an RICT, which subsequently determines how long SSCs (both individual SSCs and multiple, unrelated SSCs) controlled by TSs can remain inoperable. Therefore, the PRA results are given a very high weight in a TSTF-505 application, and the NRC staff requests additional information on the following issues that have been previously identified as potentially key FPRA assumptions.

APLB QUESTION 01 – Reduced Transient Heat Release Rates

The key factors used to justify using transient fire-reduced heat release rates (HRRs) below those prescribed in NUREG/CR-6850, “EPRI/NRC Fire PRA Methodology for Nuclear Power Facilities” (ADAMS Accession No. ML052580075), are discussed in a letter from the NRC to NEI, dated June 21, 2012 (ADAMS Package Accession No. ML12172A406).

If any reduced transient HRRs below the bounding 98th percentile HRR of 317 kilowatts (kW) from NUREG/CR-6850 were used, discuss the key factors used to justify the reduced HRRs. In this discussion, also provide the following information:

1. a) Identification of the fire areas where a reduced transient fire HRR is credited and what reduced HRR value was applied.
2. b) A description for each location where a reduced HRR is credited, and a description of the administrative controls that justify the reduced HRR, including how location-specific attributes and considerations are addressed. Include a discussion of the required controls for ignition sources in these locations and the types and quantities of combustible materials needed to perform maintenance.

Also, include discussion of the personnel traffic that would be expected through each location.

3. c) The results of a review of records related to compliance with the transient combustible and hot work controls.

APLB QUESTION 02 – Joint Human Error Probability Floor

NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines" (ADAMS Accession No. ML12216A104), discusses the need to consider a minimum value for the joint probability of multiple human failure events (HFEs) in HRAs. NUREG-1921 refers to Table 2-1 of NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA)" (ADAMS Accession No. ML051160213), which recommends that joint human error probability (HEP) values should not be below 1E-5. Table 4-4 of Electric Power Research Institute
(EPRI) 1021081, "Establishing Minimum Acceptable Values for Probabilities of Human Failure Events," provides a lower limiting value of 1E-6 for sequences with a very low level of dependence. Therefore, the guidance in NUREG-1921 allows for assigning joint HEPs that are less than 1E-5, but only through assigning proper levels of dependency. TSTF-505 evaluations use the FPRA and IEPRA. The LAR does not provide information about whether and, if so, what minimum joint HEP value(s) is currently assumed in the FPRA. Also,

even if the assumed minimum joint HEP value(s) is shown to have no impact on the current FPRA risk estimates, it is not clear to the NRC staff how it will be ensured that the impact remains minimal for future PRA revisions. In light of these observations, provide the following information:

1. a) Explain what minimum joint HEP value(s) was assumed in the FPRA.
2. b) If a minimum joint HEP value less than 1E-05 was used in the FPRA, then provide a description of the sensitivity study that was performed and the quantitative results that justify that the minimum joint HEP value(s) has no impact on the RICT application.
3. c) If, in response to item (b) above, it cannot be justified that the minimum joint HEP value(s) has no impact on the application, confirm that each joint HEP value used in the FPRA below 1E-5 includes its own separate justification that demonstrates the inapplicability of the NUREG-1792 lower value guideline (i.e., using such criteria as the dependency factors identified in NUREG-1921 to assess level of dependence). Provide an estimate of the number of these joint HEP values below the guideline value of 1E-5 for the FPRA, discuss the range of values, and provide at least two different examples where this justification is applied.

APLB QUESTION 03 – Obstructed Plume Model

NUREG-2178, Volume 1, "Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire (RACHELLE-FIRE)" (ADAMS Accession No. ML16110A140), contains refined peak HRRs, compared to those presented in NUREG/CR-6850, and guidance on modeling the effect of plume obstruction.

Additionally, NUREG-2178 provides guidance that indicates that the obstructed plume model is not applicable to cabinets in which the fire is assumed to be located at elevations of less than one-half of the cabinet height.

1. a) If obstructed plume modeling was used, then indicate whether the base of the fire was assumed to be located at an elevation of less than one-half of the cabinet height.
2. b) Justify any modeling in which the base of an obstructed plume is located at less than one-half of the cabinet's height.

APLB QUESTION 04 – Systems Not Credited in the Fire PRA

As part of its audit (ADAMS Accession No. ML20217L346), the NRC staff reviewed PRA Notebook PB-PRA-021.62, which noted that several systems were identified as not being modeled in the FPRA. The NRC staff notes that some conservative PRA modeling assumptions could have a nonconservative impact on the RICT calculations. If an SSC is part of a system not credited in the FPRA or is supported by a system that is assumed to always fail, then the risk increases due to taking that SSC out of service are masked. Therefore, provide the following information:

a) Identify the systems or components that are assumed to be always failed in the FPRA or not included in the FPRA (e.g., due to lack of cable tracing or other reasons). Justify that these assumptions have an inconsequential impact on the RICT calculations and no RMAs are required to address these items.

b) As an alternative to item (a) above, propose a mechanism to ensure that a sensitivity study is performed for the RICT calculations for applicable SSCs that accounts for the impact on the RICT of the 1) conservative FPRA assumption of failed SSCs or 2) SSCs not included in the FPRA model. The proposed mechanism should also ensure that any additional risk from correcting these assumptions is either accounted for in the RICT calculations or is compensated for by applying additional RMAs during the RICT.

APLB QUESTION 05 - Implementation Item for Cable Data for Standby Liquid Control

LAR Attachment 6 lists the following implementation item that must be completed prior to implementation of the RICT program to satisfy the guidance in NEI 06-09 that the PRA reflect the as-built, as-operated plant and that the PRA technical adequacy is acceptable:

- Exelon will ensure that the updated standby liquid control cable data will be incorporated in the Peach Bottom PRA with sufficient detail to accurately calculate the RICT.

LAR Attachment 6 also states that if implementation of this change constitutes a PRA upgrade as defined in the PRA standard, as endorsed by RG 1.200, then a focused-scope peer review will be performed on this change, and any findings will be resolved and incorporated in the PRA prior to the implementation of the RICT program. However, it is unclear to the NRC staff how the addition of this system model will meet CC-II of the PRA standard, as endorsed by

RG 1.200.

In light of this observation, describe how this system will be adequately modeled in the FPRA and in accordance with the PRA standard's CC-II.

APLB QUESTION 06 – Well-Sealed Motor Control Center Cabinets

Guidance in Frequently Asked Question (FAQ) 08-0042, "Fire Propagation from Electrical Cabinets" (from Supplement 1 of NUREG/CR-6850), applies to electrical cabinets below
440 volts (V). With respect to Bin 15 as discussed in Chapter 6 of NUREG/CR-6850, it clarifies the meaning of "robustly or well-sealed." Thus, for cabinets of 440 V or less, fires from well-sealed cabinets do not propagate outside the cabinet. For cabinets of 440 V and higher, the original guidance in Chapter 6 remains and requires that Bin 15 panels, which house circuit voltages of 440 V or greater, are counted, because an arcing fault could compromise panel integrity (an arcing fault could burn through the panel sides, but this should not be confused with the high energy arcing fault type fires).

FAQ 14-0009, "Treatment of Well-Sealed MCC Electrical Panels Greater than 440V" (ADAMS Accession No. ML15119A176), provides the technique for evaluating fire damage from MCC cabinets having a voltage greater than or equal to 440 V. Therefore, propagation of fire outside the ignition source panel must be evaluated for all MCC cabinets that house circuits of 440 V or greater.

1. a) Describe how fire propagation outside of well-sealed MCC cabinets greater than or equal to 440 V is evaluated.
2. b) If well-sealed cabinets less than 440 V are included in the Bin 15 count of ignition sources, provide justification for using this approach, as this is contrary to the guidance.

APLB QUESTION 07 – Fire Probabilistic Risk Assessment Method for Very Early Warning Fire Detection Systems

LAR Enclosure 9, Section 4, states that the Peach Bottom FPRA was developed using consensus methods outlined in NUREG/CR-6850 and interpretations of technical approaches as required by the NRC. Part (e) of TS 5.5.16 states that the approaches and methods used in the RICT program shall be acceptable to the NRC. Methods to assess risk must be those used to support the LAR or other methods approved by the NRC for generic use.

There have been some changes to the FPRA methodology since the development of the Peach Bottom FPRA that was peer reviewed. The integration of the NRC-accepted FPRA method described in NUREG-2180, "Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear

Facilities (DELORES-VEWFIRE)" (ADAMS Accession No. ML16343A058), may be relevant to this submittal and could potentially impact the TSTF-505 results, CDF, or large early release frequency (LERF).

Section 2.5.5 of RG 1.174 provides guidance that indicates additional analysis is necessary to ensure that contributions from the above influence would not change the conclusions of the LAR.

1. a) If the above guidance has not been implemented in the Peach Bottom FPRA, provide a detailed justification for why the integration of this guidance would not change the conclusions of the LAR and subsequently not impact the TSTF-505 RICT calculations and risk metrics for total CDF and total LERF. As part of this justification, identify any FPRA methodologies used in the FPRA that are no longer accepted by the NRC staff (e.g., guidance provided in FAQ 08-0046, "Closure of National Fire Protection Association 805 Frequently Asked Question 08-0046 Incipient Fire Detection Systems" (ADAMS Accession No. ML093220426), has been retired by letter dated July 1, 2016 (ADAMS Accession No. ML16167A444)). Provide technical justification for its use in TSTF-505 RICT calculations and evaluate the significance of its use on the RICT estimates.

   OR

2. b) Alternatively, if the above guidance has been implemented in the FPRA, provide the following information:
   i. Indicate whether the changes to the FPRA are PRA maintenance or a PRA upgrade as defined in the PRA standard, Section 1-5.4, as qualified by
      RG 1.200, along with justification for this determination.
   ii. Discuss any focused- or full-scope peer reviews performed to evaluate these changes that were determined in item (b)(i) above to constitute a PRA upgrade, including when the peer review was performed and when the peer review report that evaluated the upgrade was approved.

APLB QUESTION 08 – Treatment of Sensitive Electronics

FAQ 13-0004, "Clarifications on Treatment of Sensitive Electronics" (ADAMS Accession No. ML13322A085), provides supplemental guidance for application of the damage criteria

provided in Sections 8.5.1.2 and H.2 of NUREG/CR-6850, Volume 2, for solid-state and sensitive electronics.

1. a) Describe the treatment of sensitive electronics for the FPRA and explain whether it is consistent with the guidance in Frequently Asked Question (FAQ) 13-0004, including the caveats about configurations that can invalidate the approach (i.e., sensitive electronics mounted on the surface of cabinets and the presence of louver or vents).

2. b) If the approach cannot be justified to be consistent with FAQ 13-0004, then justify that the treatment of sensitive electronics has no impact on the RICT calculations.

APLB QUESTION 09 – Probabilistic Risk Assessment Treatment of Dependencies Between Units 2 and 3

Many plants have multiple units adjoined and thus have common areas. For these plants, the risk contribution from fires originating in one unit must be addressed for impacts to the other unit, given the physical proximity of the other unit, common areas, and the existence of shared systems. Therefore, address the following if Units 2 and 3 have common areas and shared systems:

1. a) Explain how the risk contribution of fires originating in one unit is addressed for the other unit, given impacts due to the physical proximity of equipment and cables in one unit to equipment and cables in the other unit. Include identification of locations where a fire in one unit can affect components in the other unit, and explain how the risk contributions of such scenarios are allocated for an RICT calculation.
2. b) Explain how the contributions of fires in common areas are addressed, including the risk contribution of fires that can impact components in both units.
3. c) Explain the extent to which systems are shared by both units and whether shared systems are credited in the PRA models (IEPRA and FPRA) for both units. If shared systems are credited in the PRA models for each unit, then explain how the PRAs address the possibility that a shared system is demanded in both units in response to a single internal events initiating event or fire initiator.

APLB QUESTION 10 - Probabilistic Risk Assessment Model Uncertainty Analysis Results

The NRC staff SE to NEI 06-09 specifies that the LAR should identify key assumptions and sources of uncertainty and should assess/disposition each as to its impact on the RMTS application. LAR Enclosure 9, Table E9-3, identifies the key assumptions and sources of uncertainty for the FPRA and provides dispositions for each source of uncertainty for this TSTF-505 application. The NRC staff reviewed the dispositions provided in LAR Table E9-3 to the key assumptions and sources of modeling uncertainty and noted that not all uncertainties that appeared to have the potential to impact the RICT calculations seemed fully resolved.

LAR Enclosure 9, Table E9-3, identifies post-fire HRA as a source of FPRA modeling uncertainty because fire HEPs must be adjusted to consider the additional challenges present given a fire. The LAR states that industry consensus modeling approaches are used and concludes that this source of uncertainty impact “is expected to be small” with apparently no sensitivities being performed. To address this source of uncertainty, the LAR states that appropriate RMAs would be required – for example, pre-job briefs. It is unclear to the NRC staff how the RMAs will adequately address the impact on RICT values. Therefore, address the following items:

1. a) Justify that the uncertainty associated with post-fire HRA modeling does not have a consequential impact on calculated RICTs for components supporting TS LCO conditions in the RICT program.

   OR

2. b) Explain what RMAs will be considered to compensate for this uncertainty.

APLB QUESTION 11 – Fire Modeling

The LAR referred to risk evaluation and the application of fire modeling technology. The NRC staff was unable to fully evaluate the fire modeling performed as part of the FPRA.

Regarding the acceptability of the FPRA approach, methods, and data, describe the fire modeling calculational model or numerical methods (e.g., fire modeling tools and techniques) used in support of the FPRA.

APLB QUESTION 12 – Damage Thresholds

Part 4 of ASME/ANS RA-Sa-2009 indicates that damage thresholds be established to support the FPRA. The PRA standard further indicates that thermal impact(s) must be considered in determining the potential for thermal damage of SSCs, and appropriate temperature and critical heat flux criteria must be used in the analysis. Therefore, provide the following information:

1. a) Describe how the installed cabling in the fire areas was characterized, specifically regarding the critical damage threshold temperatures and critical heat fluxes for thermoset and thermoplastic cables.
2. b) An IEEE-383 (Institute of Electrical and Electronics Engineers Standard 383, “IEEE Standard for Type Test of Class 1 E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations”) qualified cable may or may not meet the criteria for a ‘thermoset cable.” It is also possible that a non-IEEE-383 qualified cable actually meets the criteria for a “thermoset” cable. Provide clarification on the assumptions that were made in terms of damage thresholds of cables.
3. c) For those areas that are assumed to have thermoset damage criteria, confirm that the cables are actually thermoset and that the potential confusion about IEEE-383/thermoset is not applicable.
4. d) Describe how raceways with a mixture of thermoplastic and thermoset cables are treated in terms of damage thresholds.
5. e) In each fire area where they are credited, explain how cable tray covers, fire-resistant coatings, and fire wraps were credited in terms of delaying or preventing damage of cables. In addition, explain how holes in cable tray covers were treated regarding the fire modeling damage criteria.

f) Explain how the damage thresholds for non-cable components (i.e., pumps, valves, electrical cabinets, etc.) were determined. Identify any non-cable components that were assigned damage thresholds different from those for thermoset and thermoplastic cables, and provide a technical justification for these damage thresholds.

Probabilistic Risk Assessment Licensing Branch C (APLC) Audit Questions

APLC QUESTION 01 – Impacts from Seismic Hazard Frequencies

Section 2.3.1, Item 7 of NEI 06-09, states that the “impact of other external events risk shall be addressed in the RMTS program” and explains that one method to do this is by “performing a reasonable bounding analysis and applying it along with the internal events risk contribution in calculating the configuration risk and the associated RICT.” The NRC staff’s SE for NEI 06-09 states that “Where PRA models are not available, conservative or bounding analyses may be performed to quantify the risk impact and support the calculation of the RICT.”

In Section 3 of Enclosure 4 to the LAR, the licensee stated that the site-specific seismic PRA (SPRA) completed in response to the 10 CFR 50.54(f) request for information associated with the Fukushima Near-Term Task Force (NTTF) activities is not directly used in the RICT program but provides input into the calculation for seismic core damage frequency (SCDF) and seismic large early release frequency (SLERF). The licensee selected the seismic hazard curve that was used in the development of NTTF SPRA model, which is based on the peak ground acceleration (PGA). In the same section of the LAR, the licensee mentioned its seismic hazard and screening report (ADAMS Accession No. ML14090A247), which provided seismic hazard curves at various frequencies at 100 (PGA), 25, 10, 5, 2.5, 1, and 0.5 hertz (Hz). The NRC staff compared the seismic hazard curves between these two documents and found that the PGA hazard curve used in the LAR is different than that in the seismic hazard and screening report.

1. a) Explain the difference between the two PGA hazard curves cited above and justify the selection of the PGA hazard curve for use in the estimation of the SCDF penalty in the LAR.
2. b) Justify that the consideration of seismic hazard curves at frequencies other than the PGA does not significantly change the SCDF penalty proposed in the LAR.

APLC QUESTION 02 – Representativeness of Discretization of Seismic Hazard Curve

The licensee provided the PGA seismic hazard curve data from 0.005 gram (g) to 7.5 g in
Table E4-1 of Enclosure 4 to the LAR. The seismic hazard interval frequencies are represented by discretizing the hazard curve into eight ‘bins’ as shown in Table E4-2 of Enclosure 4 to the LAR. The representative PGA for the last ‘bin’ is selected to be 0.99 g for representing the entire hazard from 0.9 g to 7.5 g. This approach results in a mean fragility probability of 0.95 instead of 1.0 as shown in Table E4-3 of Enclosure 4 to the

LAR. As explained in Enclosure 4 to the LAR, this change has a minor impact on the estimated SCDF value. However, the NRC staff notes that sensitivity analysis 1d in the licensee's SPRA report (ADAMS Accession No. ML18240A065) shows a 17 percent increase in SLERF due to refinement in the discretization of the last 'bin.' This is likely to increase the seismic conditional large early release probability (SCLERP) estimate, and therefore, the SLERF penalty estimate. The LAR does not discuss the impact of the refinement of the discretization for the last 'bin' on the estimated SLERF penalty.

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Justify that the selected representative PGA of 0.99 g for the last 'bin' is reasonable and conservative for the estimated SLERF penalty or provide an updated SLERF penalty.

APLC QUESTION 03 – Seismic Core Damage Frequency and Large Early Release Frequency Penalty Estimate

Section 2.3.1, Item 7 of NEI 06-09, states that the "impact of other external events risk shall be addressed in the RMTS program" and explains that one method to do this is by "performing a reasonable bounding analysis and applying it along with the internal events risk contribution in calculating the configuration risk and the associated RICT." The NRC staff's SE for NEI 06-09 states that "Where PRA models are not available, conservative or bounding analyses may be performed to quantify the risk impact and support the calculation of the RICT."

The seismic penalty approach is used to quantify the risk impact and to support the RICT evaluation. The staff notes that there is a site-specific seismic PRA that could be used for this analysis. Section 3 of Enclosure 4 to the LAR states that the site-specific SPRA was not directly used in the RICT program but provided input into the calculation for SCDF and SLERF. The licensee compared the estimated SCDF penalty for the proposed RICT calculations against the point-estimate SCDF from the site-specific SPRA. In addition, the licensee used the SLERF to SCDF ratio from the site-specific SPRA to determine the SLERF penalty for use in the proposed RICT calculations.

The comparison of the estimated SCDF and SLERF penalties against the corresponding point-estimate mean values from the site-specific SPRA does not provide justification that the SCDF and SLERF penalty estimates are conservative, as stated in the NEI 06-09 guidance. There is no upper bound on the change-in-risk calculation, and the change in risk can exceed the base SCDF and SLERF. However, it appears to the NRC staff that the SPRA could provide the means to justify that the proposed SCDF and SLERF penalty estimates are conservative, and therefore, consistent with the staff's SE for NEI 06-09.

Justify that the SCDF and SLERF penalty estimates are conservative based on the results and insights from change-in-risk calculations for the proposed RICTs using the recent site-specific SPRA.

Technical Specifications Branch (STSB) Audit Questions

STSB QUESTION 01 – Technical Specification 3.5.1.E, One ADS [Automatic Depressurization System] Valve Inoperable

LAR Enclosure 1, Table E1-1 lists in the column of “TS 3.5.1.E” a condition with one ADS valve inoperable. The corresponding column of the “SSCs Covered by TS LCO Condition” indicates that ADS (five safety relief valves) are required to be operable, and the column of “Design Success Criteria” indicates that five ADS valves are available.

Clarify for TS 3.5.1.E condition with one of five required ADS valves inoperable, that the Design Success Criteria need 3 or 4 available ADS valves. Discuss the Analyses of Record (AOR) that demonstrated adequacy of 3 or 4 ADS valves for reactor pressure vessel rapid depressurization to mitigate the loss-of-coolant accident consequences and reference the NRC documents approving the AOR of the concern or address the acceptability of the AOR if it was not previously approved by the NRC.

STSB QUESTION 02 – Technical Specification 3.5.1.F, One Automatic Depressurization System valve inoperable and One Low Pressure Emergency Core Cooling System Subsystem Inoperable

LAR Enclosure 1, Table E1-1 lists in the column of “TS 3.5.1.F” a condition with one ADS valve inoperable and one low pressure Emergency Core Cooling System (ECCS) injection/spray subsystem inoperable. Clarify the same for 3.5.1.F. The corresponding column of the “SSCs Covered by TS LCO Condition” states, “See LCO Condition 3.5.1.A and 3.5.1.E,” which indicates that ADS (five safety relief valves) are required to be operable, and the column of “Design Success Criteria” indicates that five ADS valves are available.

Clarify for TS 3.5.1.F Condition with one of 5 required ADS valves inoperable, that the DSC need three or four available ADS valves. Discuss the AOR that demonstrated adequacy of three or four ADS valves for reactor pressure vessel rapid depressurization to mitigate the loss-of-coolant accident consequences and reference the NRC documents approving the AOR of the concern, or address the acceptability of the AOR if it was not previously approved by the NRC.

Electrical Engineering Branch (EEEB) Audit Questions

EEEB QUESTION 01 – Technical Specification 3.8.1.D, Two or More Offsite Alternating Current Power Circuits Inoperable

Peach Bottom’s DSC is derived from the current licensing basis of the plant, as documented in the Updated Safety Analysis Report, and should include a minimum set of required equipment that has the capacity and capability to safely shut down the reactor in case of an accident and maintain it in a safe condition. In Table E1-1 of Enclosure 1 of the LAR, the DSC for TS

LCO 3.8.1.D (two or more offsite AC power circuits inoperable) is “one of two offsite AC power sources.” The NRC staff notes that if both offsite circuits are inoperable, one offsite AC power source as listed in the DSC is not available to provide the necessary power to safely shut down the reactor and maintain it in safe condition. Therefore, it is not clear how one offsite circuit can be the DSC for TS 3.8.1.D during the RICT program entry.

Explain this apparent discrepancy in Table E1-1 of Enclosure 1 of the LAR. Additionally, describe any effect the discrepancy may have on the PRA success criteria for TS LCO 3.8.1.D.

EEEB QUESTION 02 – Technical Specification 3.8.1.B, One Diesel Generator Inoperable

Table E1-1 in Enclosure 1 of the LAR states that the DSC for TS 3.8.1 Condition B is “three of four diesel generators.” Explain the basis for this DSC. Include in the explanation, as necessary to clarify the basis, a description of the onsite AC power system’s design configuration, including each diesel generator’s capacity and loading.

EEEB QUESTION 03 – Technical Specification 3.8.4 Conditions A, B, C, D, and E

Table E1-1 in Enclosure 1 of the LAR states that the DSC for TS 3.8.4 Conditions A, B, C, D, and E is “three of four DC divisions.” Explain the basis for the DSC for these TS conditions. Include in the explanation for each applicable TS condition, as necessary to clarify the basis, a description of each unit’s 125 VDC and 250 VDC system’s configuration, including number and type of batteries and chargers with associated capacities and loading, and use of any cross ties, as applicable.

EEEB QUESTION 04 – Technical Specification 3.8.7 Conditions A, B, C, and D

Table E1-1 in Enclosure 1 of the LAR states that the DSC for TS 3.8.7 Conditions A, B, C, and D is “three of four divisions.” Explain the basis for the DSC for these TS conditions. Include in the explanation for each TS condition, as necessary to clarify the basis, a description of the associated system configuration.

EEEB QUESTION 05 – RMA Examples

As part of its evaluation, the NRC staff reviews the proposed RMA examples for reasonable assurance that the RMAs are considered to monitor and control risk and to ensure adequate defense in depth. Enclosure 12 of the LAR describes the RMAs examples for TS 3.8.1.A,
TS 3.8.1.B, TS 3.8.1.D, and TS 3.8.4.A. However, the LAR does not include the RMA examples for TS 3.8.7 conditions related to the power distribution system. Provide the RMA example(s) for TS 3.8.7.

Instrumentation & Controls Branch B (EICB) Audit Questions

EICB QUESTION 01 – Instrumentation & Controls Redundancy and Diversity

RG 1.174, Revision 3, states the licensee should assess whether the proposed licensing basis change meets the defense-in-depth principle by not overrelying on programmatic activities as compensatory measures associated with the change in the licensing basis. RG 1.174 further elaborates that human actions (e.g., manual system actuation) are considered as one type of compensatory measure.

Therefore, in LAR Attachment 5, if the diverse means identified are the manual actuations, demonstrate by one example that these "manual actuations" identified as the diverse means are modeled in the plant PRA defined in plant operation procedures to which operators are trained, and confirm the completion times associated with these actions are evaluated as adequate.

November 9, 2020 – Letter from Jonathan E. Greives, Chief Reactor Projects Branch 4 Division of Reactor Projects to Bryan C. Hanson Senior Vice President Exelon Generation Company, LLC President and Chief Nuclear Officer Exelon Nuclear with subject of PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 – INTEGRATED INSPECTION REPORT 05000277/2020003 AND 05000278/2020003

On September 30, 2020, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Peach Bottom Atomic Power Station, Units 2 and 3. On October 9, 2020, the NRC inspectors discussed the results of this inspection with Mr. Matthew Herr, Site Vice President, and other members of your staff. The results of this inspection are documented in the enclosed report.

No findings or violations of more than minor significance were identified during this inspection.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with Title 10 of the *Code of Federal Regulations* 2.390, "Public Inspections, Exemptions, Requests for Withholding."

U.S. NUCLEAR REGULATORY COMMISSION Inspection Report

Docket numbers; 05000277 and 05000278
License numbers: DPR-44 and DPR-56
Report numbers: 05000277/2020003 and 05000278/2020003

Enterprise identifier: I-2020-003-0037
Licensee: Exelon Generation Company, LLC
Facility: Peach Bottom Atomic Power Station, Units 2 and 3

Location: Delta, PA 17314
Inspection dates: July 1, 2020 to September 30, 2020

Inspectors: S. Rutenkroger, Senior Resident Inspector

P. Boguszewski, Resident Inspector
E. Andrews, Health Physicist

Approved: Jonathan E. Greives, Chief Reactor Projects Branch 4 Division of Reactor Projects

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting an integrated inspection at Peach Bottom Atomic Power Station, Units 2 and 3, in accordance with the Reactor Oversight Process. The Reactor Oversight Process is the NRC's program for overseeing the safe operation of commercial nuclear power reactors. Refer to https://www.nrc.gov/reactors/operating/oversight.html for more information.

List of Findings and Violations

No findings or violations of more than minor significance were identified.

Additional Tracking Items

None.

PLANT STATUS

Unit 2 began the inspection period at rated thermal power (RTP). The unit remained at or near RTP for the remainder of the inspection period, reaching 83 percent of RTP due to end of cycle coastdown.

Unit 3 began the inspection period at RTP. On September 12, 2020, the unit was down powered to 66 percent for a control rod sequence exchange and turbine valve testing. The unit was returned to RTP the following day, and remained at or near RTP for the remainder of the inspection period.

INSPECTION SCOPES

Inspections were conducted using the appropriate portions of the inspection procedures (IPs) in effect at the beginning of the inspection unless otherwise noted. Currently approved IPs with their attached revision histories are located on the public website at http://www.nrc.gov/reading- rm/doc-collections/insp-manual/inspection-procedure/index.html. Samples were declared complete when the IP requirements most appropriate to the inspection activity were met consistent with Inspection Manual Chapter (IMC) 2515, "Light-Water Reactor Inspection Program - Operations Phase." The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel to assess licensee performance and compliance with Commission rules and regulations, license conditions, site procedures, and standards. Starting on March 20, 2020, in response to the National Emergency declared by the President of the

United States on the public health risks of the coronavirus (COVID-19), resident and regional inspectors were directed to begin telework and to remotely access licensee information using available technology. During this time the resident inspectors performed periodic site visits each week, increasing the amount of time on site as local COVID-19 conditions
permitted. As part of their onsite activities, resident inspectors conducted plant status activities as described in IMC 2515, Appendix D; observed risk significant activities; and completed on site portions of IPs. In addition, resident and regional baseline inspections were evaluated to determine if all or portion of the objectives and requirements stated in the IP could be performed remotely. If the inspections could be performed remotely, they were conducted per the applicable IP. In some cases, portions of an IP were completed remotely and on site. The inspections documented below met the objectives and requirements for completion of the IP.

REACTOR SAFETY

71111.01 - Adverse Weather Protection
Impending Severe Weather Sample (IP Section 03.02) (1 Sample)

(1) The inspectors evaluated the adequacy of the overall preparations to protect risk-significant systems from potential severe weather given a tropical storm warning on August 4, 2020

External Flooding Sample (IP Section 03.03) (1 Sample)

(1) The inspectors evaluated that flood protection barriers, mitigation plans, procedures, and equipment are consistent with the licensee's design requirements and risk analysis assumptions for coping with external flooding on September 22, 2020

71111.04 - Equipment Alignment
Partial Walkdown Sample (IP Section 03.01) (2 Samples)

The inspectors evaluated system configurations during partial walkdowns of the following systems/trains:

1. (1) Unit 3 'B' and 'C' drywell chillers during inspection of the 'A' drywell chiller on June 30, 2020
2. (2) Emergency cooling system during 'A' emergency cooling tower system outage window (SOW) on July 8, 2020

Complete Walkdown Sample (IP Section 03.02) (1 Sample)

(1) Hardened containment vent system and severe accident water addition implementation strategy completed in conjunction with TI-193 inspection on August 12, 2020

71111.05 - Fire Protection
Fire Area Walkdown and Inspection Sample (IP Section 03.01) (5 Samples)

The inspectors evaluated the implementation of the fire protection program by conducting a walkdown and performing a review to verify program compliance, equipment functionality, material condition, and operational readiness of the following fire areas:

1. (1) Unit 3 reactor building (RB) 'C' residual heat removal (RHR) pump and heat exchanger room on July 30, 2020
2. (2) Unit 3 RB closed loop cooling water room on July 30, 2020
3. (3) Unit 2 RB, 195' General Area on July 30, 2020
4. (4) Unit 2 RB, 135' General Area on July 30, 2020
5. (5) Unit common intake structure on September 24, 2020

Fire Brigade Drill Performance Sample (IP Section 03.02) (1 Sample)

(1) The inspectors evaluated the onsite fire brigade performance during an unannounced fire drill on August 13, 2020

71111.07A - Heat Sink Performance
Annual Review (IP Section 03.01) (1 Sample)

The inspectors evaluated readiness and performance of:

(1) Unit common 'E-4' emergency diesel generator (EDG) jacket coolant, lube oil, and air coolant heat exchangers on July 13-14, 2020

71111.11Q - Licensed Operator Requalification Program and Licensed Operator Performance

Licensed Operator Performance in the Actual Plant/Main Control Room (IP Section 03.01) (1 Sample)

(1) The inspectors observed and evaluated licensed operator performance in the control room during the Unit 2 downpower to 90 percent to remove 4th stage feedwater heaters from service followed by power ascension on August 12, 2020

Licensed Operator Requalification Training/Examinations (IP Section 03.02) (3 Samples)

1. (1) The inspectors observed and evaluated licensed operator requalification training in the simulator on July 20, 2020
2. (2) The inspectors observed and evaluated licensed operator requalification training in the simulator on August 11, 2020
3. (3) The inspectors observed and evaluated licensed operator requalification training in the simulator on September 21, 2020

71111.12 - Maintenance Effectiveness
Maintenance Effectiveness (IP Section 03.01) (3 Samples)

The inspectors evaluated the effectiveness of maintenance to ensure the following structures, systems, and components (SSCs) remain capable of performing their intended function:

1. (1) Unit 3 RHR system during the week of August 6, 2020
2. (2) Unit common 'E-1' EDG with review of A(1) Maintenance Review Action Plan on

   September 14, 2020

3. (3) Unit 3 reactor core isolation cooling (RCIC) as of September 25, 2020

Quality Control (IP Section 03.02) (1 Sample)

The inspectors evaluated the effectiveness of maintenance and quality control activities to ensure the following SSC remains capable of performing its intended function:

(1) Unit common 'E-4' EDG overhaul on July 13 to July 16, 2020 71111.13 - Maintenance Risk Assessments and Emergent Work Control Risk Assessment and Management Sample (IP Section 03.01) (5 Samples)

The inspectors evaluated the accuracy and completeness of risk assessments for the following planned and emergent work activities to ensure configuration changes and appropriate work controls were addressed:

1. (1) Unit common 'E-4' EDG planned maintenance on July 14 to July 16, 2020
2. (2) Unit 3 'A' high-pressure service water (HPSW) planned maintenance on July 28, 2020
3. (3) Unit common 'E-3' EDG planned maintenance on August 10 to August 12, 2020
4. (4) Unit 3 'B' HPSW system planned maintenance on August 21, 2020
5. (5) Unit common 'E-4' EDG 24-hour endurance run on September 17, 2020

71111.15 - Operability Determinations and Functionality Assessments
Operability Determination or Functionality Assessment (IP Section 03.01) (8 Samples)

The inspectors evaluated the licensee's justifications and actions associated with the following operability determinations and functionality assessments:

1. (1) Unit common 'E-3' failure to shutdown using normal means requiring tripping of the fuel racks on June 30, 2020
2. (2) Unit 3 high-pressure coolant injection (HPCI) lube oil mechanical trip valve on July 20, 2020
3. (3) Unit 2 RCIC lube oil cooling water pressure was low on August 13, 2020
4. (4) Unit common 'E-2' EDG excessive lube oil usage during 24-hour endurance run

   completed on August 24, 2020

5. (5) Unit 2 standby liquid coolant (SBLC) degraded heater level switch on

   August 27, 2020

6. (6) Unit common '2SU' startup transformer 'A' phase high voltage bushing oil level high

   on August 27, 2020

7. (7) Unit 2 'D' RHR room safety structural supports degraded from groundwater intrusion

   on September 9, 2020

8. (8) Unit 3 'C' RHR flow control valve limit switch indicated closed and the lock nut and

   stem nut were later identified to be loose on September 15, 2020

71111.18 - Plant Modifications

Temporary Modifications and/or Permanent Modifications (IP Section 03.01 and/or 03.02) (1 Sample)

The inspectors evaluated the following temporary or permanent modification:

(1) Unit 2 RCIC staged testing equipment on August 13, 2020 71111.19 - Post-Maintenance Testing
Post-Maintenance Test Sample (IP Section 03.01) (3 Samples)

The inspectors evaluated the following post-maintenance test activities to verify system operability and functionality:

1. (1) Unit common 'E-4' EDG maintenance overhaul on July 18, 2020
2. (2) Unit 3 'B' RHR room flood switch replacement on July 21, 2020
3. (3) Unit 2 SBLC level switch post-maintenance testing on August 26, 2020

71111.22 - Surveillance Testing
The inspectors evaluated the following surveillance tests: Surveillance Tests (other) (IP Section 03.01) (2 Samples)

1. (1) Unit common 'E-2' EDG cardox system surveillance test on July 6, 2020
2. (2) Unit common 'E-3' EDG run with governor troubleshooting on July 22, 2020

Inservice Testing (IP Section 03.01) (1 Sample)
(1) Unit 3 'A' core spray pump, valve, and flow test on August 13, 2020

71114.06 - Drill Evaluation

Select Emergency Preparedness Drills and/or Training for Observation (IP Section 03.01) (1 Sample)

(1) The inspectors observed a full scope emergency preparedness drill conducted on September 9, 2020

Drill/Training Evolution Observation (IP Section 03.02) (2 Samples)

The inspectors evaluated:

1. (1) The inspectors observed a focused area emergency preparedness drill conducted on July 29, 2020
2. (2) The inspectors observed a focused area emergency preparedness drill conducted on August 19, 2020

RADIATION SAFETY

71124.03 - In-Plant Airborne Radioactivity Control and Mitigation Permanent Ventilation Systems (IP Section 03.01) (1 Sample)

The inspectors evaluated the configuration of the following permanently installed ventilation systems:

(1) Main control room emergency ventilation system and standby gas treatment system Temporary Ventilation Systems (IP Section 03.02) (1 Sample)

The inspectors evaluated the configuration of the following temporary ventilation system:

(1) Temporary ventilation for continuing independent spent fuel storage installation work Use of Respiratory Protection Devices (IP Section 03.03) (1 Sample)

(1) The inspectors evaluated the licensee’s use of respiratory protection devices Self-Contained Breathing Apparatus for Emergency Use (IP Section 03.04) (1 Sample)

(1) The inspectors evaluated the licensee’s use and maintenance of self-contained breathing apparatuses

71124.04 - Occupational Dose Assessment
Source Term Characterization (IP Section 03.01) (1 Sample)

(1) The inspectors evaluated licensee performance as it pertains to radioactive source term characterization

External Dosimetry (IP Section 03.02) (1 Sample)

(1) The inspectors evaluated licensee performance as it pertains to external dosimetry that is used to assign occupational dose

Internal Dosimetry (IP Section 03.03) (2 Samples)

The inspectors evaluated the following internal dose assessments for actual internal exposures:

1. (1) There were no actual internal dose assessments during the inspection period
2. (2) The inspectors reviewed one investigational whole body count for an individual. No

   dose assignment was required.

Special Dosimetric Situations (IP Section 03.04) (2 Samples)

The inspectors evaluated the following special dosimetric situations:

1. (1) The dose assessment for one declared pregnant worker in 2019
2. (2) The dose assessment for two declared pregnant workers in 2020

71124.07 - Radiological Environmental Monitoring Program
Environmental Monitoring Equipment and Sampling (IP Section 03.01) (1 Sample)

(1) The inspectors evaluated environmental monitoring equipment and observed collection of environmental samples

Radiological Environmental Monitoring Program (IP Section 03.02) (1 Sample)

(1) The inspectors evaluated the implementation of the licensee's radiological environmental monitoring program

GPI Implementation (IP Section 03.03) (1 Sample)

(1) The inspectors evaluated the licensee's implementation of the Groundwater Protection Initiative program to identify incomplete or discontinued program elements

OTHER ACTIVITIES – BASELINE

71151 - Performance Indicator Verification

The inspectors verified Exelon's performance indicator submittals listed below for the period October 1, 2019 through September 30, 2020 (8 samples):

MS07:

(1) (2)

MS08:

(1) (2)

MS09:

(1) (2)

MS10:

(1) (2)

High Pressure Injection Systems (IP Section 02.06) (2 Samples)

Unit 2 high pressure injection systems Unit 3 high pressure injection systems

Heat Removal Systems (IP Section 02.07) (2 Samples)

Unit 2 heat removal systems Unit 3 heat removal systems

Residual Heat Removal Systems (IP Section 02.08) (2 Samples)

Unit 2 RHR systems Unit 3 RHR systems

Cooling Water Support Systems (IP Section 02.09) (2 Samples)

Unit 2 cooling water support systems Unit 3 cooling water support systems

INSPECTION RESULTS

No findings were identified.

EXIT MEETINGS AND DEBRIEFS

The inspectors verified no proprietary information was retained or documented in this report.

- On August 28, 2020, the inspectors presented the Respiratory Protection and Dose Assessment inspection results to Mr. Matthew Herr, Site Vice President, and other members of the licensee staff.
- On September 25, 2020, the inspectors presented the Radiological Environmental Monitoring Program inspection results to Mr. David Henry, Plant Manager, and other members of the licensee staff.
- On October 9, 2020, the inspectors presented the integrated inspection results to Mr. Matthew Herr, Site Vice President, and other members of the licensee staff.

November 6, 2020 – Email from Jennifer Tobin to David P. Helker (Exelon Nuclear) with subject of Peach Bottom Verbal Relief for Penetration Nozzle (EPID: L-2020-LLR-0144)

On Friday, November 6th at 11:30 a.m., a call was held between NRC and Exelon staff. Thank you all for your participation. Participants and script are noted below and will be added to public ADAMS.

Participants:

NRC:
James Danna Matthew Mitchell Ali Rezai Jennifer Tobin Jonathan Rowley Corey Dukehart

Exelon: David Helker

John O'Neil Thomas Loomis

Jim Cirilli
Heather Malikowski Ben Jordan

Henry T. Ryan Michell Karasek Mark Weis
Sailaja Mokkapati Diane Render Jacqueline Graham Matt Rector

Sarah Ramos

Framatome: David Mancier

David Cofflin Ashok Nana

Ryan Hostler

VERBAL AUTHORIZATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR relief REQUEST I5R-14 regarding ALTERNATE REPAIR OF INSTRUMENT Penetration NOZZLE N-16A on the reactor vessel

EXELON GENERATION COMPANY
PEACH BOTTOM ATOMIC POWER STATION, UNIT 2 DOCKET NO. 50-277
EPID: L-2020-LLR-0144
November 6, 2020

Technical Evaluation read by Matthew Mitchell, Chief of the Piping and Head Penetrations Branch, Office of Nuclear Reactor Regulation

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated November 4, 2020, (Agencywide Documents Access and Management System (ADAMS) Accession ML20309B020), Exelon Generation Company (the licensee) proposed an alternative to certain requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI. The licensee submitted relief request I5R-14 to support the repair of degraded 2-inch reactor vessel (RV) instrument penetration nozzle N- 16A by a half-nozzle repair method at Peach Bottom Atomic Power Station (Peach Bottom), Unit 2, and to facilitate the return to service from the current refueling outage P2R23. The duration of the proposed alternative in this relief request is for one operating cycle (cycle 24), which is scheduled to end in the fall 2022.

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(2), the licensee requested NRC authorization to place into service an alternative repair for the RV instrument penetration nozzle N-16A for duration of operating cycle 24 on the basis that performing a repair in compliance with the specified ASME Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

During the performance of a routine system leakage test in the current refueling outage P2R23, the licensee discovered a leak at RV instrument penetration nozzle N-16A. The licensee proposed to repair the degraded nozzle using a half-nozzle method to restore the pressure boundary. To support its repair option, the licensee proposed an alternative to the following ASME Code requirements, the details of which are documented in the licensee's submittal:

Flaw characterization requirements of ASME Code, Section XI, IWB-3420 and IWB-3620(b), and flaw removal requirements of ASME Code, Section XI, IWA-4412 and IWA-4611.

The licensee has provided the following information to demonstrate that the structural integrity of repaired RV instrument penetration nozzle N-16A will be maintained for the duration of one operating cycle.

An evaluation of the repair design, welding, and nondestructive examination to be performed.

An evaluation of the worst-case flaws left in service in the original J-groove weld that could propagate into the RV shell.

An evaluation of potential loose parts dislodged from the degraded original J-groove weld left in service that could enter the RV during normal power operation.

An evaluation of general corrosion, crevice corrosion, and galvanic corrosion of the RV low-alloy steel that could be exposed to the reactor coolant as a result of the proposed repair method.

The NRC staff finds that the licensee's evaluations collectively address the safety concerns which could be raised by the proposed alternatives to the ASME Code requirements. The NRC staff finds that repaired RV instrument penetration nozzle N-16A is acceptable for the duration of operating cycle 24 based on the information provided in the licensee's submittal.

The NRC staff also finds the licensee's hardship justification is acceptable because performing the repair in accordance with the ASME Code, Section XI would result in an increased radiological exposure. In addition, the NRC staff determines that the licensee's proposed repair of RV instrument penetration nozzle N-16A provides reasonable assurance of structural integrity and leak tightness for operating cycle 24, which is scheduled to end in the fall 2022.

NRC Staff Conclusion read by James Danna, Branch Chief, Plant Licensing Branch I, Office of Nuclear Reactor Regulation

As Chief of Plant Licensing Branch I, I concur with the Piping and Head Penetration Branch's determinations.

The NRC staff concludes that the proposed alternative in relief request I5R-14 provides reasonable assurance of structural integrity and leak tightness of repaired RV instrument penetration nozzle N-16A. The NRC staff determines that complying with the ASME Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2). Therefore, as of November 6, 2020, the NRC staff authorizes the use of request I5R-14 for operating cycle 24, which is scheduled to end in the fall 2022 at Peach Bottom, Unit 2.

All other requirements of ASME Code, Section III and Section XI for which relief was not specifically requested and authorized by the NRC staff remain applicable, including the third-party review by the Authorized Nuclear Inservice Inspector.

This verbal authorization does not preclude the NRC staff from asking additional questions and clarifications regarding relief request I5R-14 while preparing the subsequent written safety evaluation.

November 6, 2020 – Letter from Craig G. Erlanger, Director Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to David P. Helker Sr. Manager, Licensing and Regulatory Affairs Exelon Generation Company, LLC with subject of PEACH BOTTOM ATOMIC POWER STATION, UNITS 1, 2, AND 3, AND INDEPENDENT SPENT FUEL STORAGE INSTALLATION – TEMPORARY EXEMPTION FROM BIENNIAL EMERGENCY PREPAREDNESS EXERCISE FREQUENCY REQUIREMENTS OF 10 CFR PART 50, APPENDIX E, SECTION IV.F.2.C (EPID L-2020-LLE-0151 [COVID-19])

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has approved the below temporary exemption from specific requirements of Appendix E to Title 10 of the *Code of Federal Regulations* Part 50, Section IV.F.2.c, for Peach Bottom Atomic Power Station (Peach Bottom), Units 1, 2, and 3, and independent spent fuel storage installation (ISFSI). This action is in response to your application dated September 25, 2020, as supplemented by letter dated October 9, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML20269A267 and ML20283A772, respectively) from the Pennsylvania Emergency Management Agency (PEMA) which requested a one-time exemption to the requirements in 10 CFR Part 50, Appendix E, Section IV.F.2.c, to exclude the participation of the offsite response organizations (OROs) in the biennial emergency preparedness (EP) exercise for calendar year (CY) 2020.

Exelon Generation Company, LLC (Exelon, the licensee) holds Facility Operating (Possession Only) License No. DPR-12 (decommissioned Unit 1) and Subsequent Renewed Facility Operating License Nos. DPR-44 and DPR-56, which authorize operation of Peach Bottom, Units 2 and 3. The licensee also holds an independent spent fuel storage installation license (License No. 72-29). These licenses are subject to the rules, regulations, and orders of the Commission. The facility consists of two boiling-water reactors located in York County, Pennsylvania.

By letter dated September 25, 2020, Exelon submitted a request for temporary exemption from Appendix E to 10 CFR Part 50, Sections IV.F.2.c, regarding performance of the offsite participation portion of the CY 2020 biennial EP exercise for responsible OROs.

The requirements in 10 CFR Part 50, Appendix E, Section IV.F.2.c, state, in part:

Offsite plans for each site shall be exercised biennially with full participation by each offsite authority having a role under the radiological response plan.

On January 31, 2020, the U.S. Department of Health and Human Services declared a public health emergency (PHE) for the United States to aid the nation’s healthcare community in responding to the Coronavirus Disease 2019 (COVID-19). Subsequently, the Centers for Disease Control and Prevention (CDC) issued recommendations (e.g., social distancing, limiting assemblies) in an attempt to limit the spread of COVID-19.[1]

In your application, you provided the following information:

- • The requested exemption supports the continued implementation of the isolation activities (e.g., social distancing, group size limitations, self-quarantining) to protect required offsite response organization (ORO) personnel in response to the COVID-19 PHE. These activities are needed to ensure supporting State and local government personnel are isolated from COVID-19 and remain capable of executing the functions of the emergency response organization (ERO), as described in the Peach Bottom Emergency Plan (EP), as well as other non-nuclear health and safety functions for the benefit of the public.
- • The ongoing threat of COVID-19 spread has resulted in the inability to safely conduct the December 8, 2020, exercise with full ORO participation. The Commonwealth of Pennsylvania; Chester County, Lancaster County, and York County in Pennsylvania; the State of Maryland; and Cecil County and Harford County in Maryland have communicated to Exelon that the current COVID-19 pandemic response has impacted their ability to prepare for the scheduled Peach Bottom biennial EP exercise and that they are unable to participate in the exercise as currently scheduled. Consequently, Exelon determined it appropriate to request this one-time exemption regarding ORO participation for the CY 2020 Peach Bottom biennial exercise.
- • This one-time schedular exemption to exclude participation of the ORO in the biennial EP exercise in CY 2020 supports continued implementation of the isolation activities (e.g., social distancing, group size limitations, self-quarantining, etc.) to protect required ERO and ORO personnel in response to the COVID-19 PHE.
- • The last biennial EP exercise was conducted on April 17, 2018. Since that time, the licensee has conducted numerous drills, exercises, and other training activities that have exercised its emergency response strategies. As addressed

in the exemption request, exercise and drill dates were provided that demonstrated the continuing level of engagement in EP activities for Peach Bottom and the actual and/or simulated participation with the Commonwealth of Pennsylvania, State of Maryland, and supporting plume exposure emergency planning zone counties in Pennsylvania (i.e., Chester, Lancaster, and York) and Maryland (i.e., Cecil and Harford). In addition to exercises and drills performed since the last biennial EP exercise at Peach Bottom, a listing of training that has been accomplished with the Commonwealth of Pennsylvania, the State of Maryland, and supporting organizations was also provided.

- • The licensee will continue to conduct drills and exercises, as evidenced by its intent "to conduct the December 8, 2020, EP Exercise as scheduled utilizing Station and Corporate personnel..."

[1] CDC, "How to Protect Yourself and Others," April 18, 2020 (ADAMS Accession No. ML20125A069)

- • The licensee made a reasonable effort to reschedule the exercise during CY 2020 with the respective OROs but was unsuccessful. By letter dated April 20, 2020 (ADAMS Accession No. ML20111A170), Exelon formally informed the NRC of its plans to reschedule its biennial EP exercise for Peach Bottom planned for April 21, 2020, to sometime later in the fourth quarter of CY 2020 due to the COVID-19 PHE. The biennial EP exercise was subsequently scheduled for December 8, 2020. During the interim period, Exelon continued to work with its OROs in preparing for the biennial EP exercise. However, the continued response to COVID-19 PHE by the OROs is impacting its ability to effectively prepare and participate in the biennial EP exercise at Peach Bottom scheduled for December 8, 2020. In addition, letters from the Commonwealth of Pennsylvania, the State of Maryland, and supporting Counties from those states communicated to Exelon and the Federal Emergency Management Agency (FEMA) to specifically request the cancellation of the biennial EP exercise due to the COVID-19 PHE. Accordingly, Exelon submitted this exemption request to exclude OROs from its December 8, 2020 biennial EP exercise.
- • The licensee also noted that its submittal contained documentation that included information from the affected OROs that (1) they would not be impacted in a manner that would adversely affect its ability to maintain response capability to support emergency response activities to actual nuclear power plant radiological emergencies, (2) they are in agreement with the requested exemption, and (3) they are committed to maintaining their radiological emergency plans.

Pursuant to 10 CFR 50.12, "Specific exemptions," the NRC may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of
10 CFR Part 50 when (1) the exemptions are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security, and (2) special circumstances are present.

The NRC staff determined that the requested exemption is permissible under the Atomic Energy Act of 1954, as amended, and that no other prohibition of law exists to preclude the activities that would be authorized by the exemption. Therefore, the NRC staff finds that the requested exemption is authorized by law.

The regulations in 10 CFR Part 50, Appendix E, IV.F.2.c, concern requirements for licensees to conduct biennial EP exercises at their facilities. No new accident precursors are created by allowing the licensee to postpone the offsite participation portion of the biennial EP exercise from CY 2020 until CY 2022. Thus, the probability and consequences of postulated accidents are not increased. In addition, the requested exemption for a one-time change to the biennial EP exercise schedule has no relation to security issues. Therefore, the NRC staff finds that the requested exemption will not present an undue risk to the public health and safety and is consistent with the common defense and security.

Special circumstances, per 10 CFR 50.12, that apply to the requested exemption include:

a. 10 CFR 50.12(a)(2)(ii): "Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule."

The regulation in 10 CFR Part 50, Appendix E, Section, IV.F.2.c, requires offsite plans for each site to be exercised biennially with full participation by each offsite authority having a role under the plan. The underlying purpose of these requirements is to ensure that the emergency organization personnel are familiar with their duties and to identify and correct any weaknesses that may exist in the licensee's EP Program. The underlying purpose of Section IV.F.2.c is also to test and maintain interfaces among affected State and local authorities and the licensee.

The NRC recognizes that even if a licensee were to be exempted from the requirement to conduct an offsite biennial exercise in CY 2020, in the event of an actual radiological emergency, offsite authorities would respond. Offsite authorities in all states are currently demonstrating response capabilities, including making decisions on protective actions for the public, in response to the COVID-19 PHE.[2] Additionally, the NRC continues to monitor U.S. nuclear power plants to ensure that they operate safely during the COVID-19 PHE and that defense in depth is maintained to prevent accidents from happening and to mitigate their consequences.

The NRC has consulted with FEMA on the readiness of OROs and the use of this information to inform the NRC decision to grant exemptions, per the NRC/FEMA Memorandum of Understanding.[3] FEMA has recently performed assessments of all offsite emergency response plan capabilities and has concluded that offsite radiological EP remains adequate to provide reasonable assurance that appropriate measures can and will be taken to protect the health and safety of the public in a radiological emergency during the COVID-19 PHE.[4] FEMA monitors response and preparedness capabilities of the OROs to ensure that the response to the current PHE does not adversely impact the ability to protect the public health and safety in the event of a radiological emergency at a commercial nuclear power plant. Exercises are just one of the many methods by which FEMA assesses and validates the adequacy of ORO plans and the ability to implement those plans. In accordance with current FEMA program guidance,[5] FEMA has alternative means of conducting these assessments.

Based on the above, granting a request for exemption from the 10 CFR Part 50, Appendix E, Section IV.F.2.c requirement for offsite biennial exercises in CY 2020, with the next performance of the exercise to be no later than the end of CY 2022, would allow State and local governments to continue to focus their essential response efforts on the COVID-19 PHE. This exemption would apply only to the requirements of 10 CFR Part 50, Appendix E, Section IV.F.2.c, and would not address 44 CFR Part 350. An exemption from Section IV.F.2.c would not prevent a State or local authority, at its discretion, from demonstrating key skills in drills and exercises for the 8-year exercise cycle or prevent a State or local authority from conducting the exercise in CY 2020 or CY 2021.

[2] COVID-19 Resources for State Leaders, Executive Orders – By State, accessed August 12, 2020, https://web.csg.org/covid19/executive-orders/
[3] "Memorandum of Understanding (MOU) Between the Department of Homeland Security/Federal Emergency Management Agency and Nuclear Regulatory Commission Regarding Radiological Response, Planning and Preparedness," December 7, 2015 (ADAMS Accession No. ML15344A371)
[4] FEMA Preparedness Assessments (ADAMS Accession Nos. ML20164A275, ML20174A603, ML20141L795, ML20170B043, ML20170B171, ML20167A175, ML20164A038, ML20154K696, ML20154K617, ML20150A110, and ML20162A056)
[5] Program Manual, Radiological Emergency Preparedness, FEMA P-1028, December 2019, accessed August 12, 2020, https://www.fema.gov/media-library-data/1577108409695-4e49a0a56c8c62695dcc301272a1eda7/FEMA_REP_Program_Manual_Dec_2019.pdf

The licensee stated that it has conducted drills, exercises, and other training activities that have exercised its emergency response strategies since the last evaluated biennial EP exercise, and that State and local OROs have participated.

Therefore, the NRC staff finds that the underlying purposes of 10 CFR Part 50, Appendix E, Section, IV.F.2.c, are met with the rescheduled offsite biennial EP exercise to occur in CY 2022.

b. 10 CFR 50.12(a)(2)(v): "The exemption would provide only temporary relief from the applicable regulation and the licensee or applicant has made good faith efforts to comply with the regulation."

The Pennsylvania Emergency Management Agency (PEMA), Maryland Emergency Management Agency (MEMA), and supporting Counties from Pennsylvania (i.e., Chester, Lancaster, and York) and Maryland (i.e., Cecil and Harford) have communicated to Exelon and to the Federal Emergency Management Agency (FEMA) that the current COVID-19 pandemic response has impacted their ability to prepare for the scheduled Peach Bottom EP Exercise and that they are unable to participate in the EP Exercise as currently scheduled (i.e., December 8, 2020).

PEMA and MEMA will continue to work with FEMA in support of further relief for offsite participation, as appropriate, under FEMA's requirements in 44 CFR 350.9.

Therefore, granting the requested exemption from the 10 CFR Part 50, Appendix E, Section IV.F.2.c requirement for offsite participation in the CY 2020 biennial EP exercise for Peach Bottom, with the next performance of the exercise to be no later than the end of CY 2022, would provide only temporary relief from the applicable regulation, and the licensee has made good faith efforts to comply with the regulation.

Based on the above, the NRC staff finds that the special circumstances of 10 CFR 50.12(a)(2)(ii) and 10 CFR 50.12(a)(2)(v) are present.

NRC approval of the requested exemption is categorically excluded under 10 CFR 51.22(c)(25), and there are no extraordinary circumstances present that would preclude reliance on this exclusion. The NRC staff determined, per 10 CFR 51.22(c)(25)(vi)(E), that the requirements from which the exemption is sought involve education, training, experience, qualification, requalification, or other employment suitability requirements.

The NRC staff also determined that approval of this exemption involves no significant hazards consideration because it does not authorize any physical changes to the facility or any of its safety systems, does not change any of the assumptions or limits used in the licensee's safety analyses, and does not introduce any new failure modes. There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite because this exemption does not affect any effluent release limits as provided in the licensee's technical specifications or by the regulations in 10 CFR Part 20, "Standards for Protection Against Radiation." There is no significant increase in individual or cumulative public or occupational radiation exposure because this exemption does not affect limits on the release of any radioactive material or the limits provided in 10 CFR Part 20 for radiation exposure to workers or members of the public. There is no significant construction impact because this exemption does not involve any changes to a construction permit. There is no significant increase in the potential for or consequences from radiological accidents because the

exemption does not alter any of the assumptions or limits in the licensee's safety analysis. In addition, the NRC staff determined that there would be no significant impacts to biota, water resources, historic properties, cultural resources, or socioeconomic conditions in the region. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the approval of the requested exemption.

Granting the requested exemption does not impact NRC findings of reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency at Peach Bottom. In the statement of considerations for the standards to be applied when considering whether to grant exemptions ("Specific Exemptions; Clarification of Standards, Final Rule," 50 FR 50764, dated December 12, 1985), the Commission stated:

While compliance with all NRC regulations provides reasonable assurance of adequate protection of the public health and safety, the converse is not correct, that failure to comply with one regulation or another is an indication of the absence of adequate protection, at least in a situation where the Commission has reviewed the noncompliance and found that it does not pose an "undue risk" to the public health and safety. Furthermore, the Commission has never defined the concept of "defense-in-depth" to preclude the granting of an exemption from a regulation as long as the applicable exemption criteria are met. In fact, the Commission has recognized that its regulations may provide for the possibility of exemptions when an appropriately high level of safety is in fact achieved and the public interest is served.

The NRC staff has determined that in accordance with 10 CFR 50.12, the requested exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; and that special circumstances are present. Therefore, the NRC hereby grants the licensee's request for a one-time schedular exemption from the requirements for the biennial EP exercise in 10 CFR Part 50, Appendix E,

Section IV.F.2.c.

This exemption expires on December 31, 2022, or when the offsite biennial EP exercise is performed in CY 2022, whichever occurs first.

If you have any questions, please contact the Peach Bottom project manager, Jennifer Tobin, at 301-415-2328 or by e-mail to Jennifer.Tobin@nrc.gov.

November 10, 2020 – Letter from Anthony Dimitriadis, Chief Decommissioning, ISFSI, and Reactor HP Branch Division of Nuclear Materials Safety to Bryan Hanson Senior Vice President, Exelon Generation, LLC President and Chief Nuclear Officer, Exelon Nuclear with subject of NRC INSPECTION REPORT NO. 05000171/2020001, EXELON GENERATION COMPANY, LLC, PEACH BOTTOM ATOMIC POWER STATION, UNIT 1

On August 31 – October 16, 2020, the U.S. Nuclear Regulatory Commission (NRC) conducted an inspection under Inspection Manual Chapter 2561, "Decommissioning Power Reactor Inspection Program," at the Peach Bottom Atomic Power Station Unit 1 (Peach Bottom 1). An on-site inspection was performed August 31 – September 2, 2020. Additional inspection activities (in-office reviews) were conducted remotely as a consequence of the COVID-19 public health emergency (PHE) during the inspection period. The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and the conditions of your license. The inspection consisted of plant walkdowns by the inspectors, observations, interviews with site personnel, and a review of procedures and records.

The results of the inspection were discussed with Mr. Matt Herr, Site Vice President, and other members of your organization on October 21, 2020, via teleconference at the conclusion of the inspection. The enclosed report presents the results of this inspection.

The report documents one NRC-identified violation of NRC requirements of very low safety significance (Severity Level IV). Because of the very low safety significance and because it was entered into your corrective action program, the NRC is treating the violation as a Non-Cited Violation (NCV) consistent with Section 2.3.2.a of the NRC Enforcement Policy.

If you contest the subject or severity of this NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region I, 2100 Renaissance Blvd., Suite 100, King of Prussia, PA 19406-2713; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Peach Bottom Atomic Power Station.

In accordance with Title 10 Code of Federal Regulations (CFR) 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure(s), and your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC document system (ADAMS), accessible from the NRC website at http://www.nrc.gov/reading- rm/adams.html. To the extent possible, your response, if any, should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction.

Current NRC regulations and guidance are included on the NRC's website at www.nrc.gov; select **Radioactive Waste**; **Decommissioning of Nuclear Facilities**; then **Regulations, Guidance and Communications**. The current Enforcement Policy is included on the NRC's website at www.nrc.gov; select About NRC, Organizations & Functions; Office of Enforcement; you may also obtain these documents by contacting the Government Printing Office (GPO) toll-free at 1-866- 512-1800. The GPO is open from 8:00 a.m. to 5:30 p.m. EST, Monday through Friday (except Federal holidays). Enforcement documents; then Enforcement Policy (Under 'Related Information').

No reply to this letter is required. Please contact Harry Anagnostopoulos at 610-337-5322 if you have any questions regarding this matter.

U.S. NUCLEAR REGULATORY COMMISSION REGION I

INSPECTION REPORT

Inspection number: 05000171/2020001
Docket number: 05000171
License number: DPR-12
Licensee: Exelon Generation Company, LLC (Exelon)

Facility: Peach Bottom Atomic Power Station, Unit 1

Address: Delta, Pennsylvania

Inspection dates: August 31 – October 16, 2020

Inspectors: Harry Anagnostopoulos, Senior Health Physicist Decommissioning, ISFSI, and Reactor HP Branch Division of Nuclear Materials Safety

Elizabeth Andrews, Health Physicist Decommissioning, ISFSI, and Reactor HP Branch Division of Nuclear Materials Safety

Approved by: Anthony Dimitriadis, Chief Decommissioning, ISFSI, and Reactor HP Branch Division of Nuclear Materials Safety

EXECUTIVE SUMMARY

Exelon Generation Company, LLC
Peach Bottom Atomic Power Station Unit 1 (Unit 1) NRC Inspection Report No. 05000171/2020001

An announced safety inspection was conducted on August 31 – October 16, 2020. The onsite inspection was conducted on August 31 – September 2, 2020, with additional inspection activities conducted remotely during the inspection period as a consequence of the COVID-19 public health emergency (PHE). The inspectors reviewed activities associated with decommissioning performance and status, management oversight, corrective action program (CAP), and site radiological programs. The inspection consisted of plant walkdowns, interviews with Exelon personnel, and a review of procedures and records. The NRC's program for overseeing the safe operation of a shut-down nuclear power reactor is described in Inspection Manual Chapter (IMC) 2561, "Decommissioning Power Reactor Inspection Program." Based on the results of this inspection, one Non-cited Violation of NRC requirements was identified.

List of Violations

A non-cited violation (NCV) associated with Technical Specification, (TS), Section 2.3 was identified by the inspectors because Exelon Generation, LLC (Exelon) had not performed semi- annual inspections of PBAPS Unit No. 1 that included a radiological survey of all of the accessible portions of the exclusion area and inspection of all of the accessible areas below ground level in the containment vessel to monitor for potential water accumulation. From approximately July 1978 until October 16, 2020, the radiological surveys and inspections did not routinely include the areas of containment that may be accessed by unlocking a barrier on the Refueling Floor in order to access the intermediate and ground-levels floors. All or portions of these floors are located within additional controlled areas inside of the containment. Upon identification, Exelon entered the issue into its corrective action program as IR4367002.

REPORT DETAILS

1.0 Background

Peach Bottom Unit 1 was a high temperature gas-cooled demonstration power reactor that operated from February 1966 until October 31, 1974. It has been permanently shut down and has been in safe storage (SAFSTOR) since that time. Exelon plans to dismantle Unit 1 in parallel with the decommissioning of the operational Units 2 and 3 after they have been permanently shut down. All fuel has been removed from the reactor and shipped to an offsite facility. The spent fuel pool has been drained, decontaminated, and radioactive liquids have been removed. Intrusion water that collects in the reactor containment sump is periodically pumped out of the sump and transferred to the common radwaste building for Peach Bottom Units 2 and 3 for processing.

The NRC's program for overseeing the safe operation of a shut-down nuclear power reactor is described in IMC 2561.

2.0 SAFSTOR Performance and Status Review

a. Inspection Scope (Inspection Procedures (IPs) 36801, 37801, 40801, 71801, 83750, 84750, 86750)

A routine announced safety inspection was conducted from August 31 – October 16, 2020 (onsite from August 31 – September 2, 2020), at Unit 1. The inspection consisted of plant walkdowns, interviews with Exelon personnel, and a review of procedures and records. The inspectors reviewed the SAFSTOR program as outlined in the Updated Final Safety Analysis Report (UFSAR), technical specifications (TS), and procedure DC-PB-800, "Unit 1 Process Control Program," to assess the adequacy of management oversight for the Unit 1 facility. Specifically, the inspectors reviewed the decommissioning management and staff organization and Exelon's implementation of Unit 1 programs for the SAFSTOR phase of decommissioning. The inspectors also conducted a walk-down to assess the material condition of the Unit 1 buildings.

The inspectors reviewed the results of the Exelon "Unit 1 Exclusion Area Inspection" semi-annual surveillance tests to ensure exclusion area barriers, radiological conditions, water intrusion, and effluent release limits are as specified in the TS. Additionally, as part of this review, the inspectors reviewed the program for structural monitoring inspections at Unit 1.

The inspectors reviewed activities, components, and documentation associated with the following SAFSTOR programs: decommissioning organization, staffing, and cost controls; safety reviews and modifications; fire protection, maintenance and surveillance, and license compliance.

The inspectors evaluated whether the following programs that remain associated with programs for Units 2 and 3 are being adequately applied and administered for Unit 1:

occupational radiation exposure, radioactive effluent technical specifications (RETS), the site radiological environmental monitoring program (REMP), and the processing and transportation of radioactive waste.

The inspectors reviewed Exelon fleet audit reports, corrective action program (CAP) documents, and the onsite and offsite safety review committee activities associated with Unit 1 to determine if issues were being appropriately identified, assessed and reviewed and that corrective actions were being appropriately implemented.

The inspectors also reviewed radiological survey reports and condition reports. b. Observations and Findings

The inspectors verified that management oversight was adequate for the SAFSTOR phase of decommissioning and that no significant changes had been made to the Unit 1 SAFSTOR organization since the previous inspection. The inspectors confirmed that no design changes or plant modifications were made since the previous inspection.

The inspectors entered the Unit 1 containment with members of the Peach Bottom staff for a plant walkdown. The entrances to containment were barricaded (as required by TS) with a locked fence and posted as a "radiologically controlled area" (RCA). All entrants were briefed on radiological conditions, signed-onto a radiological work permit, and escorted by a radiological protection technician. Upon entry, the inspectors noted that several additional areas within the containment were also barricaded and posted as RCAs. In some cases, the barricades were expanded metal barriers that were bolted into the wall. In two cases, the barricades consisted of fences with locked gates.

The inspectors questioned the use of the additional barricades, and the reasons for which the walkdowns and surveillances were limited to a small portion of the facility. Site personnel indicated that inspection of the balance of the facility was limited by the UFSAR and that entry into these "controlled areas" was only permitted if routine inspections indicated a potential problem. The inspectors completed the tour of the allowed areas of the Unit 1 containment.

The inspectors then reviewed the last two semi-annual surveillances (ST-H-099-960-2, Revision 26) for the Unit 1 Exclusion Areas, which included the Unit 1 containment. The surveillances are intended to satisfy TS 2.3(b)2 and 2.3(b)4 for radiological survey and water inspection, respectively. The inspectors also reviewed the last set of routine radiological surveys for Unit 1. The inspectors found that for the 2019 and 2020 surveillance efforts, the grating over the southwest stairway leading from the refueling floor to the lower levels (B-14) had not been unlocked or opened to allow for full inspection of the intermediate and ground level floors.

The inspectors identified a non-cited violation (NCV) for NRC License No. DPR-12, Amendment 10, Appendix A, Technical Specifications (TS) for Peach Bottom Atomic Power Station (PBAPS) Unit No. 1, Section 2.3 that requires, in part, the performance of semi-annual inspections, including a radiological survey of the accessible portions of the exclusion area and inspection of the accessible areas below ground level in the containment vessel for potential water accumulation.

TS Section 1.0 defines the exclusion area as the area within the PBAPS which is enclosed within locked barriers, and contains the Containment Vessel, Spent Fuel Pool Building and Radwaste Building. TS 2.1(b) specifies exclusion area barriers that shall remain locked except when opened to provide egress for inspections, surveys and repairs. These barriers are described in TS 2.1(b)1 and include the grating over the Southwest stairwell (B-14) on the Refueling Floor.

Contrary to the above, from 1978 until October 16, 2020, Exelon Generation, LLC (Exelon) had not performed semi-annual inspections of PBAPS Unit No. 1 that include a radiological survey of all of the accessible portions of the exclusion area and inspection of all of the accessible areas below ground level in the containment vessel for potential water accumulation. Specifically, from approximately July 1978, until October 16, 2020, the radiological surveys and water accumulation inspections had not routinely included the areas of containment that may be accessed by unlocking barrier B-14 on the Refueling Floor in order to survey and inspect the intermediate and ground-level floors. All or portions of these floors are located within additional controlled areas inside of the containment. Upon identification, Exelon entered the issue into its corrective action program as IR4367002.

Because this violation was of low safety significance and was entered into Exelon's CAP, this violation is being treated as a Non-cited Violation, consistent with Section 2.3.2.a of the Enforcement Policy.

The inspectors confirmed that other surveillances were performed as required by the TS.

During the inspection, it was noted that although Exelon had performed structural monitoring inspections at Unit 1, it did not have a specific procedure that governs the effort. In addition, while the structural inspections are documented in surveillance records, the auxiliaries building (which includes the radwaste facility) was not explicitly included in the scope of those surveillance records. The inspectors also noted that the structural inspections inside the reactor containment did not include several large areas located behind the "controlled area" barricades (as identified in the non-cited violation). Exelon initiated Issue Report 04378229 to document and evaluate the NRC's concerns regarding the auxiliaries building.

The inspectors reviewed information associated with the status of the decommissioning trust fund and its expenditures, as provided by Exelon during the on-site inspection period, and with no immediate concerns identified.

Tours of the Unit 1 buildings and grounds indicated that the level of attention expended for maintenance, repair, storage, lighting, humidity, fire loading, and adverse weather protection were appropriate for a unit placed in SAFSTOR.

The inspectors verified that any groundwater collected in Unit 1 was transferred to Units 2 and 3 in accordance with plant procedures. The annual radiological effluent and the annual REMP reports demonstrated that all calculated doses were below regulatory dose criteria of 10 Code of Federal Regulations Part 50, Appendix I.

The inspectors verified that programs that remain relevant to Units 2 and 3 were being adequately applied and administered to Unit 1.

The inspectors determined that issues were being identified and entered into the CAP in a timely manner and the issues were effectively screened, prioritized and evaluated commensurate with their safety significance.

c. Conclusions

Based on the results of this inspection, one violation of NRC requirements was identified.

3.0 Exit Meeting Summary

On October 21, 2020, the inspectors presented the inspection results to Mr. Matt Herr, Site Vice President, and other members of Exelon's staff. No proprietary information was retained by the inspectors or documented in this report.

Licensee

M. Herr
R. Stiltner
J. Laverde D. Baracco D. Dullum D. Mink
D. Hines
J. Neff
B. Miller
C. Hardee S. Patterson S. Hansell R. Gropp

None

PARTIAL LIST OF PERSONS CONTACTED

Site Vice President
Engineering Director
Design Engineering Manager Radiation Protection Manager Senior Regulatory Engineer
Site Engineering
Radiation Protection Supervisor Fire Marshall
Fire Protection Engineer Project Manager
Radiation Protection Technician Regulatory Assurance
Nuclear Licensing

ITEMS OPEN, CLOSED, AND DISCUSSED

LIST OF DOCUMENTS REVIEWED

Action Requests (AR) 4236437, 4297164, 4297294, 4299794, 4349717, 4367002, 4367005, 4367008

Condition Reports (CR) 808191, 886211, 924196, "Decommissioning Peach Bottom Unit 1," Final Report, Prepared by Catalytic, Inc., July 1978

"Decommissioned Units Audit Report," NOSA-PEA-19-10 (AR4289618)
"Entry Into Unit 1 During SAFSTOR Decommissioning Status," DC-PB-800-1000, Revision 2 Meeting minutes, PORC Meeting No. 19-15, August 2, 2019
"PBAPS Unit 1 Decommissioning Status Report – 2019," March 27, 2020
"Quality Assurance Topical Report (QATR)," NO-AA-10, Revision 95
Radiation Work Permit PB-C-20-00121, Revision 0
Radiological surveys, Unit 1, dated May 18, 2020
"Report on Status of Decommissioning Funding for Reactors and Independent Spent Fuel

Storage Installations," RS-19-045, April 1, 2019
"Report on Status of Decommissioning Funding for Shutdown Reactors," RS-20-039, March 31,

2020
Spreadsheet, Peach Bottom Expenses for 2019
"Structures Monitoring," ER-AA-450, Revision 9
Technical Specifications for Peach Bottom Atomic Power Station Unit No. 1, Appendix A to

License No. DPR-12, Amendment 11

LIST OF DOCUMENTS REVIEWED (Cont'd)

"Unit 1 Process Control Program," DC-PB-800, Revision 0
"Updated Final Safety Analysis Report," Peach Bottom Atomic Power Station Unit 1, Revision 9, April 2018
Work orders 0492695701, 04987515-01

CAP Exelon IMC
IP NRC PBAPS RCA REMP RETS TS UFSAR Unit 1

LIST OF ACRONYMS USED

Corrective Action Program
Exelon Generation Company, LLC
Inspection Manual Chapter
Inspection Procedure
Nuclear Regulatory Commission
Peach Bottom Atomic Power Station
Radiologically Controlled Area
Radiological Environmental Monitoring Program Radiological Environmental Technical Specifications Technical Specification
Updated Final Safety Analysis Report
Peach Bottom Atomic Power Station Unit 1

November 19, 2020 – Email from Jennifer Tobin to David Helker (Exelon Nuclear) cc to Richard Gropp (Exelon Nuclear) with subject of Acceptance of Requested Licensing Action: Peach Bottom One-time Exemption App. B Sect. VI.C.3.(I)(1) Force on Force [COVID-19] (EPID No. L-2020-LLE-0192)

By letter dated November 13, 2020 (ADAMS Accession No. ML20318A287), Exelon Generation Company, LLC, (the licensee) submitted an exemption request for Peach

Bottom Unit 2 and 3. The purpose of this e-mail is to provide the results of the U.S. Nuclear Regulatory Commission (NRC) staff's acceptance review of this exemption request. The acceptance review was performed to determine if there is sufficient technical information in scope and depth to allow the NRC staff to complete its detailed technical review. The acceptance review is also intended to identify whether the application has any readily apparent information insufficiencies in its characterization of the regulatory requirements or the licensing basis of the plant.

The NRC staff has reviewed your application and concluded that it does provide technical information in sufficient detail to enable the NRC staff to complete its detailed technical review and make an independent assessment regarding the acceptability of the proposed exemption in terms of regulatory requirements and the protection of public health and safety and the environment. Given the lesser scope and depth of the acceptance review as compared to the detailed technical review, there may be instances in which issues that impact the NRC staff's ability to complete the detailed technical review are identified despite completion of an adequate acceptance review. If additional information is needed, you will be advised by separate correspondence.

Based on the information provided in your submittal, the NRC staff has estimated that this licensing request will take approximately **60 hours** to complete. The NRC staff expects to complete this review no later than **December 31, 2020**. If there are emergent complexities or challenges in our review that would cause changes to the initial forecasted completion date or significant changes in the forecasted hours, the reasons for the changes, along with the new estimates, will be communicated during the routine interactions with the assigned project manager.

These estimates are based on the NRC staff's initial review of the application and they could change, due to several factors including requests for additional information, and unanticipated addition of scope to the review. Additional delay may occur if the submittal is provided to the NRC in advance or in parallel with industry program initiatives or pilot applications.

Please contact me if you have any questions. A copy of this email will be made publicly available in ADAMS.

December 1, 2020 – Letter from Blake Purnell, Project Manager Plant Licensing Branch III Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to EXELON GENERATION COMPANY, LLC with subject line of SUMMARY OF NOVEMBER 4, 2020, MEETING WITH EXELON GENERATION COMPANY, LLC REGARDING A PLANNED REQUEST FOR AN ALTERNATIVE TO REDUCE THE FREQUENCY OF UPDATES TO ITS INSERVICE TESTING AND INSPECTION PROGRAMS (EPID L-2020-LRM- 0089)

On November 4, 2020, a Category 1 public meeting was held between the U.S. Nuclear Regulatory Commission (NRC or Commission) staff and representatives of Exelon Generation Company, LLC (Exelon). The purpose of the meeting was to discuss a proposed alternative to certain requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a, "Codes and standards," for the subject facilities. The meeting notice and agenda are available in the Agencywide Documents Access and Management System (ADAMS) at Accession
No. ML20294A032. A copy of Exelon's presentation is available in ADAMS at Accession No. ML20261G732. A list of attendees is enclosed.

Background

The regulations in 10 CFR 50.55a include, in part, requirements for the use of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPV Code) and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) for the inservice inspection (ISI) and inservice testing (IST) of nuclear power plants. Specific editions and addenda of these ASME Codes have been incorporated by reference into 10 CFR 50.55a, subject to certain limitations. Every 10 years, licensees are required to update their ISI and IST programs to the latest editions and addenda of the applicable ASME Code incorporated by reference in 10 CFR 50.55a.

Discussion

For the subject facilities, Exelon proposes to reduce the frequency of updates to the ISI and IST programs from every 10 years to every 24 years. Under this proposal, each facility would remain on the current code of record for two consecutive 12-year intervals. Exelon plans to submit this proposal as a fleetwide alternative request under 10 CFR 50.55a(z). Exelon stated that it would be ready to submit in early December 2020. As part of this proposed alternative, Exelon would request an extension of the current and next ISI and IST intervals from 10 years to 12 years. Exelon would request the next ISI and IST intervals to start when the current extended intervals end. Specific details regarding the request are included in Exelon's presentation.

Exelon believes that its proposed alternative could be approved under 10 CFR 50.55a(z) based on a plain-language reading of the rule. The regulations in 10 CFR 50.55a(z) allow the NRC staff to approve alternatives to paragraphs (b) through (h) of 10 CFR 50.55a if the licensee demonstrates that (1) the proposed alternative provides an acceptable level of quality and safety or (2) the specific requirements in 10 CFR 50.55a result in a hardship without a compensating increase in quality and safety. Exelon stated that it could meet both criteria under 10 CFR 50.55a(z). In addition, Exelon believes that the Commission policy in the Staff Requirements Memorandum (ADAMS Accession No. ML003702722) for SECY-00-0011, "Evaluation of the Requirement for Licensees to Update Their Inservice Inspection and Inservice Testing Programs Every 120 Months" (ADAMS Accession No. ML003675659), does not preclude the NRC staff from approving its proposed alternative.

Exelon stated that the basis for its proposed alternative would be generic, rather than plant specific. The NRC staff stated that it would expect the application for the proposed alternative to include a comparison between the current code of record at each site and the specific edition or addenda of the applicable ASME Codes that will not be implemented or not implemented on schedule. Typically, the NRC staff reviews proposed alternatives against the requirements in 10 CFR 50.55a and the applicable ASME Codes. Exelon stated that it could not provide this information because it does not know which editions and addenda of the applicable ASME Codes will be incorporated into 10 CFR 50.55a in the future.

The NRC staff noted that the NRC Embark Venture Studios (Embark) recommended (ADAMS Accession No. ML20153A752) rulemaking to relax the requirement to update the ISI and IST programs every 10 years following the update to the 2019 or later edition of the ASME BPV Code and the 2020 or later edition of the ASME OM Code. The NRC staff asked if Exelon planned to update its current ISI and IST programs to support its proposed alternatives. Exelon stated that it planned to remain on the currently applicable ASME Code editions for each facility. Exelon stated it had considered creating a review process that would evaluate the need for potential program updates for the next interval, but this was not part of the proposed alternative presented. Exelon stated that the application could discuss its plan to update the ISI and IST programs after 24 years. However, the NRC staff noted that this would not be helpful since the update requirements in 10 CFR 50.55a would apply after the alternative had ended.

Exelon's presentation (Slide 6) provided a list of precedents that it had identified to support its proposed alternative. The NRC staff noted that the listed precedents relate to ISI and IST interval extensions or synchronization of these intervals. The NRC staff asked if Exelon had identified any precedents to support remaining on the same edition of the ASME Code for another interval. Exelon stated that they were not aware of any such precedents. The NRC staff also asked Exelon if it had identified any precedents for extending a future interval. Exelon stated that they were not aware of any such precedents.

The NRC staff described two relevant precedents which would not support Exelon's proposed fleet alternative to skip the next update to its ISI and IST programs. In October 1993 (ADAMS Accession No. ML20059A516), Entergy Operations, Inc. (Entergy) submitted a proposed alternative for its fleet to indefinitely remain on the ASME Code editions and addenda applicable at the time of its submittal. Subsequently, the NRC staff informed Entergy that its proposed alternative had generic implications that would be addressed through rulemaking and the review of the request was suspended. As discussed in SECY-00-0011, the Commission ultimately disapproved this rulemaking.

In January 1996 (ADAMS Accession No. ML20096B393), Entergy submitted a proposed alternative to the update requirements in 10 CFR 50.55a that would allow its plants to remain on the current code of record for IST for the next interval. At the time of its submittal, the required edition of the ASME BPV Code, Section XI, for the next interval was known. The NRC staff denied Entergy's January 1996 request due to insufficient plant-specific information (ADAMS Package Accession No. ML20107F554). During the meeting with Exelon, the staff stated that the NRC safety evaluation for this denial described the plant-specific information that licensees would need to provide to support future requests of this type. Specifically, licensees would need to provide a comparison between the current code of record and the latest edition of the ASME BPV Code, Section XI, incorporated by reference in 10 CFR 50.55a.

The NRC staff asked if Exelon had considered requesting an exemption. Exelon stated it had considered using an exemption, but an alternative request appeared to be easier. Specifically, Exelon indicated that it considered using the hardship provision in 10 CFR 50.12, "Specific exemptions." The NRC staff asked Exelon to explain the hardship and identify what has changed. Exelon noted that the process of updating its ISI and IST programs results in significant financial burden due to the need to update plant documents and perform training.

The NRC staff stated that it would review and respond, as appropriate, to any licensing action that Exelon submitted. However, the staff believes the changes that Exelon is seeking should be implemented through a revision of 10 CFR 50.55a. The reasons provided in the presentation do not appear to be unique to Exelon. Therefore, it would be beneficial to consider the views of the broader industry and other stakeholders regarding Exelon's proposed changes.

Additionally, the staff stated that Exelon should consider working through the ASME consensus committees to revise the ISI and IST intervals.

The NRC staff stated that it is in the process of obtaining the necessary alignment and approvals to pursue a rulemaking action based on the Embark recommendations. If approved, the staff intends to implement such a rule change as soon as feasible. To the extent allowed by NRC processes, the staff would keep stakeholders informed of the progress of any such rulemaking.

Two people provided comments during the public comment period. An official from the Pennsylvania Department of Environmental Quality asked if Exelon had received any significant violations related to ISI or IST. The NRC staff noted that it was not prepared to discuss specific violations at this meeting due to the large size of the Exelon fleet. The official also noted that plant safety should be preserved when considering the proposed alternative. An Exelon representative at the Nuclear Energy Institute stated that current levels of safety would be preserved under the proposed alternative.

Public meeting feedback forms were not received. Please direct any inquiries to me at 301-415-1380 or Blake.Purnell@nrc.gov.

December 10, 2020 – Letter from Blake Purnell, Project Manager Plant Licensing Branch III Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to EXELON GENERATION COMPANY, LLC with a subject of SUMMARY OF NOVEMBER 13, 2020, MEETING WITH EXELON GENERATION COMPANY, LLC REGARDING A PLANNED REQUEST FOR AN ALTERANTIVE TO CERTAIN DOCUMENTATION REQUIREMENTS FOR THE REPLACEMENT OF PRESSURE RETAINING BOLTING (EPID L-2020-LRM-0102)

On November 13, 2020, a Category 1 public meeting was held between the U.S. Nuclear Regulatory Commission (NRC or Commission) staff and representatives of Exelon Generation Company, LLC (Exelon, the licensee). The purpose of the meeting was to discuss a proposed alternative to certain documentation requirements for the replacement of pressure retaining bolting the subject facilities. The meeting notice and agenda are available in the Agencywide Documents Access and Management System (ADAMS) at Accession No. ML20304A229. A copy of Exelon's presentation is available in ADAMS at Accession No. ML20303A175. A list of attendees is enclosed.

The NRC regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a, "Codes and standards," include, in part, requirements for the use of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPV Code). The regulation in 10 CFR 50.55a(z) allows the NRC staff to approve proposed alternatives to the ASME BPV Code requirements if the licensee demonstrates that: (1) the proposed alternative would provide an acceptable level of quality and safety or (2) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Section XI of the ASME BPV Code includes, in part, documentation requirements for repair and replacement activities. As discussed in its presentation, Exelon has "observed that a major amount of resources are being applied to [repair and replacement] activities associated with routine bolting replacements which are not the result of inservice degradation." Exelon also stated that, although not required by regulation or the ASME BPV Code, repair and replacement plans are often prepared as a contingency for outage preparation. Exelon estimated that about 20 percent of these contingency plans are actually used.

Exelon stated that it planned to request the alternative under paragraph of 10 CFR 50.55a(z)(2) because the administrative burden of preparing the documentation for replacement of bolting is a hardship. The NRC staff noted that administrative burden is typically not considered to be a hardship under 10 CFR 50.55a(z)(2). Exelon indicated that they understood and would consider addressing the requirements of 10 CFR 50.55a(z)(1) instead.

As discussed in Slide 6 of Exelon's presentation, Exelon's proposed alternative would limit application of its repair and replacement program, as it relates to bolting, to bolting that is not currently associated with bolting categories subject to inservice inspection. The proposed alternative would apply to ASME Code Class 1, 2, and 3 components and would include safety- related equipment. Documentation would still occur through the normal processes of procurement, planning, and maintenance performed under Exelon's quality assurance program. The bolting material would be documented in work orders. Exelon is not proposing any changes to its quality assurance program in connection with this proposed alternative.

The NRC staff asked if Exelon considered working through ASME to change the requirements in Section XI of the ASME BPV Code. Exelon indicated that it has been working with ASME on a more general effort to change repair and replacement requirements. However, Exelon noted that working through ASME would take too long for this proposal.

Public meeting feedback forms were not received. Please direct any inquiries to me at 301-415-1380 or Blake.Purnell@nrc.gov.

December 10, 2020 – Letter from Robert J. Bernardo, Project Manager Integrated Program Management and Beyond-Design-Basis Branch Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to Bryan C. Hanson Senior Vice President Exelon Generation Company, LLC President and Chief Nuclear Officer Exelon Nuclear with subject of PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 – DOCUMENTATION OF THE COMPLETION OF REQUIRED ACTIONS TAKEN IN RESPONSE TO THE LESSONS LEARNED FROM THE FUKUSHIMA DAI-ICHI ACCIDENT

The purpose of this letter is to acknowledge and document that the actions required by the
U.S. Nuclear Regulatory Commission (NRC) in orders issued following the accident at the Fukushima Dai-ichi Nuclear Power Station have been completed for Peach Bottom Atomic Power Station, Units 2 and 3 (Peach Bottom). In addition, this letter acknowledges and documents that Exelon Generation Company, LLC (Exelon, the licensee), has provided the information requested in the NRC's March 12, 2012, request for information under Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.54(f), related to the lessons learned from that accident. Completing these actions and providing the requested information, in conjunction with the regulatory activities associated with the Mitigation of Beyond-Design-Basis Events (MBDBE) rulemaking, implements the safety enhancements mandated by the NRC based on the lessons learned from the accident. Relevant NRC, industry, and licensee documents are listed in the reference tables provided in the enclosure to this letter. The NRC will provide oversight of these safety enhancements through the Reactor Oversight Process (ROP).

BACKGROUND

In response to the events in Japan resulting from the Great Tōhoku Earthquake and subsequent tsunami on March 11, 2011, the NRC took immediate action to confirm the safety of U.S. nuclear power plants:

• On March 18, 2011, the NRC issued Information Notice 2011-05, "Tōhoku-Taiheiyou-Oki Earthquake Effects on Japanese Nuclear Power Plants" (Reference 1.1). The information notice was issued to inform U.S. operating power reactor licensees and applicants of the effects from the earthquake and tsunami. Recipients were expected to review the information for applicability to their facilities and consider actions, as appropriate. Suggestions contained in an information notice are not NRC requirements; therefore, no specific action or written response was required.

- • On March 23, 2011, the NRC issued Temporary Instruction (TI) 2515/183, "Followup to the Fukushima Dai-ichi Fuel Damage Event." The purpose of TI 2515/183 was to provide NRC inspectors with guidance on confirming the reliability of licensees' strategies intended to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities following events that may exceed the design basis for a plant. The results of the inspection for each licensee were documented in an inspection report (Reference 1.2).
- • On March 23, 2011, the Commission provided staff requirements memorandum (SRM) COMGBJ-11-0002, "NRC Actions Following the Events in Japan." The tasking memorandum directed the Executive Director for Operations to establish a senior level agency task force, referred to as the Near-Term Task Force (NTTF), to conduct a methodical and systematic review of the NRC processes and regulations to determine whether the agency should make additional improvements to the regulatory system and make recommendations to the Commission within 90 days for its policy direction (Reference 1.3).
- • On April 29, 2011, the NRC issued TI 2515/184, "Availability and Readiness Inspection of Severe Accident Management Guidelines (SAMGs)." The purpose of TI 2515/184 was to inspect the readiness of nuclear power plant operators to implement SAMGs. The results of the inspection were summarized and provided to the NTTF, as well as documented in

  a 2011 quarterly integrated inspection report for each licensee (Reference 1.4).

- • On May 11, 2011, the NRC issued Bulletin (BL) 2011-01, "Mitigating Strategies."
  BL 2011-01 required licensees to provide a comprehensive verification of their compliance with the regulatory requirements of 10 CFR 50.54(hh)(2), as well as provide information associated with the licensee's mitigation strategies under that section.
  In 10 CFR 50.54(hh)(2), it states, in part: "Each licensee shall develop and implement guidance and strategies intended to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities under the circumstances associated with loss of large areas of the plant due to explosions or fire... ." BL 2011-01 required a written response from each licensee (Reference 1.5). Note that the final MBDBE rule (Reference 1.15) relocated the requirements formerly in 10 CFR 50.54(hh)(2) to 10 CFR 50.155(b)(2).
- • On July 21, 2011, the NRC staff provided the NTTF report, "Recommendations for Enhancing Reactor Safety in the 21st Century: The Near-Term Task Force

Review of Insights from the Fukushima Dai-ichi Accident” to the Commission in SECY-11-0093, “Near-Term Report and Recommendations for Agency Actions Following the Events in Japan” (Reference 1.6).

- • On October 3, 2011, the staff prioritized the NTTF recommendations into three tiers in SECY-11-0137, “Prioritization of Recommended Actions to Be Taken in Response to Fukushima Lessons Learned.” The Commission approved the staff’s prioritization, with comment, in the SRM to SECY-11-0137 (Reference 1.7).

  A complete discussion of the prioritization of the recommendations from the NTTF report, additional issues that were addressed subsequent to the NTTF report, and the disposition of the issues that were prioritized as Tier 2 or Tier 3 is provided in SECY-17-0016, “Status of Implementation of Lessons Learned from Japan’s March 11, 2011, Great Tōhoku Earthquake and Subsequent Tsunami” (Reference 12.10).

A listing of the previous Commission status reports, which were provided semiannually, can be found in Table 12 in the enclosure to this letter.

The NRC undertook the following regulatory activities to address the majority of the Tier 1 recommendations:

- • On March 12, 2012, the NRC issued Orders EA-12-049, “Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events,” EA-12-050, “ Hardened Containment Vents,” and EA-12-051, “ Spent Fuel Pool Instrumentation,” and a request for information under 10 CFR 50.54(f) (hereafter referred to as the 50.54(f) letter) to licensees (References 1.8, 1.9, 1.10, and 1.11, respectively).
- • On June 6, 2013, the NRC issued Order EA-13-109, “Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions” (Reference 1.12), which superseded Order EA-12-050, replacing its requirements with modified requirements.
- • In addition to the three orders and the 50.54(f) letter, the NRC completed rulemaking, 10 CFR 50.155, “Mitigation of Beyond-Design-Basis Events,” that made generically applicable the requirements of Orders EA-12-049 and EA-12-051. The draft final rule and supporting documentation were provided to the Commission for approval in SECY-16-0142, “Draft Final Rule – Mitigation of Beyond-Design-Basis Events (RIN 3150-AJ49)” (Reference 1.13). The MBDBE rulemaking effort consolidated several of the recommendations from the NTTF report.

  On January 24, 2019, the Commission, via SRM-M190124A (Reference 1.14), approved the final MBDBE rule, with edits. The final rule approved by the Commission contains provisions that make generically applicable the requirements imposed by Orders EA-12-049 and EA-12-051 and supporting requirements. The Commission’s direction in the SRM makes it clear that the NRC will continue to follow a site-specific approach to resolve the interaction between the hazard reevaluation and mitigation strategies using information gathered in the 50.54(f) letter process. The NRC staff made conforming changes to the final rule package (Reference 1.15) as directed by the Commission, which included changes to two regulatory guides (References 1.16 and 1.17). The final rule was published in the *Federal Register* on August 9, 2019 (84 FR 39684), with an effective implementation date of September 9, 2019.

Subsequent to Commission approval of the final MBDBE rule, the staff engaged with stakeholders to pursue the expeditious closure of the remaining post-Fukushima 50.54(f) letter responses on a timeframe commensurate with each item's safety significance.

In a draft discussion paper (Reference 1.18) used to support a Category 3 public meeting held on February 28, 2019 (Reference 1.19), the NRC staff outlined the process to be used to review the reevaluated hazard and mitigation strategies assessment (MSA) information provided by licensees considering the differences between the draft final MBDBE rule and the approved final MBDBE rule. Subsequently, the NRC staff provided a screening letter (also called a "binning" letter) for both seismic and flooding hazard reevaluations (References 5.22 and 6.25), which categorized sites based on available information and the status of any commitments made in prior reports and assessments.

Order Modifying Licenses with Regard to Reliable

The process is discussed in greater detail in the "Hazard Reevaluation" and "Mitigation Strategies Assessment" sections of the discussion which follows.

This letter acknowledges and documents that the actions required by the NRC in response to the orders, as well as the information provided in response to the March 12, 2012, 50.54(f) letter, have been completed for Peach Bottom. However, the staff is not determining whether the licensee complies with the final MBDBE rule. Oversight of compliance with the final MBDBE rule at Peach Bottom will be conducted through the ROP.

DISCUSSION

Mitigation Strategies Order

Order EA-12-049, which applies to Peach Bottom, requires licensees to implement a three-phase approach for mitigation of beyond-design-basis external events (BDBEEs). It requires licensees to develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment, and spent fuel pool (SFP) cooling capabilities in the event of a BDBEE that results in a simultaneous loss of all alternating current (ac) power and loss of normal access to the ultimate heat sink (LUHS). Phases 1 and 2 of the order use onsite equipment, while Phase 3 requires obtaining sufficient offsite resources to sustain those functions indefinitely.

In August 2012, the Nuclear Energy Institute (NEI) issued Revision 0 of industry guidance document NEI 12-06, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," as guidance to comply with the order. The NRC endorsed the guidance in Revision 0 of Japan Lessons Learned Project Directorate (JLD) interim staff guidance (ISG) document JLD-ISG-2012-01, “Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events.” Subsequently, in December 2015, NEI issued Revision 2 of NEI 12-06 and the NRC endorsed that guidance in Revision 1 of JLD-ISG-2012-01 (Reference 2.1). Licensees were required to provide an overall integrated plan (OIP) to describe how they would comply with the order, along with status reports every 6 months until compliance was achieved (Reference 2.2). The NRC staff provided an interim staff evaluation (ISE) related to the OIP (Reference 2.3). The NRC concluded in the ISE that the licensee provided sufficient information to determine that there is reasonable assurance that the plan, when properly implemented, including satisfactory resolution of the open and confirmatory items, would meet the requirements of Order EA-12-049 at Peach Bottom. The NRC staff also conducted a regulatory audit of the licensee’s strategies and issued a report which documented the results of the audit activities (Reference 2.4). Upon reaching compliance with the order requirements, the licensee submitted a compliance letter and a final integrated plan (FIP) to the NRC (Reference 2.5). The FIP describes how the licensee is complying with the order at Peach Bottom.

The NRC staff completed a safety evaluation (SE) of the licensee’s FIP (Reference 2.6). The SE informed the licensee that its integrated plan, if implemented as described, provided a reasonable path for compliance with Order EA-12-049 at Peach Bottom. The staff then evaluated the implementation of the plans through inspection, using TI 2515/191, “Implementation of Mitigation Strategies and Spent Fuel Pool Instrumentation Orders and Emergency Preparedness Communications/Staffing/Multi-Unit Dose Assessment Plans.” An inspection report was issued to document the results of the TI 2515/191 inspection (Reference 2.7). The NRC will oversee implementation of the mitigation strategies requirements under the final MBDBE rule requirements through the ROP.

Phase 3 of Order EA-12-049 required licensees to obtain sufficient offsite resources to sustain the required functions indefinitely. There are two redundant National Strategic Alliance for FLEX Emergency Response (SAFER) Response Centers (NSRCs), one located in Memphis, Tennessee, and the other in Phoenix, Arizona, which have the procedures and plans in place to maintain and deliver the equipment needed for Phase 3 from either NSRC to any participating U.S. nuclear power plant when requested (Reference 2.8). The NRC staff evaluated and inspected the NSRCs and the SAFER program, plans, and procedures (References 2.9

and 2.10). Subsequently, SAFER provided two addenda to document the treatment of equipment withdrawn from the NSRCs (Reference 2.11). The NRC reviewed the addenda and documented its conclusion in an updated staff assessment (Reference 2.12). The NRC concluded that licensees may reference the SAFER program and implement their SAFER response plans to meet the Phase 3 requirements of the order. The licensee’s FIP
(Reference 2.5) includes the plans for utilizing the NSRC equipment at Peach Bottom. In its SE (Reference 2.6), the NRC staff concluded that the licensee has developed guidance that, if implemented appropriately, should allow utilization of offsite resources following a BDBEE consistent with NEI 12-06 guidance and should adequately address the requirements of the order.

Spent Fuel Pool Instrumentation Order

Order EA-12-051, which applies to Peach Bottom, required licensees to install reliable SFP level instrumentation with a primary channel and a backup channel, independent of each other, and with the capability to be powered independent of the plant's power distribution systems. The NEI issued NEI 12-02, "Industry Guidance for Compliance with NRC Order EA-12-051, 'To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation,'" as guidance to be used by licensees to comply with the order. The NRC endorsed this guidance in JLD-ISG-2012-03, "Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation" (Reference 3.1). Licensees were required to provide an OIP to describe how they would comply with the order, along with status reports every 6 months until compliance was achieved (Reference 3.2). The NRC issued an ISE, providing feedback on the OIP submittal (Reference 3.3). The NRC staff conducted a regulatory audit of the licensee's strategies and issued a report that documented the results of the audit activities

(Reference 3.4). Upon reaching compliance with the order requirements, the licensee submitted a compliance letter to the NRC (Reference 3.5), describing how the licensee complied with the order at Peach Bottom.

The NRC staff completed an SE of the actions taken by the licensee in response to the order (Reference 3.6). The SE informed the licensee that its integrated plan, if implemented as described, provided a reasonable path for compliance with Order EA-12-051 at Peach Bottom. The staff then evaluated the implementation of the plan through inspection, using TI 2515/191. An inspection report was issued to document the results of the TI 2515/191 inspection at the site (Reference 3.7). The NRC will oversee implementation of the SFP instrumentation requirements under the final MBDBE rule requirements through the ROP.

Reliable Hardened Containment Vent Order

Order EA-13-109 (Reference 1.12) is only applicable to operating boiling-water reactors (BWRs) with Mark I and Mark II containments. Because the reactors at Peach Bottom are General Electric BWRs with Mark I containments, this order is applicable to Peach Bottom.

Order EA-13-109 requires applicable licensees to implement its requirements in two phases. In Phase 1, licensees shall design and install a venting system that provides venting capability from the wetwell during severe accident conditions. In Phase 2, licensees shall either design and install a venting system that provides venting capability from the drywell under severe accident conditions, or develop and implement a reliable containment venting strategy that makes it unlikely that a licensee would need to vent from the containment drywell during severe accident conditions. Peach Bottom has elected the option to develop and implement a reliable containment venting strategy that makes it unlikely the licensee would need to vent from the containment drywell before alternate reliable containment heat removal and pressure control is reestablished.

In November 2013, NEI issued industry guidance document NEI 13-02, "Industry Guidance for Compliance with Order EA-13-109," as guidance to comply with Phase 1 of the order. The NRC endorsed the guidance in JLD-ISG-2013-02, "Compliance with Order EA-13-109, 'Order Modifying Licenses with Regard to Reliable Hardened

Containment Vents Capable of Performing under Severe Accident Conditions” (Reference 4.1).

In April 2015, NEI issued Revision 1 of industry guidance document NEI 13-02, "Industry Guidance for Compliance with Order EA-13-109," as guidance to comply with Phase 2 of the order. The NRC endorsed the guidance in JLD-ISG-2015-01, “Compliance with Phase 2 of Order EA-13-109, 'Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Performing under Severe Accident Conditions” (Reference 4.2).

Applicable licensees were required to provide an OIP to describe how they would comply with Phase 1 and Phase 2 of the order, along with status reports every 6 months until compliance was achieved (Reference 4.3). The NRC staff provided an ISE related to the OIP for both Phase 1 (Reference 4.4) and for Phase 2 (Reference 4.5). The NRC concluded in the ISEs that the licensee provided sufficient information to determine that there is reasonable assurance that the plan, when properly implemented, including satisfactory resolution of the open and confirmatory items, would meet the requirements of Order EA-13-109 at Peach Bottom. The NRC staff used a regulatory audit process to gain a better understanding of licensee activities as they came into compliance with the order. As part of this process, the staff reviewed the closeout of the ISE open items. The NRC issued an audit report to document the staff's understanding of the licensee's closeout of the ISE open items at the time of the audit (Reference 4.6). As noted in the audit report, the status of the NRC staff's review of the ISE open items could change as additional information is provided to the staff, or if the licensee changes its plans as part of final implementation. The final staff conclusions are documented in the SE.

Upon reaching compliance with the order requirements, the licensee submitted a compliance letter and a FIP to the NRC (Reference 4.7). The FIP describes how the licensee is complying with the order at Peach Bottom. The NRC staff documented its review of the FIP in an SE (Reference 4.8). The SE informed the licensee that its integrated plan, if implemented as described, provided a reasonable path for compliance with Order EA-13-109 at Peach Bottom. The staff then evaluated the implementation of the plans through inspection, using TI 2515/193, “Inspection of the Implementation of EA-13-109: Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions.” An inspection report was issued to document the results of the TI 2515/193 inspection (Reference 4.9). The NRC will oversee implementation of the containment venting requirements through the ROP.

Request for Information Under 10 CFR 50.54(f)

The 50.54(f) letter requested operating power reactor licensees to:

- • reevaluate the seismic and flooding hazards at their sites using present-day NRC requirements and guidance, and identify actions that are planned to address plant-specific vulnerabilities associated with the reevaluated seismic and flooding hazards
- • perform seismic and flooding walkdowns to verify compliance with the current licensing basis; verify the adequacy of current strategies and maintenance plans;

and identify degraded, nonconforming, or unanalyzed conditions related to seismic and flooding protection

- • provide an assessment of their current emergency communications and staffing capabilities to determine if any enhancements are needed to respond to a large-scale natural emergency event that results in an extended loss of ac power to all reactors at the site, and/or impeded access to the site

In COMSECY-14-0037, “Integration of Mitigating Strategies for Beyond-Design-Basis External Events and the Reevaluat[i]on of Flooding Hazards” (Reference 6.13), the NRC staff described issues related to the implementation of Order EA-12-049 and the related MBDBE rulemaking, and the completion of flooding reevaluations and assessments. In the SRM to COMSECY-14-0037 (Reference 6.14), the Commission directed the NRC staff to provide a plan for achieving closure of the flooding hazard assessments to the Commission for review and approval. The NRC staff provided this plan in COMSECY-15-0019, “Closure Plan for the Reevaluation of Flooding Hazards for Operating Nuclear Power Plants” (Reference 6.16), which the Commission approved in the SRM to COMSECY-15-0019 (Reference 6.17).

Hazard Reevaluations (Enclosures 1 and 2 of the 50.54(f) letter)

Each licensee followed a similar two-phase process to respond to the hazard reevaluations requested by the 50.54(f) letter. In Phase 1, licensees submitted hazard reevaluation reports using NRC-endorsed, industry-developed guidance. The guidance specified that a licensee should determine if interim protection measures were needed while a longer-term evaluation of the impacts of the hazard was completed. The NRC staff reviewed the reevaluated hazard information. Using the reevaluated hazard information and a graded approach, the NRC identified the need for, and prioritization and scope of, plant-specific assessments. For those plants that were required to perform a flooding integrated assessment (IA) or a seismic probabilistic risk assessment (SPRA), Phase 2 decisionmaking, as described by letters dated September 21, 2016, and March 2, 2020 (References 5.17 and 6.24), would determine whether additional plant-specific regulatory actions were necessary. In addition, as discussed in COMSECY-15-0019, most licensees performed an MSA to demonstrate that the licensee had adequately addressed the reevaluated hazards within their mitigation strategies developed for BDBEEs.

In a draft discussion paper (Reference 1.18) used to support a Category 3 public meeting held on February 28, 2019 (Reference 1.19), the NRC staff outlined the process to be used to review the reevaluated hazard and MSA information provided by licensees considering the differences

between the draft final MBDBE rule and the approved final MBDBE rule. The purpose of these reviews is to ensure that the conclusions in the various staff assessments continue to support a determination that no further regulatory actions are needed.

As stated in the discussion paper, the NRC subsequently issued a seismic screening letter (Reference 5.22) and a flooding screening letter (Reference 6.25), also called “binning” letters, to all operating power reactor licensees. The purpose of the binning letters is to categorize sites based on available information and the status of any commitments made in prior reports and assessments. Peach Bottom was binned as a Category 1 site for both seismic and flooding. Category 1 includes sites where no additional information or regulatory action is required. This category includes sites, such as Peach Bottom, where the licensee has previously demonstrated that existing seismic capacity or effective flood protection will address the unbounded reevaluated hazards.

Seismic Hazard Reevaluation (Enclosure 1 of the 50.54(f) letter)

Enclosure 1 of the 50.54(f) letter requested each operating power reactor licensee to complete a reevaluation of the seismic hazard that could affect their sites using updated seismic hazard information and present-day regulatory guidance and methodologies to develop a ground motion response spectrum (GMRS). The licensee was asked to compare their results to the safe-shutdown earthquake (SSE) ground motion and then report to the NRC in a seismic hazard screening report (SHSR). To provide a uniform and acceptable industry response, the Electric Power Research Institute (EPRI) developed a technical report, EPRI 1025287, “Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic,” and the NRC endorsed the guidance in a letter dated February 15, 2013 (Reference 5.1). From November 2012 to May 2014, the NRC and the industry provided guidance for the performance of the reevaluated hazard reviews

(References 5.2-5.7). The licensee provided a SHSR for Peach Bottom (Reference 5.8).

If the new GMRS was not bound by the current design basis (CDB) SSE, Enclosure 1 of the 50.54(f) letter requested more detailed evaluations of the impact from the hazard. Also, the licensee was asked to evaluate whether interim protection measures were needed while the more detailed evaluation was completed. By letter dated May 7, 2013, the NRC endorsed industry-developed guidance, a proposed path forward, and schedules, which were provided in a letter from NEI dated April 9, 2013. Attachment 1 of the NEI letter contains EPRI

Report 300200704, “Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic,” to provide the guidance needed to perform an evaluation of any needed interim protective measures (Reference 5.3). This expedited seismic evaluation process (ESEP) is a screening, evaluation, and equipment modification process performed by licensees to provide additional seismic margin and expedite plant safety enhancements for certain core cooling and containment components while the more detailed and comprehensive plant seismic risk evaluations are being performed. Peach Bottom screened in to complete an ESEP report (see References 5.10 and 5.11). The NRC staff completed a technical review of the ESEP report and documented its review in a response letter (Reference 5.13).

By letters dated May 9, 2014, and May 13, 2015 (Reference 5.10), the NRC informed licensees located in the Central and Eastern U.S. (CEUS) and Western U.S. (WUS), respectively, of the initial screening and prioritization results based on a review of the licensees’ SHSR. The NRC updated the screening and prioritization in a letter dated October 3, 2014 (Reference 5.11). The NRC provided the final determination of required seismic evaluations in a letter dated

October 27, 2015 (Reference 5.18). These evaluations could consist of an SPRA (Reference 5.1, SPID, Section 6.1.1), limited scope evaluations (High Frequency (Reference 5.14) and/or SFP evaluations (Reference 5.15)), or a relay chatter evaluation (Reference 5.4). If an SPRA was required, then additional Phase 2 regulatory decisionmaking was required (References 5.16 and 5.17).

The NRC staff completed and documented its review of the licensee's reevaluated seismic hazard in a staff assessment (Reference 5.9). In order to complete its response to the 50.54(f) letter, the licensee submitted an SFP evaluation and an SPRA report for Peach Bottom (Reference 5.19). As noted in the final screening letter, high frequency exceedances are addressed in the SPRA report. An audit was performed for each submittal (Reference 5.20). The audit results are documented in the applicable staff assessments (Reference 5.21). The NRC reviewed the SFP evaluation and confirmed that Peach Bottom met the criteria of the SFP Evaluation Guidance Report (Reference 5.15). The NRC reviewed the SPRA report using the regulatory review guidance provided in Reference 5.17. The staff's review concluded that the SPRA was of sufficient technical adequacy to support Phase 2 regulatory decisionmaking and that Peach Bottom responded appropriately to Enclosure 1, item (8) of the 50.54(f) letter (Reference 5.21). Based on the results and risk insights of the SPRA report and the SFP evaluation report, the NRC staff concluded that no further response or regulatory actions were required related to the seismic hazard reevaluation activities requested by Enclosure 1 of the 50.54(f) letter.

During an internal self-assessment review of the SPRA staff assessment, the staff uncovered an error in the spreadsheet used in the SPRA Screening Guidance to evaluate the Peach Bottom SPRA submittal. The correction of the error resulted in changes to certain numerical values that were documented in the staff's SPRA evaluation. A description of the error and corrected values for the affected portions of the staff evaluation are provided in an enclosure to an SPRA correction letter dated October 8, 2019 (Reference 5.21). The staff confirmed that the changes to the numerical values presented in the enclosure to the SPRA correction letter do not impact or change the NRC decision documented in the staff evaluation dated June 10, 2019.

Because the staff's reviews were completed prior to when the final MBDBE rule was approved, the NRC staff, using the process discussed in the seismic binning letter (Reference 5.22), re-visited these conclusions considering the final approved MBDBE rule. The staff confirmed that the conclusions in the various staff assessments continue to support a determination that no further regulatory actions are required for Peach Bottom.

The NRC staff reviewed the information provided and, as documented in the staff assessments (References 5.9 and 5.21), concluded that the licensee provided sufficient information in response to Enclosure 1 of the 50.54(f) letter. The staff acknowledges that all seismic hazard reevaluation activities requested by Enclosure 1 of the 50.54(f) letter have been completed for Peach Bottom. No further information related to the reevaluated seismic hazard is required.

Flooding Hazard Reevaluation (Enclosure 2 of the 50.54(f) letter)

Enclosure 2 of the 50.54(f) letter requested each operating power reactor licensee to complete a reevaluation of applicable flood-causing mechanisms at their site using updated flooding hazard information and present-day regulatory guidance and methodologies. Licensees were asked to compare their results to the CDB for protection and mitigation from external flood events. The NRC developed guidance to conduct the reevaluations (References 6.1 through 6.6). The licensee submitted a flood hazard reevaluation report (FHRR) for Peach Bottom (Reference 6.7)

to the NRC as requested by the 50.54(f) letter. Interim actions to protect against the reevaluated flood hazard were not needed. A regulatory audit to support the review of the FHRR was performed and the results documented in an audit report (Reference 6.8). The NRC staff reviewed the FHRR and provided an interim hazard letter (Reference 6.10) to provide feedback on the staff's review of the flooding hazard reevaluations. The interim hazard letter was used by the licensee to complete the flood hazard MSA and other flood hazard evaluations. Separately, the NRC staff documented the technical bases for its conclusions summarized in the interim hazard letters by issuing a detailed staff assessment (Reference 6.11).

In COMSECY-14-0037 (Reference 6.13), the NRC staff requested Commission direction to more clearly define the relationship between Order EA-12-049, the related MBDBE rulemaking, and the flood hazard reevaluations and assessments. Because the NRC was reevaluating its approach to the flooding evaluations, the NRC provided an extension of the due dates for any IAs in a letter dated November 21, 2014 (Reference 6.12). In the SRM to COMSECY-14-0037 (Reference 6.14), the Commission directed the NRC staff to provide a plan for achieving closure of the flooding portion of NTTF Recommendation 2.1 to the Commission for its review and approval. On May 26, 2015 (Reference 6.15), the NRC deferred, until further notice, the date for submitting the IA reports. On June 30, 2015 (Reference 6.16), the NRC staff provided a plan to the Commission in COMSECY-15-0019. On July 28, 2015 (Reference 6.17), the Commission approved the plan in the SRM to COMSECY-15-0019. On September 1, 2015, the NRC issued a letter to licensees describing the graded approach to complete the flood hazard reevaluations as approved by the Commission (Reference 6.18).

The COMSECY-15-0019 action plan required the NRC staff to develop a graded approach to identify the need for, and prioritization and scope of, plant-specific IAs and evaluation of plant-specific regulatory actions. The NRC staff's graded approach enabled a site with hazard exceedance above its CDB to demonstrate the site's ability to cope with the reevaluated hazard through appropriate protection or mitigation measures which are timely, effective, and reasonable. The IAs were focused on sites with the greatest potential for additional safety enhancements. New guidance for performing the IAs and focused evaluations (FEs) was developed for this graded approach. The guidance also provided schedule information for submission of any required IA. On July 18, 2016 (Reference 6.19), the staff issued JLD-ISG-2016-01, "Guidance for Activities Related to Near-Term Task Force Recommendation 2.1, Flooding Hazard Reevaluation, Focused Evaluation and Integrated Assessment". The ISG provided the guidance for Phase 1 flooding assessments, as described in COMSECY-15-0019, and endorsed industry guidance provided in NEI 16-05, "External Flooding Integrated Assessment Guidelines" (Reference 6.19). If an IA was necessary, then Phase 2 regulatory decisionmaking was required (References 6.23 and 6.24).

As noted in the interim hazard response letter (Reference 6.10), the storm surge, seiche, ice-induced flooding, and local intense precipitation (LIP) flood-causing mechanisms at Peach Bottom were not bounded by the CDB. Therefore, additional assessments of these flood-causing mechanisms were required. The NRC staff used a graded approach to determine if this site would need to perform an IA for the reevaluated flooding hazard, or if an FE would suffice. Based on the graded approach, Peach Bottom completed an FE (Reference 6.20) to ensure appropriate actions were identified and taken to protect the plant from the reevaluated flood hazard. The NRC staff conducted a regulatory audit (Reference 6.22), completed its review of the FE, and concluded in the staff assessment (Reference 6.21) that the licensee provided sufficient information in response to the 50.54(f) letter. Audit results were summarized in the staff assessment. No further regulatory actions are required related to the flood hazard reevaluations.

Because the staff's reviews were completed prior to when the final MBDBE rule was approved, the NRC staff, using the process discussed in the flooding binning letter (Reference 6.25), re-visited these conclusions considering the final approved MBDBE rule. The staff confirmed that the conclusions in the various staff assessments continue to support a determination that no further regulatory requirements are required for Peach Bottom.

The NRC staff reviewed the information provided by the licensee and has concluded that sufficient information was provided to be responsive to Enclosure 2 of the 50.54(f) letter. The staff acknowledges that all flooding hazard reevaluation activities requested by Enclosure 2 of the 50.54(f) letter have been completed for Peach Bottom. No further information related to the reevaluated flood hazard is required.

Mitigating Strategies Assessment

In addition to the closure plan for NTTF Recommendation 2.1, the action plan approved by the Commission in the SRM to COMSECY-15-0019 (Reference 7.4) identified the NRC staff's efforts to ensure licensees would address the reevaluated hazard information in their mitigation strategies. Proposed requirements related to the MSA were included in the draft final MBDBE rule, but were removed as a requirement from the final approved rule language. The Commission's direction in SRM-M190124A (Reference 1.14) makes it clear that the NRC will continue to follow a site-specific approach to resolve the interactions between the hazard reevaluation and mitigation strategies using information gathered in the 50.54(f) letter process.

In a draft discussion paper (Reference 1.18) used to support a Category 3 public meeting held on February 28, 2019 (Reference 1.19), the NRC staff outlined the process to be used to review the reevaluated hazard and MSA information provided by licensees considering the differences between the draft final MBDBE rule and the approved final MBDBE rule. Subsequently, the NRC staff provided a screening letter (also called a "binning" letter) for both seismic and flooding information (References 5.22 and 6.25), which categorized sites based on available information and the status of any commitments made in prior reports and assessments. The majority of MSAs had been submitted and evaluated by the staff prior to the issuance of the binning letters. For the MSA reviews that had not yet been completed, or MSAs that had not yet been submitted, the staff would evaluate the hazard impacts on the mitigation strategies, as appropriate, as part of its review of SPRA reports, flooding FEs, and/or flooding IAs.

The objective of the MSA is to determine whether the mitigation strategies developed for Order EA-12-049 can still be implemented given the reevaluated hazard levels. If it was determined that the mitigation strategies could not be implemented for the reevaluated hazard levels, the MSA could provide other options such as performing additional evaluations, modifying existing mitigating strategies, or developing alternate mitigating strategies or targeted hazard mitigating strategies to address the reevaluated hazard levels. In Revision 1 to JLD-ISG-2012-01, the NRC endorsed industry-developed guidance contained in Appendices G and H of Revision 2 to NEI 12-06 (Reference 7.5) for completing the MSAs. In Revision 2 to JLD-ISG-2012-01, the NRC endorsed the industry-developed guidance of NEI 12-06,

Revision 4 (Reference 7.5). Revision 4 of NEI 12-06, among other changes, provides additional guidance in Section H.4.5 for the performance of seismic MSAs for plants with reevaluated seismic hazard information that includes a GMRS that has spectral ordinates greater than twice the plant's SSE anywhere in the frequency range of 1 to 10 Hertz. Peach Bottom used the guidance in Section H.4.5 to complete the seismic MSA.

The licensee completed both a flood hazard MSA (Reference 7.6) and a seismic hazard MSA (Reference 7.8) for Peach Bottom. A generic regulatory audit plan (Reference 7.10) was issued to support the reviews of the seismic and flooding MSAs. As necessary, the site-specific audit results are documented in the applicable staff assessment.

The NRC staff reviewed the flooding MSA submittal and issued a staff assessment (Reference 7.7) documenting its review. The NRC staff concluded that the licensee has demonstrated that the mitigation strategies appropriately address the reevaluated hazard conditions. As discussed in the flooding binning letters (Reference 6.25), the staff re-visited this conclusion considering the final approved MBDBE rule. The staff confirmed that the conclusions in the flooding MSA staff assessment continues to support a determination that no further regulatory actions are required.

As noted in the seismic hazard binning letter (Reference 5.22), the staff suspended its review of certain seismic MSA submittals, including the MSA for Peach Bottom. For the reviews not yet completed (such as Peach Bottom), the staff evaluated the mitigation strategies as part of its review of the SPRA report (Reference 5.21). Based on the results and risk insights of the SPRA report, combined with the results of the SFP evaluation (Reference 5.21), the NRC staff concluded that no further response or regulatory actions were required related to the seismic hazard reevaluation activities requested by Enclosure 1 of the 50.54(f) letter.

Walkdowns (Enclosures 3 and 4 of the 50.54(f) letter)

Enclosures 3 and 4 of the 50.54(f) letter requested that licensees perform plant walkdowns to verify compliance with the current licensing basis as it pertains to seismic and flood protection. By letter dated May 31, 2012 (Reference 8.2), the NRC endorsed industry-developed guidance contained in Technical Report EPRI 1025286, "Seismic Walkdown Guidance" (Reference 8.1), for the performance of the seismic walkdowns. By letter dated May 31, 2012 (Reference 9.2), the NRC endorsed industry-developed guidance contained in NEI 12-07, "Guidelines for Performing Verification Walkdowns of Plant Flood Protection Features" (Reference 9.1), for performance of the flooding walkdowns. The licensee provided a report for both the seismic and flooding walkdowns at Peach Bottom (References 8.3 and 9.3). The NRC performed onsite inspections per TI 2515/188, “Inspection of Near-Term Task Force Recommendation 2.3 Seismic Walkdowns,” and TI 2515/187, “Inspection of Near-Term Task Force

Recommendation 2.3 Flooding Walkdowns,” and documented the inspection results in a quarterly integrated inspection report (References 8.4 and 9.4). The NRC staff issued staff assessments for both the seismic and flooding walkdowns (References 8.6 and 9.5). Because there were inaccessible/restricted access items identified during the initial licensee walkdowns, the licensee submitted subsequent walkdowns report after accessing the areas

(References 8.3, 8.5, and 9.6). The NRC documented its review of the subsequent walkdown reports in the staff assessment (Reference 8.6) and a memo dated September 25, 2015 (Reference 8.7).

The NRC staff reviewed the information provided by the licensee and determined that sufficient information was provided to be responsive to Enclosures 3 and 4 of the 50.54(f) letter. The staff acknowledges that all seismic and flooding walkdown activities requested by the 50.54(f) letter have been completed for Peach Bottom.

Communications and Staffing (Enclosure 5 of the 50.54(f) letter)

Enclosure 5 of the 50.54(f) letter requested licensees to assess their means to power equipment needed to communicate onsite and offsite during a prolonged station blackout event and to identify and implement enhancements to ensure that communications can be maintained during such an event. Also, licensees were requested to assess the staffing required to fill all necessary positions to respond to a multiunit event with impeded access to the site, or to an extended loss of all ac power for single unit sites. Licensees were requested to submit a written response to the information requests within 90 days or provide a response within 60 days and describe an alternative course of action and estimated completion dates. The licensee proposed an alternative course of action and schedule for Peach Bottom (Reference 10.2), which included a 90-day partial response (Reference 10.3). The NRC acknowledged the schedule changes in a letter dated July 26, 2012 (Reference 10.4).

The NRC endorsed industry-developed guidance contained in NEI 12-01, “Guideline for Assessing Beyond-Design-Basis Accident Response Staffing and Communications Capabilities” in a letter dated May 15, 2012 (Reference 10.1), for the performance of the communications and staffing assessments. The licensee provided the communications assessment and implementation schedule for Peach Bottom (Reference 10.5), and the NRC completed a staff assessment of the licensee’s communications assessment (Reference 10.6).

Licensees responded to the staffing portion of the 50.54(f) letter in two phases to account for the implementation of mitigation strategies. Phase 1 staffing assessments were based on the existing station blackout coping strategies with an assumption of all reactors at the site being affected concurrently. The Phase 1 staffing assessment is required for multiunit sites and was completed for Peach Bottom (Reference 10.7). In Phase 2, all licensees assessed the staffing necessary to carry out the mitigation strategies (Reference 10.9). The NRC staff issued staffing assessment response letters (References 10.8 and 10.10) for each submittal. The NRC performed an onsite inspection using TI 2515/191 to verify that the emergency communications and staffing plans at Peach Bottom have been implemented as described by the licensee (Reference 10.11).

Proposed Regulatory Guide 1.228 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16218A236) was expected to endorse, with clarifications,
NEI 12-01, NEI 13-06, “Enhancements to Emergency Response Capabilities for Beyond-Design-Basis Events and Severe Accidents” (Reference 11.16), and NEI 14-01, “Emergency Response Procedures and Guidelines for Beyond-Design-Basis Events and Severe Accidents” (Reference 11.7). However, the final MBDBE rule’s language was revised to remove these requirements from the rule. The NRC staff canceled proposed Regulatory Guide 1.228 to reflect the approved changes in the final rule. The NRC will oversee the licensee’s implementation of communications and staffing plans which support the mitigation strategies requirements through the ROP.

The NRC staff reviewed the information provided by the licensee and determined that sufficient information was provided to be responsive to Enclosure 5 of the 50.54(f) letter. The staff acknowledges that all emergency preparedness communications and staffing activities requested by Enclosure 5 of the 50.54(f) letter have been completed for Peach Bottom. No further information related to the communications and staffing assessments is required.

Additional Industry Commitments

Update and Maintain Severe Accident Management Guidelines

The NRC staff provided the proposed MBDBE rule to the Commission on April 30, 2015 (Reference 11.1), in SECY-15-0065, “Proposed Rulemaking: Mitigation of Beyond-Design- Basis Events (RIN 3150-AJ49)” and the Commission issued the SRM to SECY-15-0065 on August 27, 2015 (Reference 11.2). The Commission approved publication of the proposed rule subject to removal of the proposed requirements pertaining to the SAMGs. The Commission also directed the staff to update the ROP to explicitly provide periodic oversight of industry’s implementation of the SAMGs.

By letter dated October 26, 2015 (Reference 11.3), NEI described the industry initiative, approved by the Nuclear Strategic Issues Advisory Committee as mandatory for all NEI members, to update and maintain the SAMGs. Specifically, each licensee will perform timely updates of their site-specific SAMGs based on revisions to generic severe accident technical guidelines. Licensees will also ensure that SAMGs are considered within plant configuration management processes. As noted in the NEI letter, the licensee provided a letter (Reference 11.4) to establish a site-specific regulatory commitment for Peach Bottom.

In a letter to NEI dated February 23, 2016 (Reference 11.5), the staff outlined its approach for making changes to the ROP in accordance with the Commission direction. The staff engaged NEI and other stakeholders to identify the near-term and long-term changes to the ROP, consistent with the Commission direction and the licensees’ near-term and long-term SAMG commitments. In November 2016, the staff revised Inspection Procedure (IP) 71111.18, “Plant Modifications” (Reference 11.6, effective January 1, 2017), to provide oversight of the initial inclusion of SAMGs within the plant configuration management processes to ensure that the SAMGs reflect changes to the facility over time. In November 2018, the staff published a revision to IP 71111.18 (Reference 11.6, effective January 1, 2019), to provide oversight of the site-specific incorporation of generic owner’s groups SAMG guidance revisions.

Multiunit/Multisource Dose Assessments

In COMSECY-13-0010, “Schedule and Plans for Tier 2 Order on Emergency Preparedness for Japan Lessons Learned,” dated March 27, 2013 (Reference 11.13), the NRC staff requested Commission approval to implement the NTTF recommendation concerning multiunit/multisource dose assessments by having licensees document their commitment to obtain multiunit/multisource dose assessment capability by the end of 2014, rather than by issuing an order. Multiunit dose assessment capabilities would be made generically applicable through subsequent rulemaking. The Commission approved the staff’s requests in the SRM to COMSECY-13-0010, dated April 30, 2013 (Reference 11.14). The licensee commitments are documented in References 11.8 through 11.11.

The NRC staff included the multiunit/multisource dose assessment requirement in the proposed MBDBE rulemaking (Reference 11.1). However, in response to a public comment concerning the 10 CFR 50.109 backfitting justification for the proposed multiple source term dose assessment requirements, the NRC staff determined that this requirement did not meet the criteria for imposition under 10 CFR 50.109(a)(4)(ii). The NRC staff also concluded that this could not be justified as a compliance backfit or as a substantial safety improvement whose costs, both direct and indirect, would be justified considering the potential safety gain. Therefore, these requirements were removed from the draft final rule (Reference 1.13).

The licensee provided the requested information and stated that Peach Bottom will have multiunit/multisource dose assessment capabilities (Reference 11.11) by December 31, 2014. The NRC acknowledged the licensee’s submittal (Reference 11.12), verified the implementation of these dose assessment capabilities through inspection per TI 2515/191, and issued an inspection report (Reference 11.15).

CONCLUSION

The NRC staff concludes that Exelon, the licensee, has implemented the NRC-mandated safety enhancements resulting from the lessons learned from the Fukushima Dai-ichi accident through its implementation of Orders EA-12-049, EA-12-051, and EA-13-109 at Peach Bottom. The staff further concludes that the licensee has completed its response to the 50.54(f) letter for Peach Bottom. No further regulatory decisionmaking is required for Peach Bottom related to the Fukushima lessons-learned.

A listing of the applicable correspondence related to the Fukushima lessons-learned activities for Peach Bottom is included as an enclosure to this letter.

If you have any questions, please contact me at 301-415-2621 or by e-mail at Robert.Bernardo@nrc.gov.

Docket Nos. 50-277 and 50-278

Enclosure:
Documents Related to Required

Response
cc w/encl: Distribution via Listserv

Sincerely,

/RA/

Robert J. Bernardo, Project Manager Integrated Program Management

and Beyond-Design-Basis Branch Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation

Peach Bottom Atomic Power Station, Units 2 and 3

Reference Documents Related to Required Response to the Lessons Learned from the Fukushima Dai-ichi Accident

TABLE 1
Initial Actions in Response to the Events in Japan Caused by the Great Tōhoku Earthquake and Subsequent Tsunami

Ref	Document	Date	ADAMS[1] Accession No.

1. 1.1 NRC Information Notice 2011-05
2. 1.2 NRC Followup to the Fukushima Dai-ichi Fuel

   Damage Event
   Temporary Instruction (TI) 2515/183
   NRC TI 2515/183 Inspection Report 2011- 009
   NRC Integrated Inspection Report 2011- 003 (TI 2515/183 closeout)
   Summary of Observations – TI-183

3. 1.3 NRC Tasking Memorandum, Staff Requirements Memorandum (SRM) to COMGBJ-11-0002
4. 1.4 NRC Availability and Readiness Inspection of SAMG

   NRC Availability and Readiness Inspection of SAMG - TI 2515/184
   NRC Integrated Inspection Report 2011- 003 (TI 2515/184 inspection)
   NRC TI 2515/184 Inspection Results, Region 1 Summary
   NRC Summary of TI 2515/184 Results

5. 1.5 NRC Bulletin 2011-01, “Mitigating Strategies” NRC Bulletin 2011-01

Licensee 30 day response to BL 2011-01 Licensee 60 day response to BL 2011-01 NRC Request for Additional Information (RAI) regarding Licensee 60 day response to BL 2011-01

Licensee response to RAI
NRC Closeout of BL 2011-01 for the Exelon Fleet

6. 1.6 NRC NTTF Report (SECY-11-0093)
7. 1.7 NRC SECY-11-0137, Prioritization of

Recommended Actions to Be Taken in Response to Fukushima Lessons Learned

NRC SECY-11-0137

SRM-SECY-11-0137

8. 1.8 NRC Order EA-12-049
9. 1.9 NRC Order EA-12-050
10. 1.10 NRC Order EA-12-051

March 18, 2011

March 23, 2011 May 13, 2011

August 3, 2011

November 28, 2011 March 23, 2011

April 29, 2011

August 3, 2011

May 27, 2011

June 6, 2011

May 11, 2011
June 8, 2011
July 8, 2011 November 22, 2011

December 20, 2011 August 2, 2012

July 12, 2011

October 3, 2011 December 15, 2011 March 12, 2012 March 12, 2012 March 12, 2012

ML110760432

ML11077A007 ML111300540

ML112150590

ML11325A020 ML110820875

ML11115A053 ML112150590 ML111470361 ML11154A109

ML111250360 ML111600096 ML111920162 ML113120057

ML113550139 ML12178A215

ML11186A950

ML11272A111 ML113490055 ML12054A735 ML12054A694 ML12054A679

[1] Agencywide Documents Access and Management System (ADAMS)

Enclosure

Peach Bottom Atomic Power Station, Units 2 and 3

TABLE 1
Initial Actions in Response to the Events in Japan Caused by the Great Tōhoku Earthquake and Subsequent Tsunami

Ref	Document	Date	ADAMS[1] Accession No.

11. 1.11 NRC Request for Information Under 10 CFR 50.54(f) (the 50.54(f) letter)
12. 1.12 NRC Order EA-13-109
13. 1.13 NRC SECY-16-0142, “Draft Final Rule:

    Mitigation of Beyond-Design-Basis Events”

15. 1.15 Final Rule: Mitigation of Beyond-Design- Basis Events (Package)
16. 1.16 Regulatory Guide 1.226, Revision 0, Flexible Mitigation Strategies for Beyond-Design-Basis Events
17. 1.17 Regulatory Guide 1.227, Revision 0, Wide Range Spent Fuel Pool Level Instrumentation
18. 1.18 NRC Staff Preliminary Process for Treatment of Reevaluated Seismic and Flooding Hazard Information in Backfit Determinations

March 12, 2012

June 6, 2013 December 15, 2016

August 9, 2019 June 30, 2019

June 30, 2019 February 14, 2019

ML12053A340

ML13143A321 ML16301A005

ML19058A006 ML19058A012

ML19058A013 ML19037A443

1.14	SRM-M190124A: Affirmation Session-SECY- 16-0142: Final Rule: Mitigation of Beyond- Design-Basis Events (RIN 3150-AJ49) - Package	January 24, 2019	ML19023A038

1.19	Category 3 Public Meeting to Discuss Staff's Preliminary Process for Treatment of Reevaluated Seismic and Flooding Hazard Information in Backfit Determinations	February 14, 2019	ML19052A511

TABLE 2
Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events – EA-12-049

Ref

Document

Date

ADAMS Accession No.

1. 2.1 Guidance for Compliance with EA-12-049 - Diverse and Flexible Coping Strategies (FLEX)

   Industry Guidance on Diverse and Flexible Coping Strategies (FLEX) NEI 12-06, Revision 0
   NRC endorsement of NEI 12-06, Revision 0 - JLD-ISG-2012-01, Revision 0

   Industry Guidance on Diverse and Flexible Coping Strategies (FLEX) NEI 12-06, Revision 2
   NRC endorsement of NEI 12-06, Revision 2 - JLD-ISG-2012-01, Revision 1

2. 2.2 Licensee Overall Integrated Plan (OIP) Licensee OIP submittal

   OIP 1st six month status report OIP 2nd six month status report OIP 3rd six month status report OIP 4th six month status report OIP 5th six month status report OIP 6th six month status report OIP 7th six month status report OIP 8th six month status report OIP 9th six month status report

3. 2.3 NRC Interim Staff Evaluation (ISE) of OIP
4. 2.4 NRC audit of EA-12-049 OIP

   NRC Notification of Audit of EA-12-049 NRC Site-Specific Audit Plan
   NRC Audit Report

5. 2.5 Licensee Compliance Letter for EA-12-049 and Final Integrated Plan (FIP)

   Licensee Compliance Letter for EA-12- 049 for Unit 2
   Licensee Compliance Letter for EA-12-049 for Unit 3 and FIP for Units 2 and 3

6. 2.6 NRC Safety Evaluation (SE) of Implementation of EA-12-049
7. 2.7 NRC Inspection of Licensee Responses to EA-12-049, EA-12-051, and Emergency Preparedness Information

   NRC TI 2515/191, Revision 2
   NRC TI 2515/191 Inspection Report 2018- 012

August 21, 2012

August 29, 2012 December 2015

January 22, 2016 February 28, 2013

August 28, 2013 February 28, 2014 August 28, 2014 February 27, 2015 August 28, 2015 February 26, 2016 August 26, 2016 February 28, 2017 August 28, 2017 November 22, 2013

August 28, 2013 May 6, 2015 September 23, 2015

January 6, 2017 January 5, 2018

May 9, 2018

July 10, 2018 March 5, 2019

ML12242A378

ML12229A174 ML16005A625

ML15357A163

ML13059A305 ML13246A412 ML14059A222 ML14241A252 ML15058A263 ML15245A364 ML16057A009 ML16239A293 ML17059D132 ML17240A029 ML13220A105

ML13234A503 ML15119A292 ML15254A135

ML17006A167 ML18005A701

ML18113A334

ML18191B074 ML19065A010

TABLE 2
Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events – EA-12-049

Ref	Document	Date	ADAMS Accession No.

8. 2.8 Industry White Paper – National SAFER Response Centers (NSRC)
9. 2.9 NRC Staff Assessment of NSRCs
10. 2.10 NRC Inspection of Implementation of EA-12-049 Regarding the use of NSRC

    NRC Inspection Procedure (IP) 43006

    NRC Vendor Inspection of the Phoenix NSRC Report No. 99901013/2016-201
    NRC Vendor Inspection of the Memphis NSRC Report No. 99901013/2017-201

11. 2.11 Addenda I and II to industry NSRC white paper
12. 2.12 NRC Updated Staff Assessment of NSRCs

NA NRC approval of relaxation request of the schedule requirements for Order EA-12-049 for Unit 3 to align with schedule for EA-13-109

September 11, 2014 September 26, 2014

September 30, 2016
January 12, 2017

May 5, 2017 May 24, 2018

September 20, 2018
April 15, 2014

ML14259A221 ML14265A107

ML16273A318 ML17012A186 ML17117A576 ML18150A658 ML18157A014

ML14071A606

Ref

Document

1. 3.1 Guidance for Compliance with EA-12-051 – Spent Fuel Pool Instrumentation (SFPI)

   Industry Guidance for Compliance with EA-12-051 – NEI 12-02, Revision 1
   NRC endorsement of NEI 12-02, Revision 1 - JLD-ISG-2012-03, Revision 0

2. 3.2 Licensee OIP Licensee OIP

   OIP 1st six month status report OIP 2nd six month status report OIP 3rd six month status report OIP 4th six month status report OIP 5th six month status report

3. 3.3 NRC ISE of OIP
4. 3.4 NRC Audit of EA-12-051

   NRC Notification of Audit of EA-12-051 NRC Audit Report of Westinghouse SFPI design specifications
   NRC Site-Specific Audit Plan

   NRC Audit Report

5. 3.5 Licensee Compliance Letter for EA-12-051
6. 3.6 NRC SE of Implementation of EA-12-051
7. 3.7 NRC Inspection of Licensee Responses to

   EA-12-049, EA-12-051, and Emergency Preparedness Information

   NRC TI 2515/191, Revision 2
   NRC TI 2515/191 Inspection Report 2018- 012

August 2012 August 29, 2012

February 28, 2013 August 28, 2013 February 28, 2014 August 28, 2014 February 27, 2015 August 28, 2015 October 30, 2013

March 26, 2014 August 18, 2014

May 6, 2015 September 23, 2015
December 15, 2015

July 10, 2018 March 5, 2019

ML12240A307 ML12221A339

ML13059A390 ML13241A039 ML14059A227 ML14241A303 ML15058A254 ML15243A099 ML13295A303

ML14083A620 ML14211A346

ML15119A292 ML15254A135

ML15352A135

ML18191B074 ML19065A010

Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation – EA-12-051

Date

ADAMS Accession No.

1. 4.1 Guidance for Compliance with Phase 1 of EA-13-109 – Severe Accident Capable Hardened Containment Vent System (HCVS)

   Industry Guidance for Compliance with EA-13-109 – NEI 13-02, Revision 0
   NRC endorsement of NEI 13-02, Revision 0 - JLD-ISG-2013-02

2. 4.2 Guidance for Compliance with Phase 2 of EA-13-109 – Severe Accident Capable HCVS

   Industry Guidance for Compliance with EA-13-109 - NEI 13-02, Revision 1
   NRC endorsement of NEI 13-02, Revision 1 - JLD-ISG-2015-01

3. 4.3 Licensee Overall Integrated Plan (OIP) Licensee Phase 1 OIP

   OIP 1st six month status report
   OIP 2nd six month status report
   OIP 3rd six month status report - Phase 1 OIP (updated) and Phase 2 OIP submittal OIP 4th six month status report
   OIP 5th six month status report
   OIP 6th six month status report
   OIP 7th six month status report
   OIP 8th six month status report

4. 4.4 NRC ISE of Phase 1 OIP
5. 4.5 NRC ISE of Phase 2 OIP
6. 4.6 NRC Audit Activities related to EA-13-109

   NRC Notification of Audit of Phase 1 of EA-13-109
   NRC Notification of Audit of Phase 2 of EA-13-109

   NRC Audit Report

7. 4.7 Licensee Compliance Letter and FIP for

EA-13-109
Compliance Letter for Unit 2 and FIP for Units 2 and 3
Compliance Letter for Unit 3

8. 4.8 NRC SE of Implementation of EA-13-109
9. 4.9 NRC Inspection of Licensee Responses to

EA-13-109
NRC TI 2515/193, Revision 1
NRC TI 2515/193 Inspection Report 2020- 012

November 12, November 14,

April 23, 2015 April 29, 2015

June 30, 2014 December 19, June 30, 2015 December 15,

June 30, 2016 December 15, June 30, 2017 December 15, June 29, 2018 February 12, 2015 August 2, 2016

May 27, 2014 August 10, 2017 November 30, 2017

September 28, 2018
January 5, 2018 March 11, 2019

July 1, 2020 September 9, 2020

Peach Bottom Atomic Power Station, Units 2 and 3

TABLE 4
Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions – EA-13-109

Ref

Document

Date

ADAMS Accession No.

2013 2013

ML13316A853 ML13304B836

ML15113B318 ML15104A118

ML14181A301 ML14353A125 ML15181A018 ML15364A015

ML16182A012 ML16350A265 ML17181A034 ML17349A038 ML18180A032
ML15026A469 ML16099A272

ML14126A545 ML17220A328 ML17328A163

ML18271A008

ML18005A636 ML19015A422

ML19318D506 ML20253A254

2014 2015

2016 2017

TABLE 5
Request for Information Under Title 10 of the *Code of Federal Regulations*, Section 50.54(f), Enclosure 1: Recommendation 2.1 Seismic Hazard Reevaluation

Ref

Document

Date

ADAMS Accession No.

Guidance Documents

1. 5.1 Screening, Prioritization and Implementation Details (SPID)

   Industry Guidance (SPID) – EPRI 1025287
   NRC letter endorsing SPID

2. 5.2 NRC guidance for performing a Seismic Margin Assessment (SMA) –

   JLD-ISG-2012-04

3. 5.3 Expedited Seismic Evaluation Process

   (ESEP)
   Industry Letter – Proposed path forward for NTTF Recommendation 2.1: Seismic
   Industry Guidance (ESEP) - EPRI 3002000704
   NRC letter endorsing the ESEP approach. Extension of ESEP due date to
   3/31/14 for Central and Eastern U.S. (CEUS) sites

4. 5.4 Industry letter on relay chatter review
5. 5.5 NRC letter with guidance on the content of

seismic reevaluation submittals (includes

operability and reportability discussions)

6. 5.6 Industry letter on seismic risk evaluations for

CEUS plants

7. 5.7 NRC background paper - Probabilistic seismic

hazard analysis

Seismic Hazard Screening Report (SHSR)

8. 5.8 Licensee SHSR
1.5 year partial response

SHSR

9. 5.9 NRC Staff Assessment of Reevaluated

Seismic Hazard Information

Screening and Prioritization Results

10. 5.10 NRC Letter - Seismic screening and prioritization results

Central and Eastern US (CEUS) plants Western US (WUS) plants
Screening based on IPEEE results

11. 5.11 NRC Letter – Updated seismic screening and prioritization results

November 2012

February 15, 2013 November 16, 2012

April 9, 2013 April 2013 May 7, 2013

October 3, 2013 February 20, 2014

March 12, 2014 May 20, 2014

September 12, 2013
March 31, 2014 April 20, 2015

May 9, 2014
May 13, 2015 November 21, 2014 October 3, 2014

ML12333A170

ML12319A074 ML12286A029

ML13101A345 ML13102A142 ML13106A331

ML13281A308 ML14030A046

ML14083A596 ML14140A648

ML13256A070

ML14090A247 ML15051A262

ML14111A147 ML15113B344 ML14246A428 ML14258A043

5.12	NRC letter regarding development of Seismic Risk Evaluations – suitability of updated seismic hazard information for further assessments	December 10, 2014	ML14307B707

TABLE 5
Request for Information Under Title 10 of the *Code of Federal Regulations*, Section 50.54(f), Enclosure 1: Recommendation 2.1 Seismic Hazard Reevaluation

Ref

Document

Date

ADAMS Accession No.

5.13 ESEP Submittal and Evaluation Licensee ESEP Submittal

NRC Response Letter for the ESEP

Submittal

Additional Guidance Documents

14. 5.14 High Frequency Program Application Guidance

    Industry High Frequency Application Guidance - EPRI 3002004396
    NRC letter endorsing High Frequency Application Guidance

15. 5.15 Spent Fuel Pool (SFP) Evaluation Guidance Industry SFP evaluation guidance –

EPRI 3002007148
NRC letter endorsing SFP evaluation guidance

16. 5.16 NRC Letter - Treatment of Seismic and Flooding Hazard Reevaluations in the Design and Licensing Basis
17. 5.17 Phase 2 Decisionmaking Guidance NRC Guidance for Regulatory

Decisionmaking of reevaluated flooding and seismic hazards
Revision 1 of the Phase 2 guidance

Final Determinations of Required Seismic Evaluations

18. 5.18 NRC Final Determination of Required Seismic

Evaluations

19. 5.19 Licensee Required Seismic Evaluation

Submittals
High Frequency Confirmation
SFP Evaluation
Seismic Probabilistic Risk Assessment

20. 5.20 Audit plan of seismic evaluations submittals
21. 5.21 NRC Staff Assessment of Seismic

Evaluations
High Frequency Confirmation
SFP Evaluation
SPRA Response Letter
Correction regarding staff review of SPRA

22. 5.22 NRC Treatment of Reevaluated Seismic Hazard Information (seismic binning letter)

NA NRC approval of relaxation of SPRA due date from March 2018 until September 2018

December 19, 2014 June 30, 2015

July 30, 2015

September 17, 2015

February 23, 2016 March 17, 2016

September 29, 2015

September 21, 2016

March 2, 2020 October 27, 2015

In SPRA
December 15, 2017 August 28, 2018 July 6, 2017

In SPRA
July 10, 2018 June 10, 2019 October 8, 2019 July 3, 2019

April 24, 2018

ML14353A333 ML15173A385

ML15223A095 ML15218A569

ML16055A017 ML15350A158 ML15127A401

ML16237A103 ML20043D958 ML15194A015

In SPRA ML17349A096 ML18240A065 ML17177A446

In SPRA ML18187A403 ML19053A469 ML19248C756 ML19140A307

ML18093B511

TABLE 6
Request for Information Under Title 10 of the *Code of Federal Regulations*, Section 50.54(f), Enclosure 2: Recommendation 2.1 Flooding Hazard

Reevaluation

Ref

Document

Date

ADAMS Accession No.

Initial Guidance Documents

1. 6.1 NRC prioritization of plants for completing flood hazard reevaluations
2. 6.2 NRC guidance for performing an integrated assessment (IA) for external flooding (JLD-ISG-2012-05)
3. 6.3 NRC letter to industry describing when an IA is expected

4. 6.4 NRC guidance for performing a tsunami, surge, or seiche hazard assessment (JLD-ISG-2012-06)
5. 6.5 NRC letter to industry with guidance on the content of flooding reevaluation submittals
6. 6.6 NRC guidance for assessing flooding hazards due to dam failure (JLD-ISG-2013-01)

Flood Hazard Reevaluation Report (FHRR)

7. 6.7 Licensee FHRR Submittal Cover Letter
8. 6.8 FHRR Regulatory Audit

   NRC FHRR Site-Specific Audit Plan NRC FHRR Audit Report

9. 6.9 NRC Inspection of licensee interim actions
10. 6.10 NRC Interim Staff Response to Reevaluated

    Flood Hazards

11. 6.11 NRC Staff Assessment of FHRR

Modified Approach to Flood Hazard Reevaluations 6.12 NRC extension of due dates for IA reports

14. 6.14 NRC SRM for COMSECY-14-0037
15. 6.15 NRC letter on second extension of due date

    for flooding IA reports

16. 6.16 NRC COMSECY-15-0019 “Closure Plan for

    the Reevaluation of Flooding Hazards”

17. 6.17 NRC SRM-COMSECY-15-0019
18. 6.18 NRC letter describing the graded approach to

    flood hazard reevaluation directed by

    SRM-COMSECY-14-0037

19. 6.19 Flooding Assessment Guidance

    NEI 16-05, “External Flooding Assessment Guidelines”
    NRC endorsement of NEI 16-05 - JLD-ISG-2016-01

May 11, 2012 November 30, 2012

December 3, 2012 January 4, 2013

March 1, 2013 July 29, 2013

August 12, 2015 August 26, 2015

September 25, 2017

March 31, 2016 October 30, 2017 November 21, 2014

March 30, 2015 May 26, 2015

June 30, 2015

July 28, 2015 September 1, 2015

June 2016 July 11, 2016

ML12097A509 ML12311A214

ML12326A912 ML12314A412

ML13044A561 ML13151A153

ML15233A067

ML15230A235 ML17255A524

ML16091A120 ML17284A035 ML14303A465

ML15089A236 ML15112A051

ML15153A104

ML15209A682 ML15174A257

ML16165A178 ML16162A301

Not Required	Not Required

6.13	NRC COMSECY-14-0037, "Integration of Mitigating Strategies for Beyond-Design-Basis External Events and the Reevaluat[i]on of Flooding Hazards"	November 21, 2014	ML14309A256

TABLE 6
Request for Information Under Title 10 of the *Code of Federal Regulations*, Section 50.54(f), Enclosure 2: Recommendation 2.1 Flooding Hazard Reevaluation

Ref	Document	Date	ADAMS Accession No.

20. 6.20 Licensee Focused Evaluation submittal
21. 6.21 NRC Staff Assessment of Focused Evaluation
22. 6.22 NRC Generic FE and IA Regulatory Audit

    Plan

23. 6.23 NRC Letter - Treatment of Seismic and

    Flooding Hazard Reevaluations in the Design

    and Licensing Basis

24. 6.24 Phase 2 Decisionmaking Guidance

    NRC Guidance for Regulatory Decisionmaking of reevaluated flooding and seismic hazards
    Revision 1 of the Phase 2 guidance

25. 6.25 NRC Treatment of Reevaluated Flooding Hazard Information (flooding binning letter)

NA NRC approval of relaxation of FHRR due date for 1 year to develop site-specific hydrological model

NA NRC approval of relaxation of FHRR due date from March 2015 until August 2015 to compete site-specific hydrological model

March 17, 2017 November 6, 2017 July 18, 2017

September 29, 2015

September 21, 2016

March 2, 2020 August 20, 2019

July 17, 2014 February 25, 2015

ML17079A052 ML17292B763 ML17192A452

ML15127A401

ML16237A103

ML20043D958 ML19067A247

ML14174A879 ML15036A273

1. 7.1 NRC COMSECY-14-0037, Integration of Mitigating Strategies with Hazard Reevaluations
2. 7.2 NRC SRM-COMSECY-14-0037
3. 7.3 NRC COMSECY-15-0019, Closure Plan for

   Flooding Hazard Reevaluations

4. 7.4 NRC SRM-COMSECY-15-0019
5. 7.5 Process for Mitigating Strategies

   Assessments (MSA)
   Industry Guidance for performing MSAs - NEI 12-06, Revision 2, including Appendices E, G, & H
   NRC endorsement of NEI 12-06, Revision 2 - JLD-ISG-2012-01, Revision 1
   Industry Guidance for performing MSAs - NEI 12-06, Revision 4
   NRC endorsement of NEI 12-06, Revision 4 - JLD-ISG-2012-01, Revision 2

6. 7.6 Licensee's MSA submittal - Flooding
7. 7.7 NRC Staff Assessment of MSA - Flooding
8. 7.8 Licensee's MSA submittal – Seismic
9. 7.9 NRC Staff Assessment of MSA - Seismic
10. 7.10 NRC MSA Audit Plan

November 21, 2014

March 30, 2015 June 30, 2015

July 28, 2015 December 2015

January 22, 2016

December 12, 2016

February 8, 2017

June 30, 2016 January 11, 2017 September 28, 2018

Not Required December 5, 2016

ML14309A256

ML15089A236 ML15153A104

ML15209A682 ML16005A625

ML15357A163 ML16354B416 ML17005A182

ML16182A009 ML16362A208 ML18271A007

Not Required ML16259A189

Peach Bottom Atomic Power Station, Units 2 and 3

TABLE 7
Mitigating Strategies Assessments (MSA)

Ref	Document	Date	ADAMS Accession No.

Peach Bottom Atomic Power Station, Units 2 and 3

TABLE 8
Request for Information under Title 10 of the *Code of Federal Regulations*, Section 50.54(f), Enclosure 3: Recommendation 2.3 Seismic Walkdown

Ref	Document	Date	ADAMS Accession No.

1. 8.1 Industry Seismic Walkdown Guidance with NRC endorsement letter - EPRI 1025286
2. 8.2 NRC letter endorsing EPRI 1025286
3. 8.3 Licensee Seismic Hazard Walkdown Report

   Licensee Seismic Hazard Walkdown Report Package
   Exelon proposed resolution to complete seismic walkdowns for Units 2 and 3
   Response to RAIs

   Updated inaccessible items report for Unit 3 and one (1) Unit 2 item common to both units

4. 8.4 NRC Inspection of Seismic Walkdowns NRC TI 2515/188

   NRC Integrated Inspection Report 2012-

   005 (TI 2515/188 inspection results)

5. 8.5 Subsequent seismic walkdown report for Unit

   2 (addresses final inaccessible items)

6. 8.6 NRC Staff Assessment of Seismic Walkdown

   Report

7. 8.7 NRC review of delayed walkdown items

May 31, 2012

May 31, 2012

November 19, 2012

September 16, 2013
November 27, 2013 March 25, 2014

July 6, 2012 January 29, 2013

October 31, 2014 April 30, 2014

September 25, 2015

ML12188A031

ML12145A529

ML130030017

ML13260A083

ML13331B501 ML14091A479

ML12156A052 ML13029A013

ML14304A223 ML14086A624 ML15268A477

TABLE 9
Request for Information under Title 10 of the *Code of Federal Regulations*, Section 50.54(f), Enclosure 4: Recommendation 2.3 Flooding Walkdown

Ref	Document	Date	ADAMS Accession No.

1. 9.1 Industry Flooding Walkdown Guidance - NEI 12-07
2. 9.2 NRC letter endorsing NEI 12-07
3. 9.3 Licensee Flooding Hazard Walkdown Report

   Flooding Hazard Walkdown Report Update to Flooding Hazard Walkdown Report – APM Assessment

4. 9.4 NRC Inspection of Flooding Walkdowns NRC TI 2515/187

   NRC Integrated Inspection Report 2012-

   005 (TI 2515/187 inspection results)

5. 9.5 NRC Staff Assessment of Flooding Walkdown

Report

6. 9.6 Completion of restricted access items

walkdown

May 31, 2012 May 31, 2012

November 19, 2012 January 31, 2014

June 27, 2012 January 29, 2013

June 17, 2014 January 29, 2016

ML12173A215 ML12144A142

ML123250714 ML14031A443

ML12129A108 ML13029A013

ML14119A057 ML16029A336

Peach Bottom Atomic Power Station, Units 2 and 3

TABLE 10
Request for Information under Title 10 of the *Code of Federal Regulations*, Section 50.54(f), Enclosure 5: Recommendation 9.3 Emergency Preparedness Communications and Staffing

Ref

Document

Date

ADAMS Accession No.

1. 10.1 Guidance Documents
   Industry Guidance for Emergency

   Preparedness staffing and communications - NEI 12-01 NRC letter endorsing NEI 12-01

2. 10.2 Exelon 60 day response and proposed alternative course of action
3. 10.3 Exelon 90 day response to communications and staffing information requests
4. 10.4 NRC letter – status of 90-day response
5. 10.5 Licensee communications assessment

Licensee communications assessment NRC letter on generic technical issues
Licensee communications assessment supplement

6. 10.6 NRC staff assessment of licensee's communications assessment
7. 10.7 Licensee Phase 1 staffing assessment (multiunit sites only)
8. 10.8 NRC response to licensee's Phase 1 staffing assessment
9. 10.9 Licensee Phase 2 staffing assessment response

Licensee Phase 2 staffing assessment for functions related to mitigation strategies Licensee response to RAI

10. 10.10 NRC Phase 2 staff assessment response
11. 10.11 NRC Inspection of Licensee Responses to EA-12-049, EA-12-051, and Emergency Preparedness Information

NRC TI 2515/191, Revision 2
NRC TI 2515/191 Inspection Report 2018- 012

May 2012

May 15, 2012 May 10, 2012

June 11, 2012

July 26, 2012

October 31, 2012 January 23, 2013 February 22, 2013

July 12, 2013 April 30, 2013 October 23, 2013

May 8, 2015

August 28, 2015 September 30, 2015

July 10, 2018 March 5, 2019

ML12125A412

ML12131A043 ML12135A391

ML12164A572 ML12200A106

ML12306A199 ML13010A162 ML13056A135

ML13114A067 ML13121A087 ML13233A183

ML15128A252

ML15240A006 ML15268A424

ML18191B074 ML19065A010

Peach Bottom Atomic Power Station, Units 2 and 3

TABLE 11
Additional Licensee Commitments – SAMGs and Multisource Dose

Assessments

Ref

Document

Date

ADAMS Accession No.

Update and Maintain SAMGs

1. 11.1 SECY-15-0065: Proposed Rulemaking: Mitigation of Beyond-Design-Basis Events (RIN 3150-AJ49)
2. 11.2 SRM-SECY-15-0065
3. 11.3 NEI Letter describing industry initiative to

   update and maintain SAMGs

4. 11.4 Site Commitment to Maintain SAMGs
5. 11.5 NRC letter to NEI describing approach to

   SAMG oversight

6. 11.6 NRC Inspection Procedure 71111.18, “Plant

   Modifications”
   Revision effective January 1, 2017 Revision effective January 1, 2019

7. 11.7 NEI 14-01, “Emergency Response Procedures and Guidelines for Extreme Events and Severe Accidents,” Rev. 1

Multisource Dose Assessments

8. 11.8 NEI Letter: Industry survey and plan for multiunit dose assessments
9. 11.9 NRC Letter to request additional information from NEI on multiunit dose assessment capability
10. 11.10 NEI Letter: Implementation of Multiunit Dose Assessment Capability

11. 11.11 Licensee Response Regarding the Capability to Perform Multisource Offsite Dose Assessment
12. 11.12 NRC Acknowledgment of Licensee Dose Assessment Submittals
13. 11.13 COMSECY-13-0010
14. 11.14 SRM-COMSECY-13-0010
15. 11.15 NRC Inspection of Licensee Responses to

    EA-12-049, EA-12-051, and Emergency Preparedness Information

    NRC TI 2515/191, Revision 2
    NRC TI 2515/191 Inspection Report 2018- 012

16. 11.16 NEI 13-06, "Enhancements to Emergency Reponses Capabilities for Beyond-Design- Basis Accidents and Events," Rev. 1

April 30, 2015

August 27, 2015 October 26, 2015

December 4, 2015 February 23, 2016

November 17, 2016 November 19, 2018 February 2016

January 28, 2013 February 27, 2013

March 14, 2013 June 27, 2013

January 29, 2014 March 27, 2013

April 30, 2013

July 10, 2018 March 5, 2019

February 2016

ML15049A201

ML15239A767 ML15335A442

ML15338A125 ML16032A029

ML16306A185 ML18176A157 ML16224A619

ML13028A200 ML13029A632

ML13073A522 ML13179A098

ML13233A205

ML12339A262 ML13120A339

ML18191B074 ML19065A010

ML16224A618

TABLE 12
NRC Semi-Annual Status Reports to the Commission

12.2 SECY-12-0095, Enclosure 1 - Second July 13, 2012 ML12165A092 6-Month Status Update on Charter Activities -
February 2012 - July 2012

Ref	Document	Date	ADAMS Accession No.
12.1	SECY-12-0025, Enclosure 8, “Proposed Orders and Requests for Information in Response to Lessons Learned from Japan’s March 11, 2011, Great Tōhoku Earthquake and Tsunami”	February 17, 2012	ML12039A103
12.3	SECY-13-0020 - Third 6-Month Status Update on Response to Lessons Learned from Japan's March 11, 2011, Great Tōhoku Earthquake and Subsequent Tsunami	February 14, 2013	ML13031A512
12.4	SECY-13-0095 - Fourth 6-Month Status Update on Response to Lessons Learned from Japan's March 11, 2011, Great Tōhoku Earthquake and Subsequent Tsunami	September 6, 2013	ML13213A304
12.5	SECY-14-0046 - Fifth 6-Month Status Update on Response to Lessons Learned from Japan's March 11, 2011, Great Tōhoku Earthquake and Subsequent Tsunami	April 17, 2014	ML14064A520
12.6	SECY-14-0114 - Sixth 6-Month Status Update on Response to Lessons Learned from Japan's March 11, 2011, Great Tōhoku Earthquake and Subsequent Tsunami	October 21, 2014	ML14234A498
12.7	SECY-15-0059 - Seventh 6-Month Status Update on Response to Lessons Learned from Japan's March 11, 2011, Great Tōhoku Earthquake and Subsequent Tsunami	April 9, 2015	ML15069A444
12.8	SECY-15-0128: Eighth 6-Month Status Update on Response to Lessons Learned from Japan's March 11, 2011, Great Tōhoku Earthquake and Subsequent Tsunami	October 14, 2015	ML15245A473
12.9	SECY-16-0043: Ninth 6 Month Status Update on Response to Lessons Learned from Japan's March 11, 2011, Great Tōhoku Earthquake and Subsequent	April 5, 2016	ML16054A255

	Tsunami		
12.10	SECY-17-0016: Status of Implementation of Lessons Learned from Japan's March 11, 2011, Great Tōhoku Earthquake and Subsequent Tsunami	January 30, 2017	ML16356A084

PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 – DOCUMENTATION OF THE COMPLETION OF REQUIRED ACTIONS TAKEN IN RESPONSE TO THE LESSONS LEARNED FROM THE FUKUSHIMA DAI-ICHI ACCIDENT DATED DECEMBER 10, 2020

DISTRIBUTION:
PUBLIC
LPMB R/F
RidsNrrDorlLpmb Resource RidsNrrDorlLpl1 Resource RidsNrrDorl Resource RidsNrrPMPeachBottom Resource

RidsNrrLaSLent Resource RidsOgcMailCenter Resource RidsOpaMail Resource RidsACRS_MailCTR Resource RidsNrrDex Resource RidsRgn1MailCenter Resource

KMorgan-Butler, NRR RBernardo, NRR SPhilpott, NRR JTobin, NRR

ADAMS Accession No. ML20339A397
OFFICE NRR/DORL/LPMB/PM* NRR/DANU/UARL/LA* NRR/DORL/LPMB/BC(A)* NRR/DORL/LPMB/PM*
NAME RBernardo SLent KMorgan-Butler RBernardo
DATE 12/3/2020 12/4/2020 12/10/2020 12/10/2020

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December 21, 2020 – Email from Jennifer Tobin to David P Helker (GenCo-Nuc) with cc to Richard W. Gropp, Jr. (Exelon Nuclear) with subject of Peach Bottom Units 2 and 3 - Request for Additional Information - TSTF-505 (EPID L-2019-LLA-0120)

By application dated May 29, 2020, Exelon Generation Company, LLC (the licensee) submitted a license amendment request (LAR) for Peach Bottom Atomic Power Station, Units 2 and 3 (Peach Bottom) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20150A007). The proposed amendment would modify TS requirements to permit the use of Risk Informed Completion Times (RICT) in accordance with TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b," (ADAMS Accession No. ML18183A493). A model safety evaluation was provided by the NRC to the TSTF on November 21, 2018 (ADAMS Accession No. ML18253A085).

The Nuclear Regulatory Commission's (NRC) staff is reviewing your submittal and has determined that additional information is needed to complete its review. The purpose of this email is to provide a draft copy of a request for additional information (RAI) for your review to ensure that:

the draft RAI question is understandable,
the basis for the question is clear, and
to determine whether the information being requested has been previously docketed.

The specific requests for additional information (RAI) questions are provided below.

RAI #1

By application dated May 29, 2020, Exelon Generation Company, LLC (the licensee) submitted a license amendment request (LAR) for Peach Bottom Atomic Power Station, Units 2 and 3 (Peach Bottom) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20150A007). Section 2.3 of LAR Attachment 1 states that the application of a risk-informed completion time (RICT) will be evaluated using the guidance provided in Nuclear Energy Institute (NEI) Topical Report NEI 06-09, Revision 0-A, "Risk- Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines, Industry Guidance Document," dated November 6, 2006 (ADAMS Package Accession No. ML122860402) (hereafter NEI 06-09). NEI 06-09 was approved by the NRC on May 17, 2007 (ADAMS Accession No. ML071200238). The NRC safety evaluation (SE) for NEI 06-09, states, "[t]he impact of the proposed change should be monitored using performance measurement strategies." NEI 06-09 considers the use of NUMARC 93-01, Revision 4F, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants (ADAMS Accession No. ML18120A069), as endorsed by Regulatory Guide (RG) 1.160, Revision 4 (ADAMS Accession No. ML18220B281), for the implementation of the Maintenance Rule. NUMARC 93-01, Section 9.0, contains guidance for the establishment of performance criteria.

Furthermore, Section 2.3 of LAR Attachment 1 states:

In addition, the NEI 06-09-A, Revision 0 methodology satisfies the five key safety principles specified in Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decision-making: Technical Specifications," dated August 1998 (ADAMS Accession No. ML003740176), relative to the risk impact due to the application of a RICT.

NRC staff position C.3.2 provided in RG 1.177 for meeting the fifth key safety principle acknowledges the use of performance criteria to assess degradation of operational safety over a period of time. It is unclear to the NRC staff how the licensee's process for the risk- informed application captures performance monitoring for the structures, systems, and components (SSCs) within-scope of the application. In light of these observations, address either (a) or (b) below.

a) Confirm that the Peach Bottom Maintenance Rule program incorporates the use of performance criteria to evaluate SSC performance as described in the NRC-endorsed guidance in NUMARC 93-01.

OR

b) Describe the approach/method used by Peach Bottom for SSC performance monitoring as described in Regulatory Position C.3.2 referenced in RG 1.177 for meeting the fifth key safety principle. In the description, include criteria (e.g., qualitative or quantitative), along with the appropriate risk metrics, and explain how the approach and criteria demonstrate the intent to monitor the potential degradation of SSCs in accordance with the NRC SE for NEI 06-09.

RAI #2

RG 1.174, Revision 3 states the licensee should assess whether the proposed licensing basis change meets the defense-in-depth principle by not over-relying on programmatic activities as compensatory measures associated with the change in the licensing basis. RG 1.174 further elaborates that human actions (e.g., manual system actuation) are considered as one type of compensatory measure.

In LAR Attachment 5, if the only diverse means identified are the manual actuations, then provide a summary of the evaluation that these means are adequate. For example, confirm that these "manual actuations" identified as the only diverse means are modeled in the plant PRA, defined in plant operation procedures to which operators are trained, and confirm the manual action completion times associated with these actions are evaluated as adequate.

This information is needed to demonstrate compliance with 10 CFR 50.36 and 50.55(a).

Please submit your response to this request for additional information by January 29th. A clarification call was not needed.

If you have questions please don't hesitate to contact me.

January 5, 2021 – Letter from Anthony Dimitriadis, Chief Decommissioning, ISFSI, and Reactor Health Physics Branch Division of Nuclear Materials Safety to Bryan C. Hanson Senior Vice President, Exelon Generation Company, LLC President and Chief Nuclear Officer Exelon Nuclear with subject of PEACH BOTTOM ATOMIC POWER STATION – INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) NRC INSPECTION REPORT NO. 07200029/2020001

On August 14, 2020, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection of the Peach Bottom Atomic Power Station (Peach Bottom) Independent Spent Fuel Storage Installation (ISFSI) activities. On-site inspections of the welding dry run were performed on March 16-17, 2020, at the Holtec training facility in Camden, New Jersey. Additional inspection activities (in office reviews via remote means) were conducted throughout the period as a consequence of the COVID-19 public health emergency (PHE). The inspectors examined activities conducted under your licenses as they relate to safety and compliance with the Commission's rules and regulations, and the conditions of your licenses and the Certificate of Compliance (COC). The inspection consisted of observations by the inspectors, interviews with site personnel, and a review of procedures and records. The results of this inspection were discussed with Ron DiSabitino, Operations Director and other members of your staff on September 2, 2020, and are documented in the enclosed report.

The report documents one NRC-identified violation of NRC requirements of very low safety significance (Severity Level IV). Because of the very low safety significance and because it was entered into your corrective action program, the NRC is treating the violation as a Non- Cited Violation (NCV) consistent with Section 2.3.2.a of the NRC Enforcement Policy.

If you contest the violation or the significance of the NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555- 0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement; and the NRC Resident Inspector office at the Peach Bottom Atomic Power Station.

In accordance with Title 10 Code of Federal Regulations (CFR) 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure, and your response (if any) will be made available electronically for public inspection in the NRC Public Document Room or from the NRC document system (ADAMS), accessible from the NRC Web site at

http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction.

Current NRC regulations and guidance are included on the NRC's Web site at www.nrc.gov; select **Radioactive Waste; Decommissioning of Nuclear Facilities**; then **Regulations, Guidance and Communications**. The current Enforcement Policy is included on the NRC's website at www.nrc.gov; select **About NRC**, **Organizations & Functions**; **Office of Enforcement**; **Enforcement documents**; then **Enforcement Policy** (Under 'Related Information'). You may also obtain these documents by contacting the Government Printing Office (GPO) toll-free at 1-866-512-1800. The GPO is open from 8:00 a.m. to 5:30 p.m. EST, Monday through Friday (except Federal holidays).

No reply to this letter is required. Please contact John Nicholson at 610-337-5236 if you have any questions regarding this matter.

PEACH BOTTOM ATOMIC POWER STATION – INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) NRC INSPECTION REPORT NO. 200029/2020001 DATED JANUARY 5, 2021.

U.S. NUCLEAR REGULATORY COMMISSION REGION I

Inspection Report

Docket number: 072-00029
License number: DPR-44, DPR-56 07200029/2020001
Licensee: Exelon Generation Company, LLC

Facility: Peach Bottom Atomic Power Station

Location: Delta, PA
Dates: March 16 - August 14, 2020

Inspectors: J. Nicholson, Senior Health Physicist Decommissioning, ISFSI, and Reactor HP Branch Division of Nuclear Materials Safety, Region I

M. Davis,Senior Transportation and Storage Safety Inspector Inspections and Operations Branch Division of Fuel Management, NMSS

M. Learn, Senior Transportation and Storage Safety Inspector Inspections and Operations Branch Division of Fuel Management, NMSS

J. Schoppy, Senior Reactor Inspector Division of Reactor Safety, Region I

Approved by: Anthony Dimitriadis, Chief Decommissioning, ISFSI, and Reactor HP Branch Division of Nuclear Materials Safety, Region I

EXECUTIVE SUMMARY

Exelon Generation Company, LLC Peach Bottom Atomic Power Station
NRC Inspection Report No. 07200029/2020001

On August 14, 2020, the U.S. Nuclear Regulatory Commission (NRC) competed a period of onsite and remote inspections of Peach Bottom's Independent Spent Fuel Storage Installation (ISFSI) activities. Onsite inspections were performed on March 16-17, 2020 at Holtec's training facility in Camden, New Jersey to inspect Peach Bottom's pre-operational ISFSI testing. Additional inspection activities (in office reviews via remote means) were conducted throughout the period as a consequence of the COVID-19 public health emergency (PHE). The inspection consisted of observations by the inspectors, interviews with site personnel, and a review of procedures and records. The NRC's program for overseeing the safe operation of dry storage of spent fuel at an ISFSI is described in Inspection Manual Chapter 2690, "Inspection Program for Dry Storage of Spent Reactor Fuel at Independent Spent Fuel Storage Installations and for 10 Code of Federal Regulations (CFR) Part 71 Transportation Packagings."

List of Violations

A Severity IV Non-Cited Violation (NCV) , of 10 CFR 72.48 was identified by the inspectors because Exelon Generation, LLC., (Exelon) did not perform written evaluations which provide the bases for the determination that a change did not require a Certificate of Compliance (CoC) amendment pursuant to paragraph 72.48(c)(2). Specifically, Exelon did not perform a written evaluation to demonstrate that transporting the HI-TRAC VW and MPC on a HI-PORT transporter without redundant drop protection feature on site at the cask handling facility created a possibility for a malfunction with a different result than any previously evaluated in the Holtec Final Safety Analysis Report (FSAR).

REPORT DETAILS

1. **1.0 Independent Spent Fuel Storage Installation**
2. 1.1 Background

Owner/Operator Exelon – Peach Bottom Atomic Power Station (PBAPS), selected Holtec HI-STORM FW Cask System technology to allow spent nuclear fuel assemblies currently stored at PBAPS Units 2 and 3 spent fuel pool (SFP) to be relocated and stored using an ISFSI. The Holtec system is listed in 10 CFR 72.214, “List of Approved Spent Fuel Storage Casks,” under Certificate of Compliance (CoC) No. 1032 with an effective date of December 17, 2014 and a 20-year term. Holtec updated Final Safety Analysis Report (UFSAR), Revision 4 applies to the Holtec ISFSI system that was placed in service under CoC number 1032.

3. 1.2 Pre-operational Testing of an ISFSI (IP 60854)

a. Inspection Scope

The inspectors evaluated PBAPS performance during NRC observed pre-operational dry run activities that were performed in order to fulfill requirements in the NRC-issued Certificate of Compliance (CoC) No. 1032, Amendment 1, Revision 1 (CoC 1032-1R1). The inspectors observed PBAPS dry run activities on March 16-17, 2020, at the Holtec facility in Camden, New Jersey. Due to NRC travel restrictions associated with the COVID-19 PHE, subsequent inspections of the remaining three dry runs and the initial dry cask loading campaign were performed via remote means (Skype) on March 30 - April 3, April 24 - 28, and May 14, 26 – 27, 2020.

During the dry run activities, the inspectors observed cask processing activities to determine whether Exelon had developed, implemented, and evaluated preoperational testing activities to safely process the multi-purpose canister (MPC) to be used in storage of spent fuel at the PBAPS site. The inspectors observed MPC activities including blowdown, vacuum drying, helium backfilling, welding, hydrogen monitoring, and non-destructive examinations. The inspectors verified that the vacuum drying system was leak tight and the helium flow path was operable. The inspectors examined the MPC processing equipment and reviewed worker qualification records. The inspectors also observed cask loading and cask movement activities to determine whether Exelon had developed the capability to properly load and move the MPC to be used in storage of spent fuel at PBAPS. The inspectors observed: (a) movement of a dummy fuel assembly into the MPC, (b) down-ending the HI-TRAC transfer cask/MPC onto the self-propelled motorized transporter (SPMT), (c) transportation of the HI-TRAC/MPC to the ISFSI pad, (d) upending the HI-TRAC/MPC, stack-up and transfer of the MPC from the HI-TRAC to the HI-STORM at the cask transfer facility (CTF), (e) retrieval of the MPC from the HI-STORM back into the HI-TRAC, (f) installation of the HI-STORM lid, (g) lifting of the HI-STORM out of the CTF, and (h) placement of the HI-STORM on the ISFSI pad.

The inspectors attended select PBAPS pre-job briefings to assess Exelon’s ability to identify critical steps of the evolution, potential failure scenarios, and human performance tools to prevent errors. The inspectors reviewed the training program and

training records of personnel assigned to ISFSI activities. The inspectors reviewed MPC loading, unloading, and processing procedures to determine if they contained commitments and requirements specified in the CoC, technical specifications (TSs), Final Safety Analysis Report (FSAR), and Title 10 of the CFR Part 72. The inspectors also reviewed fuel selection procedures to ensure they appropriately incorporated the requirements in the TSs.

The inspectors reviewed radiation protection procedures and radiation work permits associated with the proposed ISFSI loading campaign. The inspectors also reviewed the radiological controls which would be established during a MPC loading campaign.

The inspectors reviewed corrective action reports associated with preparations for the ISFSI loading campaign to ensure that issues were being properly identified, prioritized, and evaluated commensurate with their safety significance.

b. Findings
No findings of significance were identified.

1.2 Operation of an ISFSI at Operating Plants (IP 60855) a. Inspection Scope

From June 1 - 19, 2020, the inspectors observed and evaluated Exelon's loading of the first MPC associated with its initial Holtec HI-STORM FW Cask System dry cask campaign. The inspectors also reviewed the licensee's planned activities related to long-term operation and monitoring of the ISFSI. The inspectors evaluated compliance with the CoC, TSs, and station procedures.

The inspectors observed fuel assemblies being loaded into the MPC. The inspectors also observed MPC processing operations including installation of the automated welding system, welding, non-destructive weld examinations, blowdowns, vacuum drying, helium backfill, and survey activities. During performance of these activities, the inspectors verified that procedure use, communication, and coordination of ISFSI activities met established Exelon standards and requirements.

The inspectors reviewed PBAPS's program associated with fuel characterization and selection for storage. The inspectors reviewed the first cask fuel selection package to determine if the licensee was loading fuel in accordance with the CoC, TSs, and procedures. Inspectors reviewed a recording made of the fuel assemblies loaded into the first DSC to ensure the loading was in accordance with PBAPS's loading plan.

The inspectors observed radiation protection surveys and job coverage for the cask loading workers. The inspectors reviewed survey data maps and radiological records from the first MPC loading to determine if radiation survey levels measured were within limits specified by the TSs and consistent with values specified in the FSAR.

The inspectors reviewed corrective action reports and the associated follow-up actions that were generated since PBAPS's dry run demonstrations to ensure that issues were entered into the corrective action program, prioritized, and evaluated commensurate with their safety significance.

b. Findings
No findings of significance were identified.

1.3 Review of 10 CFR 72.212 (b) Evaluations (IP 60856)

1. Inspection Scope

   PBAPS selected the Holtec International HI-STORM FW Cask System for the storage of spent fuel at the onsite expanded ISFSI. The HI-STORM FW casks augment the TN-68 casks already in service at the original ISFSI, which began operation in calendar year 2000.

   The review of the HI-STORM FW Cask System was based on NRC-issued Certificate of Compliance (CoC) No. 1032, Amendment 1, Revision 1 (CoC 1032-1R1) and its associated Safety Evaluation Report (SER), and HI-STORM FW Final Safety Analysis Report (FSAR) Revision 4. The review of the Part 50 facility site-specific parameters utilized the Updated Final Safety Analysis Report (UFSAR) and other applicable plant- specific design and licensing basis information.

   The inspectors evaluated Exelon’s compliance with the requirements outlined in 10 CFR 72.212. The inspectors examined the licensee’s written evaluations to determine if they were in accordance with 10 CFR 72.212(b)(5) and evaluated the conditions set forth in the CoC to determine if conditions had been met prior to use and if the radiological requirements of 72.104 were met. The inspectors examined applicable reactor site parameters, such as fire and explosions, tornadoes, wind-generated missile impacts, seismic qualifications, lightning, flooding and temperature, to determine if they had been evaluated for acceptability with bounding values specified in the FSAR and the NRC SER. The inspectors also examined 50.59 evaluations related to the construction and operation of the ISFSI and plant interfaces to determine if they were performed and to determine if changes to certain facility design bases and UFSAR commitments required NRC approval. The reactor emergency plan, quality assurance program, training program, and radiation protection program were reviewed to determine if there was a decrease in effectiveness and if changes made required prior NRC approval.

2. Findings Introduction

   The inspectors identified a Severity Level IV Non-Cited Violation (NCV) of 10 CFR 72.48 because the licensee did not perform written evaluations which provide the bases for the determination that the change did not require a Certificate of Compliance (CoC) amendment pursuant to paragraph 72.48(c)(2). Specifically, the licensee performed changes that required a written evaluation with one of the changes requiring NRC review and approval in accordance with 72.48(c)(2)(6).

   Description

The inspectors evaluated engineering change (EC) number (No.) 618376 and assessed screening No. PB-2017-7248-009-S, Revision 1, as required by 10 CFR 72.48 associated with several of the design changes to expand the PBAPS ISFSI.

The inspectors noted that the Holtec FSAR addressed the environmental phenomena loads, design criteria and accident condition in sections 1.2.1.3.b, 2.2.3.e and 12.2.6.2, respectively. Section 1.2.1.3 of the Holtec FSAR states, in part, that the HI-TRAC VW transfer cask provide protection of the multiple purpose canister (MPC) against extreme environmental phenomena loads, such as tornado-borne missiles, during short term operations (e.g., on-site handling of a loaded HI-TRAC VW transfer cask).

Section 2.2.3 of the Holtec FSAR states, in part, the kinematic stability of the HI-STORM FW overpack, and continued integrity of the MPC confinement boundary, within the storage overpack or HI-TRAC VW transfer cask, must be demonstrated under impact from potential tornado-generated missiles in conjunction with the wind loadings.

Section 12.2.6.2 of the Holtec FSAR states, in part, that it is not credible that a potential large tornado missile and/or wind could tip-over the loaded HI-TRAC VW transfer cask while being handled in the vertical orientation because it shall be attached to a lifting device designed in accordance with the requirements specified in the FSAR, section 2.3.3. Section 2.3.3 describes the equipment for redundant drop protection features at a handling facility outside the reactor structure.

The inspectors identified that the Holtec FSAR describing accident conditions for a tornado analysis was different than the configuration used at the PBAPS ISFSI and affected a design function, which constituted a change. Specifically, PBAPS transports the HI-TRAC VW and MPC on a transporter called the HI-PORT with no redundant drop protection feature outside the reactor structure. The inspectors reviewed the guidance provided in the licensee's Manual LS-AA-114-1000, "72.48 Resource Manual," Revision 1 and Regulatory Guide (RG) 3.72, "Guidance for Implementation of 10 CFR 72.48, Changes, Tests, and Experiments," which endorsed Nuclear Energy Institute (NEI) 96-07, Appendix B for the industry guidance to determine if the screening required an evaluation and required prior NRC review and approval before implementing the change.

As stated, in part, in LS-AA-114-1000, any change that adversely affects a Holtec FSAR described design function, a method of performing or controlling design functions, or evaluation that demonstrates that the intended design function will be accomplished, is screened in as a written evaluation to provide the bases for the determination that the change, test, or experiment does not require a license amendment pursuant to paragraph (c)(2). Furthermore, the following is an example of a change that the industry guidance document consider adverse and must be screened in to an evaluation: (1) any change that alters a design basis limit for fission product barrier positively or negatively is considered adverse and must be screened in; (2) if the effect of a change is such that existing safety analyses would no longer be bounding and therefore UFSAR safety analyses must be re-run to demonstrate that all required safety functions and design requirements are met, the change is considered to be adverse and must be screened in (B4.2.1, Screening for Adverse Effects).

Based on the above guidance documents, the inspectors noted that the licensee stopped at a screening and did not perform a full evaluation. The inspectors identified a non-cited violation (NCV) of 10 CFR 72.48 because Exelon did not perform written evaluations which provide the bases for the determination that a change did not require a Certificate of Compliance (CoC) amendment pursuant to paragraph 72.48(c)(2). Specifically, Exelon did not address the applicable criteria identified in 10 CFR 72.48(c)(2) because the change was adverse and the protection against natural phenomena and environmental conditions were established as a part of the design requirements for general licensees. Therefore, the inspectors noted that criterion (vi) of 10 CFR 72.48(c)(2) requires, in part, that a general licensee shall request that the certificate holder obtain a CoC amendment pursuant to 10 CFR 72.244, prior to implementing a proposed change if the change would create a possibility for a malfunction with a different result than any previously evaluated in the FSAR. The inspectors assessed that the change increased the likelihood of a malfunction previously thought to be incredible since accident analysis states that it is not credible that a large tornado missile could tip-over the loaded HI-TRAC VW transfer cask while being handled in the vertical orientation due to redundant drop protection feature.

Analysis

In accordance with Section 2.2 of the Enforcement Policy and Inspection Manual Chapter 0612, Appendix B, “Issue Screening,” ISFSIs are not subject to the Significance Determination Process and are not subject to the Reactor Oversight Process, therefore, violations identified at ISFSIs are assessed using traditional enforcement. Traditional enforcement violations are not assessed for cross-cutting aspects.

The inspectors assessed the significance of the violation using the NRC Enforcement Policy and Enforcement Manual. The inspectors determined that the violation had the potential for impacting the NRC’s ability to perform its regulatory oversight function because the licensee did not receive prior NRC approval for changes in licensed activities. The inspectors determined that the violation was more than minor because the licensee did not seek prior NRC review and approval. The inspectors characterized the violation as a Severity Level IV violation because the licensee implemented an administrative control to preclude any possibility of an unwanted system interaction by limiting the movement of the HI-PORT if adverse weather is expected.

The licensee entered the issue into its corrective action program under IR 04352694. Because the violation was of low safety significance and was entered into Exelon’s CAP, the issue was not repetitive or willful, this is being treated as a Non-Cited Violation (NCV), consistent with Section 2.3.2.a of the Enforcement Policy.

Enforcement

10 CFR 72.48(d)(1) requires, in part, that the licensee shall maintain records of changes in the facility or spent fuel storage cask design, of changes in procedures, and tests and experiments made pursuant to paragraph (c) of this section. These records must include a written evaluation which provides the bases for the determination that the change does not require a Certificate of Compliance (CoC) amendment pursuant to paragraph (c)(2) of this section.

10 CFR 72.48(c)(2)(vi) requires, in part, that a general licensee shall request that the certificate holder obtain a CoC amendment pursuant to 10 CFR 72.244, prior to

implementing a proposed change if the change would create a possibility for a malfunction with a different result than any previously evaluated in the FSAR.

Contrary to the above, as of May 27, 2020, Exelon did not include a written evaluation that provided the bases for the determination that the change does not require a CoC amendment pursuant to 10 CFR 72.48(c)(2) and implemented a change that would create a possibility for a malfunction with a different result than any previously evaluated in the FSAR without prior NRC review and approval. Specifically, Exelon did not perform a written evaluation to demonstrate that transporting the HI-TRAC VW and MPC on a HI-PORT transporter with no redundant drop-protection feature on site at its cask handling facility outside the reactor structure created a possibility for a malfunction with a different result than any previously evaluated in the Holtec FSAR. Because this violation was of low safety significance and was entered into Exelon's CAP, the issue was not repetitive or willful, this is being treated as a Severity Level IV, Non-Cited Violation (NCV), consistent with Section 2.3.2.a of the Enforcement Policy.

2.0 Exit Meeting

On September 2, 2020, the inspectors presented the inspection results to Mr. Ron DiSabitino, Operations Director, and other Exelon personnel who acknowledged the inspection results. No proprietary information was retained by the inspectors or documented in this report.

ATTACHMENT: SUPPLEMENTAL INFORMATION

Licensee
R. DiSabitino A. Stathes P. Gregory

None

SUPPLEMENTAL INFORMATION PARTIAL LIST OF PERSONS CONTACTED

Operations Director
Dry Cask Storage Project Manager Dry Cask Storage Program Manager

ITEMS OPENED, CLOSED, AND DISCUSSED

LIST OF DOCUMENTS REVIEWED Section 1.1 Review of 10 CFR 72.212 (b) Evaluations

10 CFR 50.59 and 10 CFR 72.48 Screenings/Evaluations
618371 618374 618375 618375 618376 618377

Calculations
CoC 2601006A-001; HI-PORT Test Results; Revision 0
DOC-104-209-117; MPC Lift Cleats Test Records; Revision 0
DOC-104-729-121; Lift Links, Brackets Test Results; Revision1
DOC-104-759-127; Lift Yoke Test Results; Revision 0
DOC-2601-012; Mating Device Test Results; Revision 0
HI-0004; Peach Bottom ISFSI Expansion – HI-STORM FW Site Dose Calculation; Revision 000 HI-2135647; Structural Qualification of 415 KIP VCT' Revision 000
HI-2135677; Evaluation of Effects of Tracked VCT Fire on HI-STORM FS System; Revision 000 HI-2177674; Thermal Evaluation of HI-STORM FW System Placed in a CTF at Peach Bottom;

Revision 1
HI-2177675; Evaluation of Effect of Combined HI-PORT and VCT Fire on HI-TRAC VW for

Peach Bottom; Revision 1
HI-2177738; Seismic Stability of HI-TRAC on HI-PORT; Revision 000
HI-2177767; Seismic Structural Analysis of the CTF at Peach Bottom; Revision 000 HI-2177817; Stability Assessment of HI-TRAC in SFP and on Refueling Floor at Peach Bottom;

Revision 000
HI-2177829; VCT Seismic Stability Analysis for PBAPS; Revision 000
HI-2188482; Evaluation of the Structural Integrity of the SFP Wall Liner Impacted by a Loaded

HI-TRAC VW; Revision 000
HI-2188652; HI-TRAC Tipover Analysis Under Explosion Event for Peach Bottom; Revision 000 PS-1120; Purchase Specification for the Vertical Cask Transporter; Revision 9
PS-1208; PB ISFSI Expansion – Rock Run Creek Bridge Structural Capacity; Revision 000 PS-1210; Buried Commodities Evaluation; Revision 000
PS-1213; Seismic Soil Liquefaction of Haul Path; Revision 000
PS-1223; PB ISFSI Expansion – ISFSI Fire Radiant Heat and Explosion Overpressure Analysis;

Revision 0
PS-1227; RB El 234' Floor Evaluation for DCS Equipment Loading for HI-STORM; Revision

7 Enclosure

Calculations (Cont'd)
PS-1228; RB Floor Evaluation for HI-STORM FW-XL System 000
PS-1228; RB Floor Evaluation for HI-STORM FW-XL System; Revision
PS-1232; Lateral Displacement of Suspended HI-TRAC Cask in Spent Fuel Pool; Revision 000 VCT DOC PKG - 2601 PB #1734 Test Results

Miscellaneous
10 CFR 72.212 Evaluation Report for the HI-STORM FW XL MPC Storage System; Revision 0 50.54(q) Program Evaluation / Assessment Review; EP-AA-1007 Addendum 3; Revision 9 OU-AA-630, Dry Cask Storage Program Implementation; Revision 11 PBAPS ISFSI Fire Hazards Analysis for the HI-STORM FW MPC Storage System; Revision 0 TN-68 10 CFR 72.212 Evaluation Report; Revision 18

Procedures
EP-AA-1007; Emergency Action Levels for Peach Bottom Atomic Power Station; Revision 9 LS-AA-114; Exelon 72.48 Review Process; Revision 3
OU-PB-630-206; Radiation Protection Requirements for Holtec HI-STORM FW / MPC Loading and Transport Operations; dated March 3, 2020
RP-AA-305; Holtec HI-TRAC Radiation Survey; Revision 3
RP-AA-306; Holtec HI-STORM Radiation Survey; Revision 1
RP-AA-307; Holtec ISFSI Radiation Survey; Revision 2

LIST OF ACRONYMS USED

CAP Corrective Action Program CoC Certificate of Compliance CFR Code of Federal Regulations EC Engineering Change

Exelon Exelon Generation Company, LLC
FSAR Final Safety Analysis Report
ISFSI Independent Spent Fuel Storage Installation MPC Multi-Purpose Canister
NRC U.S. Nuclear Regulatory Commission PBAPS Peach Bottom Atomic Power Station
PHE Public Health Emergency
SER Safety Evaluation Report
SPMT Self-Propelled Motorized Transporter
TS Technical Specifications
UFSAR Updated Final Safety Analysis Report

January 12, 2021 – Letter from Donald E. Jackson, Chief Operations Branch Division of Reactor Safety to David P. Rhoades Senior Vice President Exelon Generation Company, LLC President and Chief Nuclear Officer Exelon Nuclear with subject of PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 – REQUALIFICATION PROGRAM INSPECTION

In a telephone conversation on January 5, 2021, Mr. T. Fish, Senior Operations Engineer, and Mr. B. Woodard, Licensed Operator Requalification Lead, made arrangements for the U.S. Nuclear Regulatory Commission (NRC) to inspect the licensed operator requalification program at the Peach Bottom facility. The inspection is planned for the week of March 8, 2021, which coincides with your scheduled requalification examination cycle. The staff at your facility should prepare and conduct the requalification examinations in accordance with your NRC-approved requalification program.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 55.59(c), the NRC may request facility licensees to submit their biennial comprehensive requalification written examinations or annual operating tests as necessary to support the NRC's inspection program needs. In order for the NRC to adequately prepare for this inspection, please furnish: (1) an index or summary of condition reports written in the past 2 years in which licensed operator errors were determined to be the root or contributing cause; (2) an index or summary of condition reports written in the past 2 years on simulator performance problems; (3) an index or summary of simulator performance test results in the past 2 years; and 4) the examination schedule, sample plan, and operating test scheduled for the week of the inspection to the NRC by March 1, 2021. Mr. Woodard has been advised of this request and provided with the name and address of the NRC lead inspector assigned to this inspection.

This letter contains information collections that are subject to the *Paperwork Reduction Act of 1995* (44 U.S.C. 3501 et seq.). These information collections were approved by the Office of Management and Budget, approval number 3150-0018.

The public reporting burden for this collection of information is estimated to average 4 hours per response, including the time for reviewing instructions, gathering and maintaining the data needed, and completing and reviewing the collection of information. Send comments regarding this burden estimate or any other aspect of these information collections, including suggestions for reducing the burden, to the Records and FOIA/Privacy Services Branch (T-5 F52),

U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001, or by Internet electronic mail to INFOCOLLECTS@NRC.GOV; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0018), Office of Management and Budget, Washington, D.C. 20503.

The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid OMB control number. In accordance with
10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC's document system, Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC's Website at http://www.nrc.gov/reading-rm/adams.html (The Public Electronic Reading Room).

Thank you for your cooperation in this matter. If you have any questions regarding this inspection, please contact Mr. T. Fish at (610) 337-5369.

February 4, 2021 – Letter from Glenn T. Dentel, Chief Engineering Branch 2 Division of Reactor Safety to David P. Rhoades Senior Vice President Exelon Generation Company, LLC President and Chief Nuclear Officer Exelon Nuclear with subject of PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 – NOTIFICATION OF CONDUCT OF A FIRE PROTECTION TEAM INSPECTION

The purpose of this letter is to notify you that U.S. Nuclear Regulatory Commission (NRC) staff will conduct a fire protection team inspection at your Peach Bottom Atomic Power Station,
Units 2 and 3, starting in June 2021. The inspection will be conducted in accordance with Inspection Procedure 71111, Attachment 21N.05, "Fire Protection Team Inspection

(FPTI)," dated June 12, 2019. The inspection team will be led by Mr. Eugene DiPaolo, a Senior Reactor Inspector from the NRC Region I Office.

The inspection will verify that plant structures, systems, and components, and/or administrative controls credited in the approved fire protection program can perform their licensing basis function.

The schedule for the inspection is as follows:

- Information Gathering Visit: May 25 – 27, 2021
- Onsite Inspection: Weeks of June 7 and June 21, 2021

  The purpose of the information gathering visit is to obtain information and documentation needed to support the inspection and to become familiar with the station's fire protection program, fire protection features, post-fire safe shutdown capabilities, and plant layout. During the information gathering visit, the team leader will select the specific samples to be reviewed during the onsite inspection weeks. The information gathering visit may be conducted remotely due to the pandemic situation.

  The enclosure lists the types of documents that will be needed prior to the information gathering visit. Please provide the referenced information to the Region I office by May 14, 2021. Following sample selection, additional documents will be requested specific to those samples. Your cooperation and support during this inspection will be appreciated.

If you have questions concerning this inspection, or the inspection team's information request or logistical needs, please contact Mr. Eugene DiPaolo, Team Leader at 610-337-6959, or via e-mail at eugene.dipaolo@nrc.gov.

This letter does not contain new or amended information collection requirements subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). Existing information collection requirements were approved by the Office of Management and Budget, under Control Number 3150-0011. The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid Office of Management and Budget control number.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with 10 CFR 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Fire Protection Team Inspection Supporting Documentation

If you have any questions regarding this information request, please contact Mr. Eugene DiPaolo at (610) 337-6959 or via e-mail at eugene.dipaolo@nrc.gov.

Electronic format is preferred. If electronic media is made available via an internet-based document management system, then document access must allow inspectors to download, save, and print the documents in the NRC's Regional office. Additionally, word-searchable documents are preferred, if available. Paper records (hard copy) are acceptable. At the end of the inspection, the documents in the team's possession will not be retained.

This document request is based on *typical documents* that a generic plant might have. As such, this generic document request is not meant to imply that any specific plant is required to have all of the listed documents. It is recognized that some documents listed below may not be available for your plant. In addition, the document titles listed below are based on typical industry document names; your plant specific document titles may vary.

Note that following sample selection, additional documents will be requested specific to those samples.

Please provide these documents to the inspection team leader in the Region I Office by May 14, 2021:

1. Post-Fire Safe Shutdown or Alternative Shutdown Analysis
2. List of post-fire safe shutdown components (i.e., safe shutdown equipment list), if not already included in item (1) above
3. Fire Hazards Analysis Report
4. Fire Probabilistic Risk Assessment (Fire PRA) Summary Document. If a Fire PRA is not available, please provide the Individual Plant Examination for External Events (Fire Chapter Only)
5. Fire Protection Program and/or Fire Protection Plan Document(s)
6. Fire Protection Program implementing procedures, if not already included in item (5). This could include procedures for programs such as transient combustible controls, hot work, changes to the fire protection program, etc.
7. Fire Protection Design Basis Document(s), if available
8. List of all fire protection system impact screening reviews for any design changes, modifications, or temporary modifications completed since May 2018 (e.g., a Generic Letter 86-10 review, LS-AA-128 review, etc.). Include a short description and/or title of each review.
9. List of fire protection system, post-fire safe shutdown, or alternative shutdown design changes completed since May 2018. Include a short description and/or title of each change.
10. List of all Generic Letter 86-10 evaluations completed since May 2018. Include a short description and/or title of each evaluation.
11. List of the top 25 highest fire CDF scenarios, if available.
12. List of current fire protection system impairments, including description
13. List of time critical operator actions and associated program procedure
14. Thermal-hydraulic calculation or analysis that determined the time requirements for time- critical operator actions
15. Copy of the Updated Final Safety Analysis Report
16. Copy of the Technical Requirements Manual (sections associated with fire protection as a minimum)

17. Copy of the Corrective Action Program Procedure(s)
18. Copy of station procedures used for fire safe shutdown from inside and outside of the control room (e.g., alarm response procedure, procedure for shutdown from the remote shutdown panel, etc.)
19. List of open and closed condition reports for post-fire safe shutdown or alternative shutdown issues since May 2018 (e.g., issues affecting safe shutdown analysis, fire hazards analysis, safe shutdown operating procedures and/or training, timeline evaluations for operator actions, etc.). Include the issue report number and a brief description.
20. List of open and closed condition reports for fire protection system issues (e.g., fire pumps, detection, suppression, etc.) since May 2018. Include the issue report number and a brief description.
21. List of open and closed condition reports related to the fire brigade or fire drills since May 2018. Include the issue report number and a brief description.
22. Copies of the following condition reports:

4115309 4125405 4122474 4122883 4122970 4129171 4127608 4128972

23. Copies of any self-assessments performed, and corrective action documents generated, in preparation for this fire protection team inspection.
24. Copies of aging management programs applicable to fire protection including, but not limited to, the following:
    - Fire protection
    - Fire water system
    - Aboveground metallic tanks
    - Buried and underground piping and tanks

25. Copies of procedures, work orders, preventive maintenance tasks, or other documents which implement the commitments made as part of the license extension related to fire protection.

February 11, 2021 – Letter from Jennifer Tobin, Project Manager
Plant Licensing Branch I Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to David P. Rhoades Senior Vice President Exelon Generation Company, LLC President and Chief Nuclear Officer Exelon Nuclear with subject of PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 − REGULATORY AUDIT SUMMARY REGARDING LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT TSTF-505, REVISION 2, “PROVIDE RISK-INFORMED EXTENDED COMPLETION TIMES - RITSTF INITIATIVE 4B” (EPID L-2020-LLA-0120)

By letter dated May 29, 2020 (Agencywide Documents Access and Management System Accession No. ML20150A007), Exelon Generation Company, LLC (Exelon) requested an amendment to the Renewed Facility Operating Licenses for Peach Bottom Atomic Power Station, Units 2 and 3, to revise Technical Specifications to adopt risk-informed completion times based on Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, “Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b.”

To support its review, the U.S. Nuclear Regulatory Commission staff conducted a virtual regulatory audit November 9-12, 2020. The staff reviewed documents and held discussions with members of Exelon and its contractors. The regulatory audit summary is enclosed with this letter.

If you have any questions, please contact me at (301) 415-2328 or Jennifer.Tobin@nrc.gov. Sincerely,

OFFICE OF NUCLEAR REACTOR REGULATION REGULATORY AUDIT SUMMARY FOR NOVEMBER 9-12, 2020, AUDIT IN SUPPORT OF LICENSE AMENDMENT REQUEST TO ADOPT TSTF-505 EXELON GENERATION COMPANY, LLC PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 DOCKET NOS. 50-277 AND 50-278

1.0 BACKGROUND

By application dated May 29, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20150A007), Exelon Generation Company, LLC (Exelon, the licensee) submitted a license amendment request (LAR) for Peach Bottom Atomic Power Station, Units 2 and 3 (Peach Bottom). The amendment would revise technical specification (TS) requirements to permit the use of risk-informed completion times for actions to be taken when limiting conditions for operation are not met. The proposed changes are based on Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF [Risk-Informed TSTF] Initiative 4b," dated July 2, 2018 (ADAMS Accession No. ML18183A493). The U.S. Nuclear Regulatory Commission (NRC) issued a final model safety evaluation approving TSTF-505, Revision 2, on November 21, 2018 (ADAMS Package Accession No. ML18269A041).

An audit team, consisting of NRC staff and contractors from the Pacific Northwest National Laboratory (PNNL), conducted a remote regulatory audit to support the review of the LAR on November 9-12, 2020. The purpose of the audit was to gain an understanding of the information needed to support the NRC staff's licensing decision regarding the LAR and to develop requests for additional information (RAIs). The information submitted in support of the LAR is under final review, and any additional information needed to support the LAR review will be formally requested by the staff using the RAI process in accordance with Office of Nuclear Reactor Regulation Office Instruction LIC–101, "License Amendment Review Procedures" (ADAMS Accession No. ML19248C539, not publicly available).

2.0 AUDIT ACTIVITIES

The NRC audit team consisted of staff from the Division of Risk Assessment, Probabilistic Risk Assessment (PRA) Licensing Branches (A, B, and C) and one PNNL contractor. Several NRC observers, including staff from the Division of Operating Reactor Licensing, Division of Engineering, and Division of Safety Systems were in attendance for the audit. Attachment 1 provides the list of attendees from NRC, PNNL, Exelon, and other participants.

The NRC audit team held an entrance meeting on Monday, November 9, 2020, with the licensee's staff and contractors. During the remainder of the audit, the NRC audit team participated in technical discussions with the licensee based on discipline according to the audit plan (ADAMS Accession No. ML20217L346). Technical discussions were focused on the following major areas: PRA, external hazards, fire protection, TSs, electrical engineering, and instrumentation and controls (I&C). The NRC audit team participated in an audit exit meeting

with the licensee on Thursday, November 12, 2020, where the NRC staff provided a brief conclusion of the team's goals, objectives, and technical discussions.

The NRC staff provided a brief conclusion of the audit objectives that were met and details on the path forward. There were no open items in the discussion and no deviation from the audit plan. Exelon committed to providing a supplement to the application to address audit discussion points and potential RAIs. Attachment 2 contains a list of documents reviewed by the team during the audit.

3.0 RESULTS OF THE AUDIT

In response to discussions during the audit, Exelon submitted a supplement to the LAR to the NRC on December 2, 2020 (ADAMS Accession No. ML20337A301). The NRC issued RAIs to the licensee on January 5, 2021, and January 13, 2021 (ADAMS Accession Nos. ML20357A097 and ML21012A130, respectively), with a request to submit the responses by February 5, 2021. [Has Exelon submitted the responses?]

Attachments:

1. List of Participants
2. List of Documents Reviewed During Audit

List of Participants U.S. Nuclear Regulatory Commission (NRC) Audit Team

---

Tobin, Jennifer Hilsmeier, Todd Danna, James Pascarelli, Robert Borromeo, Joshua Cusumano, Vic Titus, Brett

Circle, Jeff
Vettori, Robert
Wu, De Valentin-Olmeda, Milton Russell, Andrea Wyman, Stephen Nguyen, Khoi

Li, Ming
Carte, Norbert Karipineni, Nageswara Wong, Yuken
Bedi, Gurjendra
Wilk, Mark
Biro, Mihaela
Chang, James Dukehart, Corey Hartage, Kayleh Kichline, Michelle Marchlewski, Henry
Patel, Jigar
Tetter, Keith

Acronyms:

Project Manager, NRR/DORL/LPL1
Team Lead, PRA Licensing Reviewer, DRA/APLA Plant Licensing Branch Chief, NRR/DORL/LPL1 PRA Licensing A Branch Chief, DRA/APLA
PRA Licensing B Branch Chief, DRA/APLB Technical Specifications Branch Chief, DSS/STSB Electrical Engineering Branch Chief, DEX/EEEB PRA Licensing Reviewer, DRA/APLA
PRA Licensing Reviewer, DRA/APLB
PRA Licensing Reviewer, DRA/APLC
PRA Licensing Reviewer, DRA/APLC
Technical Specifications Reviewer, DSS/STSB Electrical Engineering Reviewer, DEX/EEEB Electrical Engineering Reviewer, DEX/EEEB Instrumentation & Controls Reviewer, DEX/EICB Instrumentation & Controls Reviewer, DEX/EICB Containment Systems Reviewer, DSS/SCPB Mechanical Engineering Reviewer, DEX/EMIB Mechanical Engineering Reviewer, DEX/EMIB Contractor, PNNL
Observer, DRA/APLA
Observer, RES/HFRB
Observer, NRAN
Observer, NRAN
Observer, DRA/APOB
Observer, NRAN
Observer, DRA/APLA
Observer, DRA/APLC

APLA – PRA Licensing Branch A; APLB – PRA Licensing Branch B; APLC – PRA Licensing Branch C; APOB – PRA Oversight Branch; DEX – Division of Engineering and External Hazards; DORL – Division of Operating Reactor Licensing; DRA – Division of Risk Assessment; DSS – Division of Safety Systems; EEEB – Electrical Engineering Branch; EICB – Instrumentation & Controls Branch; EMIB – Mechanical Engineering and Inservice Testing Branch; HFRB – Human Factors and Reliability Branch; LPL1 – Plant Licensing Branch 1; NRAN – Nuclear Regulator Apprenticeship Network Program; PNNL – Pacific Northwest National Laboratory; RES – Office of Nuclear Regulatory Research; SCPB – Containment and Plant Systems Branch; STSB – Technical Specifications Branch.

Attachment 1

Exelon Generation Company, LLC Participants

Licensing:

Shannon Rafferty-Czincila, Dir. Rick Gropp, Engr.
Glenn Stewart, Engr.

PEA Site:

Matthew Rector, Reg Assurance, Mgr. Mike Smith, I&C Support Engr.
Brian Wright, OPS, SSV (RTR)
Ross Moonitz, Sr, OPS Training Instr. Victor Molina, OPS, SSV (CAL RTR)

Risk Management:

Jeff Stone, Director.
Gene Kelly, Sr. Mgr.
Suzanne Loyd, Sr. Mgr.
Rachelle Slawta, Engr., RICT RTR Phil Tarpinian, Engr

Corporate Eng:

Jenna Burr, Engr, MRule SME Scott Diven, Engr, MRule SME

Jensen Hughes 10 CFR 50.69 Support Team

Nick Sternowski, Director, PM
Leo Shanley, RM, Mgr.
Ed Parsley, RM, Mgr., R-I Services Arthur Holtz, RM, FRME (lead)
Eric Heilman, RM, SRME
Zach Ballert, RM, co-SRME
Vicki Warren, RM, Engr.
Jon Facemire, RM, RICT RTR SME Brian Albinson, RM, FPIE M.O.
Greg Zucal, RM, FPRA M.O.
Charlie Young, RM, Engr
Jeff Schappaugh, RM, FRME
Dave Passehl, RM, Engr
Ben Chen, RM, Engr
Lynn Kolonauski, RM, HRA Analyst Larry Lee, RM, Seismic SME
Vincent Andersen, RM, Seismic SME Don MacLeod, RM, HRA Analyst

RM = Risk Management (PRA) SRME = Site Risk Mgmt Engr. FRME = Fleet Risk Mgmt Engr MO = Model Owner

List of Documents Reviewed During the Audit

The licensee made available for review an extensive list of supporting documents (e.g., analyses, calculations, reports, drawings, and procedures) on the Peach Bottom document portal during the week of the audit.

Application Specific Documents

- - Exelon Report PB-LAR-017, Revision 2, "Peach Bottom Atomic Power Station, Units 2 and 3, PRA Application Notebook, RICT Estimates for TSTF-505 (RICT) Program LAR Submittal."
- - Exelon Report PB-ASM-15, Revision 0, "Peach Bottom Atomic Power Station, Probabilistic Risk Assessment, Application-Specific Model (ASM)."
- - Exelon Report PB-ASM-20, Revision 0, "Peach Bottom Atomic Power Station, Probabilistic Risk Assessment, Application-Specific Model (ASM)."

- \- Exelon Report PB-MISC-045, Revision 0, "Peach Bottom 10 CFR 50.69 NRC Safety Evaluation LAR Implementation Items."

Internal Events PRA

- \- Exelon Report PB-PRA-013, Revision 6, "Peach Bottom Atomic Power Station, Probabilistic Risk Assessment, Summary Notebook PBB218A2 and PB318A2 Models."
- \- Exelon Report PB-MISC-043, Revision 1, "Assessment of Key Assumptions and Sources of Uncertainty for the Peach Bottom Atomic Power Station PRA."
- \- Exelon Report PB-PRA-004, Revision 5, "Peach Bottom Atomic Power Station, Probabilistic Risk Assessment, Human Reliability Analysis Notebook Volume 1."
- \- Exelon Report PB-PRA-005.25, Revision 0, "Peach Bottom Atomic Power Station, Probabilistic Risk Assessment, Portable Equipment (FLEX) System Notebook."
- \- Exelon Report PB-PRA-005.07, Revision 3, "Peach Bottom Atomic Power Station,

  Probabilistic Risk Assessment, Offsite & 13kV AC (13kV) System Notebook."

- \- Exelon Report PB-PRA-005.08, Revision 3, "Peach Bottom Atomic Power Station,

  Probabilistic Risk Assessment, 4kV and 480V AC (4kV/480V) System Notebook."

- \- Exelon Report PB-PRA-005.11, Revision 4, "Peach Bottom Atomic Power Station,

  Probabilistic Risk Assessment, Emergency Service Water (ESW) Emergency Cooling

  Water (ECW) System Notebook."

- \- Exelon Report PB-PRA-005.12, Revision 4, "Peach Bottom Atomic Power Station,

  Probabilistic Risk Assessment, High Pressure Service Water (HPSW) System

  Notebook."

- \- Exelon Report PB-PRA-005.15, Revision 3, "Peach Bottom Atomic Power Station,

  Probabilistic Risk Assessment, Instrument Air/Service Air (IA/SA) System Notebook."

- \- Exelon Report PB-PRA-005.17, Revision 4, "Peach Bottom Atomic Power Station, Probabilistic Risk Assessment, Emergency Diesel Generator (EDG) System Notebook."

- - Exelon Report PB-PRA-010, Revision 4, "Peach Bottom Atomic Power Station,

  Probabilistic Risk Assessment, Component Data Notebook, Volume 1, 2018 PRA Update."

  Fire PRA

- Exelon Report PB-PRA-021.62, Revision 2, "Peach Bottom Atomic Power Station, Fire Probabilistic Risk Assessment, Uncertainty and Sensitivity Notebook."

- - Exelon Report PB-PRA-021.61, Revision 2, "Peach Bottom Atomic Power Station, Fire Probabilistic Risk Assessment, Summary & Quantification Notebook."
- - Exelon Report PB-PRA-021.56, Revision 2, "Peach Bottom Atomic Power Station, Fire Probabilistic Risk Assessment, Fire Ignition Frequency Notebook."
- - Exelon Report PB-PRA-021.57.01, Revision 2, "Peach Bottom Atomic Power Station, Fire Probabilistic Risk Assessment, Fire Scenario Development Notebook."
- - Exelon Report PB-PRA-021.57.02, Revision 2, "Peach Bottom Atomic Power Station, Fire Probabilistic Risk Assessment, Fire Modeling Treatments Notebook."
- - Exelon Report PB-PRA-021.57.05, Revision 2, "Peach Bottom Atomic Power Station, Fire Probabilistic Risk Assessment, Control Room Fire Modeling Notebook."
- - Exelon Report PB-PRA-021.59, Revision 1, "Peach Bottom Atomic Power Station, Fire Probabilistic Risk Assessment, Human Reliability Analysis Notebook, Volume 1."
- - Exelon Report PB-PRA-021.59, Revision 2, "Peach Bottom Atomic Power Station, Fire Probabilistic Risk Assessment, Human Reliability Analysis Notebook, Volume 2."

PRA Acceptability

- - Exelon Report 032299-RPT-001, Revision 1, "Risk Management Finding Level F&O Technical Review & Focused-Scope Peer Review, Peach Bottom Atomic Power Station, Units 2 and 3."
- - Exelon Report 032434-RPT-02, Revision 1, "Peach Bottom Nuclear Generating Station FPIE and Fire PRA Finding Level Fact and Observation Closure by Independent Assessment."
- - Exelon Report 023424-RPT-03, Revision 0, "Peach Bottom PRA Focused-Scope Peer Review."
- - BWR Owners Group Report, "Peach Bottom Atomic Power Station PRA Peer Review Report Using ASME PRA Standard Requirements," dated May 2011.
- - BWR Owners Group Report, "Peach Bottom Atomic Power Station (PB), Unit 2 Fire PRA Peer Review Report Using ASME/ANS PRA Standard Requirements," dated April 2013.

- - Exelon Report PB-MISC-046, Revision 0, “Peer Review Findings Closing Summary.” External Hazards

- Exelon Report PB-MISC-027, Revision 5, “External Hazards Assessment for Peach Bottom Atomic Power Station.”

Plant Procedures

- - Exelon Procedure OP-AA-108-118, Revision 1, “Risk-Informed Completion Time.”
- - Exelon Procedure ER-AA-600-1014, Revision 8, “Risk Management Configuration

  Control.”

- - Exelon Procedure ER-AA-600-1015, Revision 20, “FPIE PRA Model Update.”
- - Exelon Procedure ER-AA-320-1004, Revision 01, “Maintenance Rule 18-10-

  Performance Monitoring and Dispositioning Between (a)(1) and (a)(2).”

- - Exelon Procedure ER-AA-320-1007, Revision 00, “Maintenance Rule 18-10-Periodic

  (a)(3) Assessment.”

- - Exelon Procedure ER-AA-600-1042, Revision 12, “On-line Risk Management.”
- - Exelon Procedure OP-AA-108-117, Revision 5, “Protected Equipment Program.”
- - Exelon Procedure OP-AA-108-118, Revision 2, “Risk Informed Completion Time.”
- - Exelon Procedure OP-PB-102-106, Revision 10, “Operator Response Time Program at

  Peach Bottom.”

PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 − REGULATORY AUDIT SUMMARY REGARDING LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT TSTF-505, REVISION 2, “PROVIDE RISK-INFORMED EXTENDED COMPLETION TIMES - RITSTF INITIATIVE 4B” (EPID L-2020-LLA-0120) DATED FEBRUARY 11, 2021

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ADAMS Accession No.: ML21026A289

OFFICE NRR/DORL/LPL1/PM NRR/DORL/LPL1/LA NRR/DORL/LPL1/BC NRR/DORL/LPL1/PM
NAME JTobin JBurkhardt JDanna JTobin
DATE 1/27/2021 1/28/2021 2/11/21 2/11/21

OFFICIAL RECORD COPY

March 24, 2021 – Email from Blake Purnell to Bradley Fewell (Exelon) cc to Nancy Salgado and David Helker (Exelon) with subject of Exelon Generation Company, LLC - Acceptance of License Transfer Application (EPID L-2021-LLM-0000)

By letter dated February 25, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21057A273), Exelon Generation Company, LLC (EGC) on behalf of itself and Exelon Corporation; Exelon FitzPatrick, LLC; Nine Mile Point Nuclear Power Station, LLC; R. E. Ginna Nuclear Power Plant, LLC; and Calvert Cliffs Nuclear Power Plant, LLC (collectively, the Applicants) requested that the U.S. Nuclear Regulatory Commission (NRC) consent to the indirect transfer of control of the following licenses (collectively, the licenses):

- Renewed Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2, respectively;
- Renewed Facility Operating License Nos. NPF-37 and NPF-66 for Byron Station, Unit Nos. 1 and 2, respectively;
- Renewed Facility Operating License Nos. DPR-53 and DPR-69 for Calvert Cliffs Nuclear Power Plant (Calvert Cliffs), Units 1 and 2, respectively;
- Facility Operating License No. NPF-62 for Clinton Power Station, Unit No. 1;
- Facility Operating License No. DPR-2 and Renewed Facility Operating License Nos. DPR-19 and

  DPR-25 for Dresden Nuclear Power Station, Units 1, 2, and 3, respectively;

- Renewed Facility Operating License No. DPR-59 for James A. FitzPatrick Nuclear Power Plant

  (FitzPatrick);

- Renewed Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station, Units 1

  and 2, respectively;

- Renewed Facility Operating License Nos. NPF-39 and NPF-85 for Limerick Generating Station,

  Units 1 and 2, respectively;

- Renewed Facility Operating License Nos. DPR-63 and NPF-69 for Nine Mile Point Nuclear Station

(NMP), Units 1 and 2, respectively;

- Facility Operating License No. DPR-12 and Subsequent Renewed Facility Operating License Nos.

  DPR-44 and DPR-56 for Peach Bottom Atomic Power Station, Units 1, 2, and 3, respectively;

- Renewed Facility Operating License Nos. DPR-29 and DPR-30 for Quad Cities Nuclear Power

  Station, Units 1 and 2, respectively;

- Renewed Facility Operating License No. DPR-18 for R. E. Ginna Nuclear Power Plant (Ginna);
- Renewed Facility Operating License Nos. DPR-70 and DPR-75 for Salem Nuclear Generating

  Station, Unit Nos. 1 and 2, respectively;

- Renewed Facility License No. DPR-50 for Three Mile Island Nuclear Station, Unit 1;
- Facility Operating License Nos. DPR-39 and DPR-48 for Zion Nuclear Power Station (Zion), Units 1

  and 2, respectively;

- Materials License No. SNM-2505 for the independent spent fuel storage installation (ISFSI) at

  Calvert Cliffs; and

- the general licenses for the ISFSIs at the other facilities.

Specifically, the Applicants requested that the NRC consent to the indirect transfer of control of the licenses to support a proposed transaction in which Exelon Corporation will transfer its 100 percent ownership of EGC to a newly-created subsidiary that will then be spun off to Exelon Corporation shareholders, becoming EGC’s new ultimate parent company. Once the spin transaction is completed, the new ultimate parent company, EGC (operating under a new name), and its subsidiaries will no longer be affiliated with Exelon Corporation. EGC will remain the same Pennsylvania limited liability company and will continue to own and/or operate the facilities, as applicable, and hold the licenses, but it will be renamed and reorganized. The Applicants also requested that the NRC consent to the indirect transfer of control of the licenses for FitzPatrick, NMP, Ginna, and their ISFSIs to support the reorganization of EGC. In addition, the Applicants requested conforming amendments to the specific licenses to reflect the proposed transfer.

The Applicants also requested NRC approval to replace existing nuclear operating services agreements and financial support agreements associated with the ownership and operation of Calvert Cliffs, NMP, Ginna, and FitzPatrick. The application requested NRC approval to transfer the qualified and non- qualified trusts for FitzPatrick from Exelon Generation Consolidation, LLC (a subsidiary of EGC) to the renamed Exelon FitzPatrick, LLC. The application requested amendments to the NMP, Ginna, and Calvert Cliffs licenses to delete conditions referencing the Constellation Energy Nuclear Group, LLC (a subsidiary of EGC) Board and its operating agreement to reflect the internal reorganization of EGC described in the application.

By Order dated November 26, 2019 (ADAMS Accession No. ML19228A130), as modified by Order dated October 21, 2020 (ADAMS Accession No. ML20259A469), the NRC authorized the direct transfer of Facility Operating License Nos. DPR-39 and DPR-48 for Zion, Units 1 and 2, respectively, and Zion's generally licensed ISFSI from ZionSolutions, LLC to EGC. According to the February 25, 2021, application, the Zion license transfer will be completed prior to the spin transaction.

The purpose of this e-mail is to provide the results of the NRC staff's acceptance review of the February 25, 2021, license transfer request. The acceptance review was performed to determine if there is sufficient technical information in scope and depth to allow the NRC staff to complete its detailed technical review. The acceptance review is also intended to identify whether the application has any readily apparent information insufficiencies in its characterization of the regulatory requirements or the licensing bases of the plants.

Consistent with Sections 50.80 and 72.50 of Title 10 of the Code of Federal Regulations (10 CFR), an application for transfer of a license shall include as much of the information as described in 10 CFR 50.33, 50.34, 72.22, and 72.28, as applicable, with respect to identity and technical and financial qualifications of the proposed transferee. Consistent with 10 CFR 50.90 and 72.56, as applicable, an amendment to the license must fully describe the changes requested, and following as far as applicable, the form prescribed for original applications. Sections 50.34 and 72.24 of 10 CFR address the content of technical information required for licenses issued under 10 CFR Parts 50 and 72, respectively.

The NRC staff has reviewed your application and concluded that it does provide technical information in sufficient detail to enable the NRC staff to complete its detailed technical review and make an independent assessment regarding the acceptability of the proposed amendment in terms of regulatory requirements and the protection of public health and safety and the environment. Given the lesser scope and depth of the acceptance review as compared to the detailed technical review, there may be instances in which issues that impact the NRC staff's ability to complete the detailed technical review

are identified despite completion of an adequate acceptance review. You will be advised of any further information needed to support the NRC staff's detailed technical review by separate correspondence.

Based on the information provided in your submittal, the NRC staff has estimated that the review of this licensing request will take approximately 700 hours to complete. The NRC staff expects to complete this review by November 30, 2021. If there are emergent complexities or challenges in our review that would cause changes to the initial forecasted completion date or significant changes in the forecasted hours, the reasons for the changes, along with the new estimates, will be communicated during the routine interactions with the assigned project manager. These estimates are based on the NRC staff's initial review of the application and they could change, due to several factors including requests for additional information, unanticipated addition of scope to the review, or hearing-related activities.

If you have any questions, please contact me at (301) 415-1380. Sincerely,

Blake Purnell, Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation U.S. Nuclear Regulatory Commission

Docket Nos. STN 50-456, STN 50-457, 72-73, STN 50-454, STN 50-455, 72-68, 50-317, 50-318, 72-8, 50- 461, 72-1046, 50-010, 50-237, 50-249, 72-37, 50-333, 72-12, 50-373, 50-374, 72-70, 50-352, 50-353, 72-65, 50-220, 50-410, 72-1036, 50-171, 50-277, 50-278, 72-29, 50-254, 50-265, 72-53, 50-244, 72-67, 50-272, 50-311, 72-48, 50-289, 72-77, 50-295, 50-304, and 72-1037

April 5, 2021 – Letter from Blake Purnell, Project Manager Plant Licensing Branch III Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to David P. Rhoades Senior Vice President Exelon Generation Company, LLC President and Chief Nuclear Officer (CNO) Exelon Nuclear with subject of EXELON GENERATON COMPANY, LLC – REQUEST FOR WITHHOLDING INFORMATION FROM PUBLIC DISCLOSURE (EPID L-2021-LLM-0000)

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated February 25, 2021, Exelon Generation Company, LLC (EGC)[1] submitted an affidavit dated February 25, 2021, executed by Bryan P. Wright, Senior Vice President and Chief Financial Officer of EGC, requesting that information contained in Enclosures 6A, 8A, and 10A to the letter be withheld from public disclosure pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 2.390. The letter, with all enclosures except Enclosures 6A, 8A, and 10A, has been made publicly available in the Agencywide Documents Access and Management System (ADAMS) under Accession No. ML21057A273. The letter also provided nonproprietary versions of Enclosures 6A, 8A,

and 10A, as Enclosures 6, 8, and 10, respectively.

The affidavit stated that the information provided in Enclosures 6A, 8A, and 10A constitutes proprietary commercial and financial information that should be considered exempt from mandatory public disclosure because:

i. This information is and has been held in confidence by [the] Applicants.
ii. This information is of a type that is customarily held in confidence by the Applicants, and there is a rational basis for doing so because the information contains sensitive financial information.

iii. This information is being transmitted to the NRC voluntarily and in confidence.
iv. This information is not available in public sources and could not be gathered readily from other publicly available information.
v. Public disclosure of this information would create substantial harm to the competitive position of the Applicants by disclosing their internal data.

[1] On behalf of itself and Exelon Corporation; Exelon FitzPatrick, LLC; Nine Mile Point Nuclear Power Station, LLC; R. E. Ginna Nuclear Power Plant, LLC; and Calvert Cliffs Nuclear Power Plant, LLC (collectively, the Applicants).

We have reviewed your letter and the material in accordance with the requirements of 10 CFR 2.390 and, based on the statements in the affidavit, have determined that the submitted information sought to be withheld contains proprietary commercial information and should be withheld from public disclosure. Therefore, the information marked as proprietary will be withheld from public disclosure pursuant to 10 CFR 2.390(b)(5) and Section 103(b) of the Atomic Energy Act of 1954, as amended.

Withholding from public inspection shall not affect the right, if any, of persons properly and directly concerned to inspect the documents. If the need arises, we may send copies of this information to our consultants working in this area. We will, of course, ensure that the consultants have signed the appropriate agreements for handling proprietary information.

If the basis for withholding this information from public inspection should change in the future such that the information could then be made available for public inspection, you should promptly notify the NRC. You also should understand that the NRC may have cause to review this determination in the future, for example, if the scope of a Freedom of Information Act request includes your information. In all review situations, if the NRC makes a determination adverse to the above, you will be notified in advance of any public disclosure.

If you have any questions regarding this matter, I may be reached at 301-415-1380.

April 28, 2021 – Letter from Chris M. Lally, Acting Chief Reactor Projects Branch 4 Division of Operating Reactor Safety to David P. Rhoades Senior Vice President Exelon Generation Company, LLC President and Chief Nuclear Officer Exelon Nuclear with subject of PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 – INTEGRATED INSPECTION REPORT 05000277/2021001 AND 05000278/2021001

On March 31, 2021, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Peach Bottom Atomic Power Station, Units 2 and 3. On April 16, 2021, the NRC inspectors discussed the results of this inspection with Mr. Matthew Herr, Site Vice President, and other members of your staff. The results of this inspection are documented in the enclosed report.

One finding of very low safety significance (Green) is documented in this report. This finding involved a violation of NRC requirements. One Severity Level IV violation without an associated finding is documented in this report. We are treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2 of the Enforcement Policy.

If you contest the violations or the significance or severity of the violations documented in this inspection report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement; and the NRC Resident Inspector at Peach Bottom Atomic Power Station, Units 2 and 3.

If you disagree with a cross-cutting aspect assignment in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; and the NRC Resident Inspector at Peach Bottom Atomic Power Station, Units 2 and 3.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with Title 10 of the *Code of Federal Regulations* 2.390, “Public Inspections, Exemptions, Requests for Withholding.”

U.S. NUCLEAR REGULATORY COMMISSION Inspection Report

Docket number: 05000277 and 05000278
License numbers: DPR-44 and DPR-56
Report numbers: 05000277/2021001 and 05000278/2021001

Enterprise Identifier: I-2021-001-0083
Licensee: Exelon Generation Company, LLC
Facility: Peach Bottom Atomic Power Station, Units 2 and 3

Location: Delta, PA 17314
Inspection dates: January 1, 2021 to March 31, 2021

Inspectors: S. Rutenkroger, Senior Resident Inspector

P. Boguszewski, Resident Inspector
J. Brand, Reactor Inspector
N. Mentzer, Reactor Inspector

J. Rady, Emergency Preparedness Inspector

Approved by: Chris M. Lally, Acting Chief Reactor Projects Branch 4
Division of Operating Reactor Safety

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting an integrated inspection at Peach Bottom Atomic Power Station, Units 2 and 3, in accordance with the Reactor Oversight Process. The Reactor Oversight Process is the NRC's program for overseeing the safe operation of commercial nuclear power reactors. Refer to https://www.nrc.gov/reactors/operating/oversight.html for more information.

List of Findings and Violations

High-Pressure Service Water Valve Failed to Open			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000277/2021001-01 Open/Closed	[H.12] - Avoid Complacency	71111.15
A self-revealing Green non-cited violation (NCV) of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," because Exelon did not accomplish work in accordance with instructions during maintenance and staging activities. Specifically, during planned work staging, Exelon personnel did not effectively prevent contact with plant equipment which resulted in unplanned unavailability of high-pressure service water (HPSW) to the 'D' residual heat removal (RHR) heat exchanger.			

Degraded Condition Due to Reactor Pressure Vessel Instrument Nozzle Leakage			
Cornerstone	Severity	Cross-Cutting Aspect	Report Section
Not Applicable	Severity Level IV NCV 05000277/2021001-02 Open/Closed	Not Applicable	71153
A Severity Level IV NCV of Peach Bottom Atomic Power Station, Unit 2, Technical Specification (TS) 3.4.4, "Reactor Coolant System (RCS) Operational Leakage," was self-revealed when leakage was identified from the 'N16A' 2 inch reactor pressure vessel (RPV) instrument nozzle during a RPV pressure test on October 29, 2020. Specifically, RCS pressure boundary leakage reasonably began on an unknown date that was more than 36 hours before the shutdown for the refueling outage on October 21, 2020.			

Additional Tracking Items

Type	Issue Number	Title	Report Section	Status
LER	05000277/2020-002-00	Licensee Event Report (LER) 2020-002-00 for Peach Bottom Atomic Power Station, Unit 2, Degraded Condition due to RPV Instrument Nozzle Leakage	71153	Closed

PLANT STATUS

Unit 2 began the inspection period at rated thermal power (RTP). On March 5, 2021, the unit was down powered to 75 percent for a planned control rod pattern adjustment, turbine valve testing, and general unit maintenance. The unit was returned to RTP the following day. The unit remained at or near RTP for the remainder of the inspection period.

Unit 3 began the inspection period at RTP. On February 25, 2021, the '3C' reactor feedwater pump tripped, and the unit was down powered to 60 percent. New control cabling for the pump was installed and re-routed, the pump was returned to service, and the unit was returned to RTP on February 27, 2021. The unit remained at or near RTP for the remainder of the inspection period.

INSPECTION SCOPES

Inspections were conducted using the appropriate portions of the inspection procedures (IPs) in effect at the beginning of the inspection unless otherwise noted. Currently approved IPs with their attached revision histories are located on the public website at http://www.nrc.gov/reading- rm/doc-collections/insp-manual/inspection-procedure/index.html. Samples were declared complete when the IP requirements most appropriate to the inspection activity were met consistent with Inspection Manual Chapter (IMC) 2515, “Light-Water Reactor Inspection Program - Operations Phase.” The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel to assess licensee performance and compliance with Commission rules and regulations, license conditions, site procedures, and standards. Starting on March 20, 2020, in response to the National Emergency declared by the President of the United States on the public health risks of the coronavirus (COVID-19), resident and regional inspectors were directed to begin telework and to remotely access licensee information using available technology. During this time the resident inspectors performed periodic site visits each week, increasing the amount of time on site as local COVID-19 conditions
permitted. As part of their onsite activities, resident inspectors conducted plant status activities as described in IMC 2515, Appendix D; observed risk significant activities; and completed on site portions of IPs. In addition, resident and regional baseline inspections were evaluated to determine if all or portion of the objectives and requirements stated in the IP could be performed remotely. If the inspections could be performed remotely, they were conducted per the applicable IP. In some cases, portions of an IP were completed remotely and on site. The inspections documented below met the objectives and requirements for completion of the IP.

REACTOR SAFETY

71111.01 - Adverse Weather Protection
Seasonal Extreme Weather Sample (IP Section 03.01) (1 Sample)

(1) The inspectors evaluated readiness for seasonal extreme weather conditions prior to the onset of seasonal cold temperatures for the following systems: emergency diesel generators (EDGs), HPSW, and emergency service water (ESW) during the week of January 10, 2021

Impending Severe Weather Sample (IP Section 03.02) (1 Sample)

(1) The inspectors evaluated the adequacy of the overall preparations to protect risk-significant systems from potential severe weather given a winter storm warning from January 26, 2021 to February 2, 2021

71111.04 - Equipment Alignment
Partial Walkdown Sample (IP Section 03.01) (5 Samples)

The inspectors evaluated system configurations during partial walkdowns of the following systems/trains:

1. (1) Unit 2 'A' RHR while 'B' RHR was inoperable due to a failed surveillance test on January 14, 2021
2. (2) Unit 2 'A' standby liquid control during 'E-2' EDG maintenance on February 5, 2021
3. (3) Unit common, 'A' ESW during 'B' ESW maintenance on February 23, 2021
4. (4) Unit 3 'B' core spray during 'C' core spray maintenance on March 26, 2021
5. (5) Unit common, 'E-4' EDG during 'E-2' EDG maintenance on March 24, 2021

71111.05
Fire Area Walkdown and Inspection Sample (IP Section 03.01) (6 Samples)

The inspectors evaluated the implementation of the fire protection program by conducting a walkdown and performing a review to verify program compliance, equipment functionality, material condition, and operational readiness of the following fire areas:

1. (1) PF-132, diesel generator building general area on January 6, 2021
2. (2) PF-27, Unit 3 north isolation valve room on February 24, 2021
3. (3) PF-31, Unit 3 south isolation valve room on February 24, 2021
4. (4) PF-132, diesel generator building general area on February 17, 2021
5. (5) PF-1, Unit 2 'A' RHR on March 11, 2021
6. (6) PF-13C, Unit 3 reactor building torus room on March 16, 2021

Fire Brigade Drill Performance Sample (IP Section 03.02) (1 Sample)

(1) The inspectors evaluated the onsite fire brigade training and performance during an unannounced fire drill on March 4, 2021

71111.06 - Flood Protection Measures
Inspection Activities - Internal Flooding (IP Section 03.01) (1 Sample)

The inspectors evaluated internal flooding mitigation protections in the: (1) Unit 3 torus room during the week of March 22, 2021

71111.11Q - Licensed Operator Requalification Program and Licensed Operator Performance

Licensed Operator Performance in the Actual Plant/Main Control Room (IP Section 03.01) (1 Sample)

(1) The inspectors observed and evaluated licensed operator performance in the main control room during a power reduction to 55 percent for Unit 3 'C' reactor feed pump repairs on February 26, 2021

Licensed Operator Requalification Training/Examinations (IP Section 03.02) (1 Sample)

(1) The inspectors observed and evaluated licensed operator requalification training in the simulator on January 25, 2021

71111.12 - Maintenance Effectiveness
Maintenance Effectiveness (IP Section 03.01) (2 Samples)

The inspectors evaluated the effectiveness of maintenance to ensure the following structures, systems, and components (SSCs) remain capable of performing their intended function:

1. (1) Unit common, 'E-2' EDG as of February 13, 2021
2. (2) Unit 2 instrument air system during the week of March 15, 2021

Quality Control (IP Section 03.02) (1 Sample)

The inspectors evaluated the effectiveness of maintenance and quality control activities to ensure the following SSC remains capable of performing its intended function:

(1) Unit common, 'E-1' EDG maintenance overhaul from January 31 to February 5, 2021
71111.13 - Maintenance Risk Assessments and Emergent Work Control
Risk Assessment and Management Sample (IP Section 03.01) (4 Samples)

The inspectors evaluated the accuracy and completeness of risk assessments for the following planned and emergent work activities to ensure configuration changes and appropriate work controls were addressed:

(1) Unit common, 'E-2' EDG planned maintenance on January 27, 2021 (2) Unit 2 'D' HPSW planned valve replacement on February 11, 2021 (3) Unit 2 'C' RHR planned maintenance on March 3, 2021
(4) Unit 2 'A' and 'C' RHR planned relay testing on March 17, 2021

71111.15 - Operability Determinations and Functionality Assessments
Operability Determination or Functionality Assessment (IP Section 03.01) (7 Samples)

The inspectors evaluated the licensee's justifications and actions associated with the following operability determinations and functionality assessments:

1. (1) Unit 2 high-pressure coolant injection (HPCI) leak from the pressure sensing line to the cooling water header pressure control value on December 29, 2020
2. (2) Unit 2 low-pressure coolant injection (LPCI) max light did not illuminate, and separately the LPCI min light did not extinguish, within the band required by procedure on January 13 and 15, 2021
3. (3) Unit common, review of multiple switch failures for potential common cause failure mode on January 17, 2021
4. (4) Unit 2 'D' HPSW outlet valve for the 'B' loop RHR heat exchanger did not open on February 5, 2021

5. (5) Unit common, FLEX diesel generators staged without heaters energized during the week of February 8, 2021
6. (6) Unit 2 'A' station batteries during battery cell replacements on February 22, 2021
7. (7) Unit 3 reactor core isolation cooling (RCIC) low lube oil cooling water pressure on

   March 2, 2021

71111.19
Post-Maintenance Test Sample (IP Section 03.01) (5 Samples)

The inspectors evaluated the following post-maintenance test activities to verify system operability and functionality:

1. (1) Unit 3 wide range neutron monitor period recorder 'NR-03-07-049B' testing after replacement on January 20, 2021
2. (2) Unit 2 'B' RHR control valve 'CV-2-10-2677B' limit switch testing after LPCI max light issues and maintenance on January 22, 2021
3. (3) Unit common, 'E-2' EDG testing after the two-year maintenance overhaul on February 10, 2021
4. (4) Unit common, EDGs CARDOX fire suppression system leak testing after valve replacements on February 17, 2021
5. (5) Unit 2 drywell purge valve 'AO-2-07B-2520' leak testing after installation of a nitrogen bottle modification on February 17 and February 18, 2021

- Post-Maintenance Testing

71111.22
The inspectors evaluated the following surveillance tests: Surveillance Tests (other) (IP Section 03.01) (3 Samples)

1. (1) Unit 3 'A' and 'C' LPCI pump start time delay relay calibration on January 5, 2021
2. (2) Unit 3 'A' RHR pump, valve, flow, and unit cooler test on March 1, 2021
3. (3) Unit 3 RCIC pump, valve, and flow test on March 2, 2021

Inservice Testing (IP Section 03.01) (3 Samples)

1. (1) Unit 3 'B' RHR pump, valve, flow, and unit cooler test on January 15, 2021
2. (2) Unit 3 'C' RHR room unit cooler test on January 19, 2021
3. (3) Unit common 'E-2' EDG simulated Unit 3 emergency core cooling system signal auto

   start test on February 24, 2021
   6

- Surveillance Testing

71114.02 - Alert and Notification System Testing Inspection Review (IP Section 02.01-02.04) (1 Sample)

(1) The inspectors evaluated Exelon's maintenance and testing of the Peach Bottom Atomic Power Station Alert and Notification System on February 8–11, 2021, for the period of March 2019 through January 2021

71114.03 - Emergency Response Organization Staffing and Augmentation System Inspection Review (IP Section 02.01-02.02) (1 Sample)

(1) The inspectors evaluated the readiness of Exelon's Emergency Preparedness organization on February 8–11, 2021

71114.04 - Emergency Action Level and Emergency Plan Changes Inspection Review (IP Section 02.01-02.03) (1 Sample)

(1) The inspectors evaluated the following submitted Emergency Action Level and Emergency Plan changes onsite on February 8–11, 2021:

- Evaluation 18-86, EP-AA-1007, "Exelon Nuclear Radiological Emergency Plan Annex for Peach Bottom Atomic Power Station," Revision 34
- Evaluation 19-60, EP-AA-1007 Addendum 1, "Peach Bottom Atomic Power Station On-Shift Staffing Technical Basis," Revision 2
- Evaluation 19-66, EP-AA-1007, "Exelon Nuclear Radiological Emergency Plan Annex for Peach Bottom Atomic Power Station," Revision 35

  This evaluation does not constitute NRC approval. 71114.05 - Maintenance of Emergency Preparedness Inspection Review (IP Section 02.01 - 02.11) (1 Sample)

(1) The inspectors evaluated the maintenance of the Emergency Preparedness Program on February 8–11, 2021, for the period of March 2019 through January 2021

OTHER ACTIVITIES – BASELINE

71151 - Performance Indicator Verification
The inspectors verified licensee performance indicators submittals listed below: EP01: Drill/Exercise Performance (IP Section 03.12) (1 Sample)

(1) Unit common for the period October 1, 2020 - December 31, 2020
IE01: Unplanned Scrams per 7000 Critical Hours Sample (IP Section 03.01) (2 Samples)

(1) Unit 2 for the period January 1, 2020 to December 31, 2020 7

(2) Unit 3 for the period January 1, 2020 to December 31, 2020 EP02: ERO Drill Participation (IP Section 03.13) (1 Sample)

(1) Unit common for the period October 1, 2020 - December 31, 2020

IE03: Unplanned Power Changes per 7000 Critical Hours Sample (IP Section 03.02) (2 Samples)

1. (1) Unit 2 for the period January 1, 2020 to December 31, 2020
2. (2) Unit 3 for the period January 1, 2020 to December 31, 2020

EP03: Alert & Notification System Reliability (IP Section 03.14) (1 Sample) (1) Unit common for the period October 1, 2020 - December 31, 2020

IE04: Unplanned Scrams with Complications (USwC) Sample (IP Section 03.03) (2 Samples)

1. (1) Unit 2 for the period January 1, 2020 to December 31, 2020
2. (2) Unit 3 for the period January 1, 2020 to December 31, 2020

71152 - Problem Identification and Resolution
Annual Follow-up of Selected Issues (IP Section 02.03) (1 Sample)

The inspectors reviewed the licensee's implementation of its corrective action program related to the following issues:

(1) 10 CFR Part 21, Battery Charger Oscillator Card Short to Ground 71153 - Followup of Events and Notices of Enforcement Discretion
Event Report (IP Section 03.02) (1 Sample)

The inspectors evaluated the following licensee event report (LER):

(1) LER 05000277/2020-002-00, Degraded Condition Due to RPV Instrument Nozzle Leak (ADAMS Accession No. ML20357B113)

The inspection conclusions associated with this LER are documented in this report under Inspection Results Section.

INSPECTION RESULTS

High-Pressure Service Water Valve Failed to Open			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000277/2021001-01 Open/Closed	[H.12] - Avoid Complacency	71111.15
A self-revealing Green non-cited violation (NCV) of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," because Exelon did not accomplish work in accordance with instructions during maintenance and staging activities. Specifically, during planned work staging, Exelon personnel did not effectively prevent contact with plant equipment which resulted in unplanned unavailability of high-pressure service water (HPSW) to the 'D' residual heat removal (RHR) heat exchanger.			

Description: The safety objective of the Unit 2 HPSW system is to provide a reliable supply of cooling water for the Unit 2 RHR system under post-accident conditions. Unit 2 HPSW consists of four 4500 gpm pumps installed in parallel in the pump structure with its normal water supply to the suction of the pumps from the Conowingo pond. The 'D' loop of HPSW provides cooling water to the 'D' RHR heat exchanger. On February 6, 2021, during surveillance testing, the outlet valve did not open when the control switch was taken to the open position. Operators attempted to open the outlet valve a second time, and the valve did not stroke and no change in position indication was observed at the power supply breaker.

Maintenance personnel then performed troubleshooting of the valve. Technicians determined the amphenol connector back-shell was cross-threaded and associated control circuitry was unexpectedly open. The technicians disconnected and re-installed the amphenol connector which restored the control circuit, and the valve stroked satisfactorily. Subsequent investigation determined that the cable conduit attached to the amphenol connector was previously disturbed in a manner that exerted force on the amphenol connection.

Exelon determined that the connection was most likely disturbed during a work staging activity on February 5 or 6, 2021. In particular, valve rigging was installed above and in the immediate proximity of the valve for a planned valve replacement the following week. In addition, the conduit appeared to have been disturbed because the protective jacket was pulled out of the junction box, such as from being stepped upon or pushed against. Therefore, Exelon concluded that the most likely apparent cause was a human performance error event during the mobilization and preparation field activities.

The inspectors reviewed the maintenance and performance history of the outlet valve and considered the operating environment, proximity and spacing of equipment, and the orientation of the control wiring conduit and amphenol connector. The inspectors concluded that Exelon's determination that the most likely cause was an event on February 5 that disturbed the connection was appropriate. However, the inspectors determined that the condition of the amphenol connector back-shell being not installed completely, and apparently cross-threaded, was most likely due to a pre-existing contributing cause to the event.

In particular, the inspectors noted that no damage was identified with the amphenol connector or its threads, including the back-shell, and the lock-wire was found intact and undisturbed. The inspectors further questioned the as-found conditions which revealed that the back-shell was threaded down further upon re-installation relative to the as-found condition. Therefore, during the last valve maintenance in 2014, the back-shell was most likely not installed correctly and not fully threaded in place, potentially cross-threaded, which was a failure to implement the work instructions correctly. Then, on February 5, the conduit was likely disturbed which pulled the inserted plug up to the back-shell, stressed the back-shell and threads, and disengaged the amphenol pin connection.

Although this non-qualified condition likely existed since 2014, the inspectors determined the most probable safety impact was limited to February 5 and the ability of the valve to function was not compromised until then. This determination was made, in part, on there being no malfunction for a seven year period, the configuration of the conduit to amphenol connection likely applying a holding force to the connection, the horizontal orientation of the amphenol connector, and the mild operating environment, including vibration, being judged unlikely to introduce a pulling force on the plug under design accident conditions, and unlikely to challenge the pin connection. Therefore, the most probable cause of the valve failing to open was a human performance error event in which the work instructions for mobilizing and staging equipment in the field, including the rigging for the outlet valve replacement, were not accomplished as intended by disturbing the cable conduit and amphenol connector. Finally, the human performance error prevention tools contained in Exelon's procedure HU-AA-101, "Human Performance Tools and Verification Practices," if performed correctly, would have prevented contact with the cable conduit and prevented the failure.

Corrective Actions: Exelon disassembled and reinstalled the amphenol connector which restored the circuit continuity and restored the valve's functionality on February 7, 2021, which was approximately two days from February 5, out of a seven-day allowed outage time. Then, in the following week, the valve operator and wiring were replaced during planned maintenance.

Corrective Action References: IR 4400698 and IR 4400876

Performance Assessment:

Performance Deficiency: The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawing," because Exelon did not accomplish work in accordance with instructions during maintenance and staging activities. Specifically, during a planned work staging activity, Exelon personnel did not effectively prevent contact with plant equipment which resulted in unplanned unavailability of HPSW to the 'D' RHR heat exchanger.

Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the Configuration Control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the impact on the cable conduit to the amphenol connector resulted in the unplanned unavailability of the HPSW system to the 'D' RHR heat exchanger.

Significance: The inspectors assessed the significance of the finding using Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." Using IMC 0609, Appendix A, Exhibit 2, the inspectors determined that this finding was of very low safety

significance (Green). Specifically, the finding did not represent a loss of probabilistic risk assessment (PRA) system or function and did not represent the loss of the PRA function of one train of a multi-train system for greater than its TS allowed outage time. Therefore, the inspectors determined the finding to be of very low safety significance (Green). Cross-Cutting Aspect: H.12 - Avoid Complacency: Individuals recognize and plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful outcomes. Individuals implement appropriate error reduction tools. On February 5, 2021, Exelon personnel did not properly implement human error reduction tools such that adverse contact impacted the control wiring connecting to the operator of the HPSW outlet valve to the 'D' RHR heat exchanger.
Enforcement: Violation: 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that work which can affect safety-related equipment be accomplished in accordance with instructions appropriate to the circumstances. Contrary to the above, on February 5 or 6, 2021, work was not accomplished in accordance with instructions appropriate to the circumstances. Specifically, during preparation and field mobilization work activities for the valve, the valve was intended to be maintained operable according to the work instructions, but this was not accomplished in accordance with the instructions due to a human error event which disturbed the equipment. Enforcement Action: This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy.

Observation: Battery Charger Oscillator Card Short to Ground	71152
The inspectors reviewed Condition Reports 04338712 and 04364596 which document Exelon's evaluation, extent of condition reviews, and corrective actions associated with a Part 21 notification issued by United Controls International (UCI) on June 25, 2020. This Part 21 is specific to Peach Bottom Atomic Power Station. The inspectors focused on Exelon's planned and/or implemented corrective actions to ensure they were commensurate with the safety significance of the problem. The notification documents a manufacturing and modification error involving ABB/Thomas & Betts/Cyberex P/N: 93-41-119385 Time Delay Oscillator Printed Circuit Boards supplied to Exelon Nuclear Peach Bottom Station by UCI on May 8, 2009. Specifically, the error was determined to involve insufficient electrical clearances between the board's mounting holes and the mounting washers to the energized traces such that when the subject boards are installed, the mounting flat washers may come in contact with the conductive traces resulting in short circuit to the mounting chassis. Exelon engineers identified this degraded condition on April 21, 2020, during installation of one of the available battery charger oscillator circuit boards onto a safety-related station battery charger. The original board was reinstalled, and the battery ground fault condition cleared. Exelon's evaluation determined the cause of the failure was due to manufacturing and modification errors that occurred in 2008 when the boards were returned to UCI for inspection and repairs to address a short to ground condition identified in 2008 during installation of a similar board in a non-safety-related battery charger (IR 0767543). In 2008, staff at the Peach Bottom plant identified the ground was caused by metal to metal contact of the mounting washer to the trace metal on the back of the board. Their investigation also identified that some similar boards had mounting holes that were too small and needed to be	

enlarged and that the trace metal on the back of the boards needed to be cut to prevent the hard ground/short circuit. Peach Bottom staff worked with the supplier to implement these two actions and the refurbished circuit boards were supplied back to Peach Bottom on May 8, 2009. The problem reoccurred in April 21, 2020, resulting in a Part 21 notification.

The inspectors reviewed the Part 21 notification and associated documentation, interviewed station personnel, reviewed implemented corrective actions, and independently inspected several of the boards currently in stock to verify adequate electrical clearances exist between the mounting holes and the traces that would be energized. The inspectors concluded the issue that resulted in the Part 21 notification was evaluated sufficiently to identify the causes and develop effective corrective actions, and that the extent of condition review was adequate to identify affected boards. Corrective actions included removal of the affected boards, visual inspection and testing of similar circuit boards in stock, removal of all affected boards from stock and returning them to the manufacturer for detailed inspections of electrical clearances between mounting holes and energized traces, and testing of the boards at Peach Bottom with ground detection. Additionally, the inspectors determined there were no common mode failure concerns associated with this issue, because only one of the susceptible boards supplied in 2008 is currently installed in a non-safety-related battery charger and there is sufficient redundancy and design separation between the safety-related battery chargers such that a ground in one of the two chargers would not affect the second charger. Additionally, this board has operated properly since installation. Regarding actions taken in 2008 for this problem, the inspectors concluded that Peach Bottom staff implementing their corrective action program missed opportunities to work out with their vendor appropriate testing acceptance protocols for these circuit board cards to correct this problem, which was appropriately addressed in their corrective action program in 2020.

<table>
<tr><td colspan="4">Degraded Condition Due to Reactor Pressure Vessel Instrument Nozzle Leakage</td></tr>
<tr><td>Cornerstone</td><td>Severity</td><td>Cross-Cutting Aspect</td><td>Report Section</td></tr>
<tr><td>Not Applicable</td><td>Severity Level IV
NCV 05000277/2021001-02
Open/Closed</td><td>Not Applicable</td><td>71153</td></tr>
<tr><td colspan="4">A Severity Level IV NCV of Peach Bottom Atomic Power Station, Unit 2, Technical Specification (TS) 3.4.4, "Reactor Coolant System (RCS) Operational Leakage," was self-revealed when leakage was identified from the 'N16A' 2 inch reactor pressure vessel (RPV) instrument nozzle during a RPV pressure test on October 29, 2020. Specifically, RCS pressure boundary leakage reasonably began on an unknown date that was more than 36 hours before the shutdown for the refueling outage on October 21, 2020.</td></tr>
</table>

Description: On October 29, 2020, while Peach Bottom Atomic Power Station, Unit 2, was in cold shutdown for a refueling outage during a RPV pressure test, a through-wall leak was identified from the 'N16A' 2 inch RPV instrument nozzle. A visual examination detected active leakage in the form of slight weeping at the nozzle's interface with the RPV. The condition was reported in event notification 54971, as required by 10 CFR 50.72(b)(3)(ii)(A), because it represented a degradation of a principal safety barrier.

Exelon evaluated the flaw and determined the RCS pressure boundary leakage was most likely caused by intergranular stress corrosion cracking (IGSCC) in which a single radial-axial oriented IGSCC flaw initiated in the J-groove weld and then propagated through the J-groove weld until it reached a depth where a leak path in the annulus between the nozzle and reactor vessel penetration existed. Exelon's corrective actions included a half nozzle repair, a post-

leakage test, and an extent of condition review. The inspectors reviewed the LER, Exelon's root cause evaluation of the event, and performed visual inspection of the leak conditions and determined that RCS pressure boundary leakage reasonably began on an unknown date that was more than 36 hours before the shutdown for the refueling outage on October 21, 2020. Corrective Actions: A nozzle repair was completed on November 11, 2020. The repair method involved installing a weld pad and then installing a half nozzle to the weld pad, making the nozzle resistant to IGSCC. The original partial penetration attachment weld and a remnant of the original nozzle remained in place. A failure assessment and flaw evaluation were completed prior to startup to demonstrate the acceptability of leaving the original partial penetration attachment weld, with a maximum postulated flaw, in place for one operating cycle. Exelon submitted Relief Request 15R-14 for this, which was granted. Exelon plans to submit a separate Relief Request to allow continued use of the nozzle repair for the life of the plant. Corrective Action References: IR 4380514
Performance Assessment: The NRC determined this violation was not reasonably foreseeable and preventable by the licensee and therefore is not a performance deficiency.
Enforcement: Severity: The inspectors informed the significance of the pressure boundary leakage, and determined that had the condition represented a performance deficiency, it would be very low safety significance (Green) because it would not result in exceeding the RCS leak rate for a small loss of coolant accident, and would not have likely affected other systems used to mitigate a loss of coolant accident. Therefore, this violation is being characterized as Severity Level IV. Violation: Peach Bottom Atomic Power Station, Unit 2, TS 3.4.4, "RCS Operational Leakage," requires, in part, that RCS operational leakage be limited to no pressure boundary leakage, and if pressure boundary leakage exists that the unit be in Mode 3 within 12 hours and Mode 4 in 36 hours. Contrary to the above, on an unknown date more than 36 hours prior to the unit being in Mode 3 and Mode 4 on October 21, 2020, RCS pressure boundary leakage existed. The disposition of this violation closes LER 05000277-2020-002-00. Enforcement Action: This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy.

EXIT MEETINGS AND DEBRIEFS

The inspectors verified no proprietary information was retained or documented in this report.

- On April 16, 2021, the inspectors presented the integrated inspection results to Mr. Matthew Herr, Site Vice President, and other members of the licensee staff.
- On February 11, 2021, the inspectors presented the Emergency Preparedness Program inspection results to Mr. Dave Henry, Plant Manager and other members of the licensee staff.

- On March 10, 2021, the inspectors presented the Problem Identification and Resolution Sample, Part 21, Battery Charger Oscillator Card Short to Ground inspection results to Mr. Ryan Stiltner, Site Engineering Director and other members of the licensee staff.

DOCUMENTS REVIEWED

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
71111.01	Corrective Action Documents	04388255 04394312		
	Procedures	RT-O-040-630-2	Winterizing Procedure	Revision 16
71111.04	Corrective Action Documents Resulting from Inspection	IR 4411488		
71111.05	Procedures	CC-AA-211	Fire Protection Program	Revision 9
		OP-AA-201-007	Fire Protection System Impairment Control	Revision 0
		OP-AA-201-009	Control of Transient Combustible Material	Revision 25
		OP-AA-201-012-1001	Operations On-line Fire Risk Management	Revision 4
		PF-132	Diesel Generator Building, General Area, Elevation 127'-0"	Revision 9
71111.12	Corrective Action Documents	Issue Reports (IRs) 4374949 4380174 4380183 4389490 4397504 4397775 4398057 4397997 4398001 4399074 4399081 4399270 4399351 4399950 4402021		
71111.15	Corrective Action Documents	IR 4393071		
		IR 4400698		

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		IRs 4395836 4396360		
		IRs 4399671 4286215		
		IRs 4406032 4406048		
	Procedures	ST-O-010-307-2	'B' RHR Loop, Pump, Value, Flow, and Unit Cooler Functional and Inservice Comprehensive Test	Revision 17
		ST-O-032-301-2	HPSW Pump, Valve, and Flow Functional and Inservice Test	Revision 38
	Work Orders	WO 05082136 WR 01485131		
		WO 4882738		
71111.19	Procedures	ST-I-37G-392-2	E-2 Diesel Generator Cardox System Simulated Actuation and Air Flow Test	Revision 0
		ST-I-37G-394-2	E-4 Diesel Generator Cardox System Simulated Actuation and Air Flow Test	Revision 10
	Work Orders	WO 04303155		
		WO 04648426		
		WO 05079121		
71111.22	Corrective Action Documents	IR 4396187		
	Procedures	RT-I-033-631-3	RHR Cooler ESW Heat Transfer Test	Revision 14
		S13K-10-TDR-A1C2	Calibration Functional Check of Low-Pressure Coolant Injection Pump Start Time Delay Relays	Revision 8
		ST-O-010-306-3	'B' RHR Loop Pump, Valve, Flow, and Unit Cooler Functional and Inservice Test	Revision 48
		ST-O-052-152-3	'E-2' DG Simulated Unit 3 ECCS Signal Auto Start with Off-Site Power Available and Full Load Test	Revision 14
	Work Orders	WO 04944974		
		WO 4866766		
		WO 5089892		
71114.02	Miscellaneous	Design Report	Peach Bottom Atomic Power Station Public Alert and Notification System Design Report	Revision 1
71114.03	Miscellaneous	EP-AA-1000	Exelon Nuclear Standardized Radiological Emergency Plan	Revision 33

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Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		EP-AA-1007	Exelon Nuclear Radiological Emergency Plan Annex for Peach Bottom Atomic Power Station	Revision 35
		EP-AA-1007 Addendum 1	Peach Bottom Atomic Power Station On-Shift Staffing Technical Basis	Revision 2
71114.04	Corrective Action Documents Resulting from Inspection	04401709		
	Procedures	EP-AA-120-1001	10 CFR 50.54(q) Change Evaluation	Revision 11
71114.05	Corrective Action Documents Resulting from Inspection	04401349		
		04401467		
		04401565		
	Procedures	EP-AA-121-F-07	Peach Bottom Equipment Matrix	Revision 9
71152	Corrective Action Documents	01465239		
		04338712		
		04364596		
	Miscellaneous	10 CFR Part 21	Notification for ABB/Thomas & Betts / Cyberex P/N: 93-41-119385 Time Delay Oscillator Printer Circuit Boards	June 25, 2020

May 14, 2021 - Email from Jennifer Tobin to David Helker (Exelon Nuclear) cc to Richard Grupp, Jr. (Exelon Nuclear) with subject of Acceptance of Requested Licensing Action: Peach Bottom Ventilation Filter Testing LAR (EPID No. L-2021-LLA- 0078)

By letter dated April 29, 2021 (ADAMS Accession No. ML21119A141), Exelon (the licensee) submitted a license amendment request for Peach Bottom Units 2, and 3. The licensee proposed to revise TS 5.5.7, "Ventilation Filter Testing Program (VFTP)," to change the frequency for performing certain testing requirements from 12 months as currently specified to 24 months. The VFTP establishes the required testing and testing frequency of Engineered Safety Feature (ESF) filter ventilation systems.. The purpose of this e-mail is to provide the results of the U.S. Nuclear Regulatory Commission (NRC) staff's acceptance review of this exemption request. The acceptance review was performed to determine if there is sufficient technical information in scope and depth to allow the NRC staff to complete its detailed technical review. The acceptance review is also intended to identify whether the application has any readily apparent information insufficiencies in its characterization of the regulatory requirements or the licensing basis of the plant.

The NRC staff has reviewed your application and concluded that it does provide technical information in sufficient detail to enable the NRC staff to complete its detailed technical review and make an independent assessment regarding the acceptability of the proposed exemption in terms of regulatory requirements and the protection of public health and safety and the environment. Given the lesser scope and depth of the acceptance review as compared to the detailed technical review, there may be instances in which issues that impact the NRC staff's ability to complete the detailed technical review are identified despite completion of an adequate acceptance review. If additional information is needed, you will be advised by separate correspondence.

Based on the information provided in your submittal, the NRC staff has estimated that this licensing request will take approximately **250 hours** to complete. The NRC staff expects to complete this review no later than **May 31, 2022**. If there are emergent complexities or challenges in our review that would cause changes to the initial forecasted completion date or significant changes in the forecasted hours, the reasons for the changes, along with the new estimates, will be communicated during the routine interactions with the assigned project manager.

These estimates are based on the NRC staff's initial review of the application and they could change, due to several factors including requests for additional information, and unanticipated addition of scope to the review. Additional delay may occur if the submittal is provided to the NRC in advance or in parallel with industry program initiatives or pilot applications.

Please contact me if you have any questions. A copy of this email will be made publicly available in ADAMS.

<u>May 14, 2021</u> – Email from Jennifer Tobin to David P. Helker (Exelon Nuclear) cc to Richard W. Gropp, Jr. and Jason Drake with subject of Acceptance of Requested Licensing Action: Peach Bottom Ventilation Filter Testing LAR (EPID No. L-2021-LLA-0078)

By letter dated April 29, 2021 (ADAMS Accession No. ML21119A141), Exelon (the licensee) submitted a license amendment request for Peach Bottom Units 2, and 3. The licensee proposed to revise TS 5.5.7, "Ventilation Filter Testing Program (VFTP)," to change the frequency for performing certain testing requirements from 12 months as currently specified to 24 months. The VFTP establishes the required testing and testing frequency of Engineered Safety Feature (ESF) filter ventilation systems.. The purpose of this e-mail is to provide the results of the U.S. Nuclear Regulatory Commission (NRC) staff's acceptance review of this exemption request. The acceptance review was performed to determine if there is sufficient technical information in scope and depth to allow the NRC staff to complete its detailed technical review. The acceptance review is also intended to identify whether the application has any readily apparent information insufficiencies in its characterization of the regulatory requirements or the licensing basis of the plant.

The NRC staff has reviewed your application and concluded that it does provide technical information in sufficient detail to enable the NRC staff to complete its detailed technical review and make an independent assessment regarding the acceptability of the proposed exemption in terms of regulatory requirements and the protection of public health and safety and the environment. Given the lesser scope and depth of the acceptance review as compared to the detailed technical review, there may be instances in which issues that impact the NRC staff's ability to complete the detailed technical review are identified despite completion of an adequate acceptance review. If additional information is needed, you will be advised by separate correspondence.

Based on the information provided in your submittal, the NRC staff has estimated that this licensing request will take approximately **250 hours** to complete. The NRC staff expects to complete this review no later than **May 31, 2022**. If there are emergent complexities or challenges in our review that would cause changes to the initial forecasted completion date or significant changes in the forecasted hours, the reasons for the changes, along with the new estimates, will be communicated during the routine interactions with the assigned project manager.

These estimates are based on the NRC staff's initial review of the application and they could change, due to several factors including requests for additional information, and unanticipated addition of scope to the review. Additional delay may occur if the submittal is provided to the NRC in advance or in parallel with industry program initiatives or pilot applications.

Please contact me if you have any questions. A copy of this email will be made publicly available in ADAMS.

May 18, 2021

This placed into ADAMS today of May 18, 2921 Event.
N2
MJK
ML21196A485
https://adamswebsearch2.nrc.gov/webSearch2/main.jsp?AccessionNumber=ML21196A485

Document Title:	LER 2-2021-002-00 for Peach Bottom Atomic Power Station, Unit 2, Safety Relief Valve Inoperability Due to Nitrogen Leakage from Braided Hose Wear
Document Type:	Letter Licensee Event Report (LER)
Document Date:	07/16/2021

August 4, 2021 – Letter from Jonathan E. Greives, Chief Projects Branch 4 Division of Operating Reactor Safety to David P. Rhoades Senior Vice President Exelon Generation Company, LLC President and Chief Nuclear Officer Exelon Nuclear with subject of PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 – INTEGRATED INSPECTION REPORT 05000277/2021002 AND 05000278/2021002

On June 30, 2021, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Peach Bottom Atomic Power Station, Units 2 and 3. On July 16, 2021, the NRC inspectors discussed the results of this inspection with Mr. Matthew Herr, Site Vice President, and other members of your staff. The results of this inspection are documented in the enclosed report.

No findings or violations of more than minor significance were identified during this inspection.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with Title 10 of the *Code of Federal Regulations* 2.390, "Public Inspections, Exemptions, Requests for Withholding."

U.S. NUCLEAR REGULATORY COMMISSION Inspection Report

Docket numbers: 05000277 and 05000278
License numbers: DPR-44 and DPR-56
Report numbers: 05000277/2021002 and 05000278/2021002

Enterprise identifier: I-2021-002-0024
Licensee: Exelon Generation Company, LLC
Facility: Peach Bottom Atomic Power Station, Units 2 and 3

Location: Delta, PA 17314
Inspection dates: April 1, 2021 to June 30, 2021

Inspectors: S. Rutenkroger, Senior Resident Inspector

P. Boguszewski, Senior Resident Inspector

J. Brand, Reactor Inspector
T. Corcoran, Project Engineer

T. Fish, Senior Operations Engineer

T. Hedigan, Operations Engineer

Approved by: Jonathan E. Greives, Chief Projects Branch 4
Division of Operating Reactor Safety

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting an integrated inspection at Peach Bottom Atomic Power Station, Units 2 and 3, in accordance with the Reactor Oversight Process. The Reactor

Oversight Process is the NRC's program for overseeing the safe operation of commercial nuclear power reactors. Refer to https://www.nrc.gov/reactors/operating/oversight.html for more information.

List of Findings and Violations

No findings or violations of more than minor significance were identified.

Additional Tracking Items

None.

PLANT STATUS

Unit 2 began the inspection period at rated thermal power (RTP). On May 17, 2021, the unit was down powered to 9 percent for work inside the drywell that identified and repaired two conditions: nitrogen leakage from the instrument gas system and an increasing unidentified leak rate. The unit was returned to RTP on May 19, 2021. The unit was down powered to 70 percent for a follow-up control rod pattern adjustment on May 20, 2021, and returned to 100 percent RTP the following day. The unit remained at or near RTP for the remainder of the inspection period.

Unit 3 began the inspection period at RTP. On April 16, 2021, the unit was down powered to 53 percent for a control rod pattern adjustment, main turbine valve testing, and waterbox cleaning. The unit was returned to RTP the following day. On May 15, 2021, the unit was down powered to 68 percent for a control rod pattern adjustment and main turbine valve testing. The unit was returned to RTP the following day. On June 12, 2021, the unit was down powered to 68 percent for a control rod pattern adjustment, main turbine bypass valve exercising, and restoration of two hydraulic control units following maintenance. The unit was restored to RTP the following day. The unit remained at or near RTP for the remainder of the inspection period.

INSPECTION SCOPES

Inspections were conducted using the appropriate portions of the inspection procedures (IPs) in effect at the beginning of the inspection unless otherwise noted. Currently approved IPs with their attached revision histories are located on the public website at http://www.nrc.gov/reading- rm/doc-collections/insp-manual/inspection-procedure/index.html. Samples were declared complete when the IP requirements most appropriate to the inspection activity were met consistent with Inspection Manual Chapter (IMC) 2515, "Light-Water Reactor Inspection Program - Operations Phase." The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel to assess licensee performance and compliance with Commission rules and regulations, license conditions, site procedures, and standards. Starting on March 20, 2020, in response to the National Emergency declared by the President of the United States on the public health risks of the coronavirus (COVID-19), resident and regional inspectors were directed to begin telework and to remotely access licensee information using available technology. During this time, the resident inspectors performed periodic site visits each week, increasing the amount of time on site as local COVID-19 conditions permitted. As part of their onsite activities, resident inspectors conducted plant status activities as described in IMC 2515, Appendix D; observed risk significant activities; and completed on site portions of IPs. In addition, resident and regional baseline inspections were evaluated to determine if all or a portion of the objectives and requirements stated in the IP could be performed remotely. If the inspections could be performed remotely, they were conducted per the applicable IP. In some cases, portions of an IP were completed remotely and on site. The inspections documented below met the objectives and requirements for completion of the IP.

REACTOR SAFETY

71111.01 - Adverse Weather Protection
Seasonal Extreme Weather Sample (IP Section 03.01) (1 Sample)

(1) The inspectors evaluated readiness for seasonal extreme weather conditions prior to the onset of seasonal hot temperatures for the following systems: emergency diesel generators (EDGs), station emergency batteries, 4kV switchgear rooms, control room, and drywell chillers as of June 10, 2021

71111.04 - Equipment Alignment
Partial Walkdown Sample (IP Section 03.01) (3 Samples)

The inspectors evaluated system configurations during partial walkdowns of the following systems/trains:

1. (1) Unit 3 'A' residual heat removal (RHR) during Unit 3 'B' RHR testing and maintenance on April 26, 2021
2. (2) Unit 2 high-pressure coolant injection (HPCI) during Unit 2 reactor core isolation coolant (RCIC) maintenance on June 1, 2021
3. (3) Unit 3 'B' RHR following swing bus testing and prior to RCIC planned maintenance on June 11, 2021

Complete Walkdown Sample (IP Section 03.02) (1 Sample)

(1) The inspectors evaluated system configurations during a complete walkdown of the Unit 3 RCIC during the weeks of May 10 and May 31, 2021

71111.05 - Fire Protection
Fire Area Walkdown and Inspection Sample (IP Section 03.01) (5 Samples)

The inspectors evaluated the implementation of the fire protection program by conducting a walkdown and performing a review to verify program compliance, equipment functionality, material condition, and operational readiness of the following fire areas:

1. (1) PF-132, Unit common diesel generator building, general area on April 1, 2021
2. (2) PF-57, Unit 2 refuel floor on April 15, 2021
3. (3) PF-78H, Unit common cable spreading and computer rooms on May 12, 2021

4. (4) PF-127, Unit 2 turbine building, emergency battery and switchgear rooms, elevation

   135' on June 23, 2021

5. (5) PF-132, Unit common diesel generator building, general area on June 23, 2021

71111.06
Inspection Activities - Internal Flooding (IP Section 03.01) (1 Sample)

The inspectors evaluated internal flooding mitigation protections in the: (1) Unit 2 RCIC room on June 25, 2021

- Flood Protection Measures

71111.11A - Licensed Operator Requalification Program and Licensed Operator Performance Requalification Examination Results (IP Section 03.03) (1 Sample)

(1) The inspectors reviewed and evaluated the licensed operator examination failure rates for the requalification annual operating exam administered on
April 6, 2021

71111.11B - Licensed Operator Requalification Program and Licensed Operator Performance Licensed Operator Requalification Program (IP Section 03.04) (1 Sample)

(1) Biennial Requalification Written Examinations

The inspectors evaluated the quality of the licensed operator biennial requalification written examination administered in March 2020

Annual Requalification Operating Tests

The inspectors evaluated the adequacy of the facility licensee's annual requalification operating test administered in March 2021

Administration of an Annual Requalification Operating Test

The inspectors evaluated the effectiveness of the facility licensee in administering requalification operating tests required by 10 CFR 55.59(a)(2) and that the facility licensee is effectively evaluating their licensed operators for mastery of training objectives

Requalification Examination Security

The inspectors evaluated the ability of the facility licensee to safeguard examination material, such that the examination is not compromised

Remedial Training and Re-examinations

The inspectors evaluated the effectiveness of remedial training conducted by the licensee, and reviewed the adequacy of re-examinations for licensed operators who did not pass a required requalification examination

Operator License Conditions

The inspectors evaluated the licensee's program for ensuring that licensed operators meet the conditions of their licenses

Control Room Simulator

The inspectors evaluated the adequacy of the facility licensee's control room simulator in modeling the actual plant, and for meeting the requirements contained in 10 CFR 55.46

Problem Identification and Resolution

The inspectors evaluated the licensee's ability to identify and resolve problems associated with licensed operator performance

71111.11Q - Licensed Operator Requalification Program and Licensed Operator Performance

Licensed Operator Performance in the Actual Plant/Main Control Room (IP Section 03.01) (1 Sample)

(1) The inspectors observed and evaluated licensed operator performance in the main control room during preparations for a Unit 2 drywell entry, including a power reduction to 10 percent, on May 17, 2020, and again during EDG testing on
June 29, 2021

Licensed Operator Requalification Training/Examinations (IP Section 03.02) (1 Sample)

(1) The inspectors observed and evaluated licensed operator requalification training in the simulator on April 19, 2021

71111.12 - Maintenance Effectiveness
Maintenance Effectiveness (IP Section 03.01) (2 Samples)

The inspectors evaluated the effectiveness of maintenance to ensure the following structures, systems, and components remain capable of performing their intended function:

1. (1) Unit common, 'E-4' EDG as of April 26, 2021
2. (2) Unit common, power supply inverters as of June 10, 2021

71111.13 - Maintenance Risk Assessments and Emergent Work Control Risk Assessment and Management Sample (IP Section 03.01) (6 Samples)

The inspectors evaluated the accuracy and completeness of risk assessments for the following planned and emergent work activities to ensure configuration changes and appropriate work controls were addressed:

1. (1) Unit 2 RCIC planned maintenance on April 6, 2021
2. (2) Unit 3 RCIC planned maintenance on April 7, 2021
3. (3) Unit 2 HPCI planned maintenance on April 12, 2021
4. (4) Unit 3 'A' RHR heat exchanger planned cleaning and maintenance on April 19, 2021
5. (5) Unit 2 emergent HPCI flow controller inverter failure on April 30, 2021
6. (6) Unit 2 emergent 'K' safety-relief valve maintenance on May 18, 2021

71111.15
Operability Determination or Functionality Assessment (IP Section 03.01) (4 Samples)

- Operability Determinations and Functionality Assessments

The inspectors evaluated the licensee's justifications and actions associated with the following operability determinations and functionality assessments:

(1) Unit 2 high-pressure service water (HPSW) RHR heat exchanger outlet valve junction box supported only by attached conduit on March 30, 2021

(2) Unit common, reactor building water curtain fire systems' automatic actuation circuits were not being properly tested during surveillance testing identified on April 14, 2021

(3) Unit common, HPSW hangers with gaps identified between the hanger shims and the piping on May 10 and 11, 2021

(4) Unit 3 RCIC with degraded support spring can on May 11, 2021 71111.18 - Plant Modifications

Temporary Modifications and/or Permanent Modifications (IP Section 03.01 and/or 03.02) (2 Samples)

The inspectors evaluated the following temporary or permanent modifications:

(1) Unit common, 'E-4' EDG's emergency service water pipe flexible expansion joints engineering change evaluation of dimensions as of April 14, 2021

(2) Modification that installed steel reinforcing angles for seismic restraint with respect to Unit 2 masonry wall 'BW-76.8' as of May 5, 2021

71111.19 - Post-Maintenance Testing
Post-Maintenance Test Sample (IP Section 03.01) (9 Samples)

The inspectors evaluated the following post-maintenance test activities to verify system operability and functionality:

1. (1) Unit 3 RCIC remote shutdown panel after inverter failure and replacement on April 7, 2021
2. (2) Unit 3 'A' RHR heat exchanger testing after planned maintenance and cleaning on April 19 through April 22, 2021
3. (3) Unit common, 'E-323' breaker testing after replacement on April 28, 2021
4. (4) Unit 2 HPCI flow controller inverter testing after replacement on April 30, 2021
5. (5) Unit common, high-pressure lube water pump pressure switch replacement and

   testing on May 5, 2021

6. (6) Unit 2 RCIC system testing following planned system maintenance activities,

   including inverter replacement and value work on June 3, 2021

7. (7) Unit 2 'B' RHR and HPSW testing after valve maintenance on June 7, 2021
8. (8) Unit common, 'E-3' EDG testing following scheduled maintenance overhaul on

   June 27, 2021

9. (9) Unit 3 RCIC support spring can after repair on June 28, 2021

71111.22
The inspectors evaluated the following surveillance tests: Surveillance Tests (other) (IP Section 03.01) (3 Samples)

1. (1) Unit common, 'E-2' EDG slow start full load surveillance test on April 28, 2021
2. (2) Unit 2 RHR alternate shutdown panel testing on April 29, 2021

- Surveillance Testing

(3) Unit 3 RCIC pump, valve, and flow on June 15, 2021 71114.06 - Drill Evaluation Drill/Training Evolution Observation (IP Section 03.02) (1 Sample)

(1) The inspectors evaluated a limited scope emergency preparedness drill conducted on June 25, 2021

OTHER ACTIVITIES – BASELINE

71151 - Performance Indicator Verification
The inspectors verified licensee performance indicators submittals listed below:
BI01: Reactor Coolant System (RCS) Specific Activity Sample (IP Section 02.10) (2 Samples)

1. (1) Unit 2 April 1, 2020 to March 31, 2021

2. (2) Unit 3 April 1, 2020 to March 31, 2021

BI02: RCS Leak Rate Sample (IP Section 02.11) (2 Samples)

1. (1) Unit 2 April 1, 2020 to March 31, 2021
2. (2) Unit 3 April 1, 2020 to March 31, 2021

71152 - Problem Identification and Resolution Semi-annual Trend Review (IP Section 02.02) (1 Sample)

(1) The inspectors conducted a semi-annual trend review by evaluating sample issues that occurred in the first and second quarters of 2021. During the evaluation, the inspectors verified the issues identified were addressed within the scope of the corrective action program (CAP)

Annual Follow-up of Selected Issues (IP Section 02.03) (1 Sample)

The inspectors reviewed the licensee's implementation of its CAP related to the following issues:

(1) Fire penetration seals issues
71153 - Follow Up of Events and Notices of Enforcement Discretion Personnel Performance (IP Section 03.03) (1 Sample)

(1) The inspectors evaluated a declared unusual event due to the existence of smoke in the Unit 2 drywell and personnel unable to verify that no fire existed within 30 minutes and the licensee's subsequent performance on May 17, 2021

INSPECTION RESULTS

Observation: Fire Penetration Seals Issues	71152
The inspectors reviewed Condition Reports 04312522, 04195447, and 04310888 which document Exelon's evaluation, extent of condition reviews, and corrective actions associated with fire penetration seals inspection deficiencies identified by their nuclear oversight (NOS) personnel during routine internal reviews of fire seals inspections per station procedures ST-M-037-311-2 and 3, "Detailed Visual Inspection of Fire Penetration Seals." The inspectors focused on Exelon's planned and/or implemented corrective actions to determine whether they were commensurate with the safety significance of the problems. The NOS findings which were initially identified on November 15, 2018 (AR 04195447), involved a review of the Peach Bottom Unit 2 fire penetration seals surveillance inspections completed in August 2018. Specifically, NOS identified documentation and tracking errors of numerous fire penetration seals that were inaccessible and could not be inspected, as well as other inspection documentation errors and the failure to resolve missing and incorrect information in the penetration Component Record List. Additionally, in January 2020, NOS' review of the Peach Bottom Unit 3 draft surveillance procedure resulted in identification of errors in the surveillance tracking tool used as a basis for the inspection procedure. As a result, 21 penetration seals were identified to be beyond the specified inspection frequency (AR 04310888). On January 24, 2020, NOS issued an elevated finding (AR 04312522) for continued lack of rigor and low sense of urgency to address indications of non-compliance with procedural requirements that contributed to on-going fire penetration seals inspection implementation gaps. Exelon's evaluation determined there were several causal factors that resulted in the deficiencies identified by NOS. These included deficiencies in the inspection scoping process (which was maintained manually and without peer reviews), documentation errors in completed inspection records, and incorrect equipment database. The inspectors review of corrective actions noted that Exelon staff formed a cross-discipline fire penetration seals recovery team comprised of design engineers, site and corporate fire protection program engineers, and maintenance to work through recovery actions and extent of condition reviews to address all the deficiencies identified by NOS. The inspectors reviewed the associated ARs and NOS information to assess whether the issue was accurately documented, evaluated, and to verify corrective actions have been timely and adequate. The inspectors also reviewed the engineering assessment to validate the assumptions and conclusions were supported, interviewed personnel including the fire protection system engineer, the mechanical design engineer, and the fire protection program manager, and performed independent fire seals inspections in safety-related areas of both Units 2 and 3 including; the switch gear rooms, battery rooms, cable spreading rooms, and the EDG rooms. No walkdown seal issues of significance were identified. The inspectors concluded the issue that resulted in the NOS findings were evaluated sufficiently to identify the causes and develop effective corrective actions, and that the extent of condition reviews were adequate. The inspectors also noted NOS performed a final review of the corrective actions implemented and closed out their findings on March 2, 2020. Corrective actions included: extent of condition reviews, employee training, complete re-baseline of the fire penetration seals inspection program, re-evaluation and creation of a new seal data base which included removal of all penetration seals determined to be inaccessible, revision of procedures to properly include the new list of all seals required to be inspected, and	

inspection of seals identified as not inspected. The inspectors did not identify any findings of more than minor significance.

Observation: 71152 Semi-Annual Trend Review	71152
The inspectors conducted a semi-annual trend review by evaluating a sample of issues that occurred in the first and second quarters of 2021. During the evaluation, the inspectors verified the issues identified were addressed within the scope of the CAP. The inspectors reviewed health reports and related databases for trends and considered prior issues while performing routine walkdowns and attending the plan of the day meetings. No substantive adverse performance trends or repetitive equipment failures were identified during this time. However, the inspectors' trend review noted some equipment failures during the quarter with similarity and overlapping risk. In particular, there were two failures of power supply inverters during the period, one in the Unit 3 RCIC system and one in the Unit 2 high-pressure injection system, and a failure of nitrogen supply to the Unit 2 '71K' safety-relief valve due to hose fretting. The inspectors noted that aggregate equipment failures can represent greater cumulative combined risk relative to the simple sum of the risk from individual events. As a result, systems and components may warrant additional focus to improve maintenance effectiveness when considering these impacts. Exelon's evaluations were not yet complete at the end of the quarter. However, the inspectors reviewed Exelon's initially planned actions and confirmed that the planned evaluations were robust, manufacturer and/or laboratory inputs were sought, the immediate corrective actions were appropriate, and effectiveness reviews were included when warranted. The inspectors also observed a number of issues in the plant that was higher than recent prior periods. The inspectors shared an observation with Exelon that many of the issues were reasonably identifiable by their staff during routine activities. As one example, the inspectors identified a step-ladder designated for emergency use only, via an attached placard, which had been removed from its storage location and left setup in a standing position in a normally traversed room. The condition represented multiple breakdowns in Exelon's processes: a failure to adhere to the posted signage by using the ladder, a failure to adhere to station procedures to take down the ladder and lay it down and/or restrain it when not in use, a failure to adhere to station procedures to return the ladder to an approved storage location when work at the jobsite was complete, and a failure of personnel traversing through the room to question the ladder being upright and unattended for a period of time and still displaying the placard designating the ladder for emergency use only. The inspectors determined that all of the issues were of minor safety significance with no adverse impact to a cornerstone objective. However, the inspectors shared an observation regarding the potential for an adverse trend in a lack of attention to detail and non-compliance with procedures. In addition to addressing the individual issues, Exelon initiated an issue report to document a potential emerging trend and perform an analysis in order to improve station performance. Exelon also issued site-wide communications and engaged supervisors and staff to improve performance. The inspectors noted improved performance afterwards. Based on the overall results of the semi-annual trend review, the inspectors determined that Exelon had identified adverse trends at Peach Bottom Atomic Power Station before they could become more significant safety problems. The inspectors continue to monitor the CAP and maintenance effectiveness during routine inspection activities.	

EXIT MEETINGS AND DEBRIEFS

The inspectors verified no proprietary information was retained or documented in this report.

- On July 16, 2021, the inspectors presented the integrated inspection results to Mr. Matthew Herr, Site Vice President, and other members of the licensee staff.
- On May 26, 2021, the inspectors presented the PI&R Sample, Fire Penetration Seals Issues inspection results to Mr. Mark Parrish, Design Engineering Manager Lead, and other members of the licensee staff.

DOCUMENTS REVIEWED

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
71111.04	Corrective Action Documents	04422956		
		04426956		
		04428901		
	Procedures	M-359	RCIC System	Revision 50
		SO 13.1.A-3 COL	RCIC System	Revision 17
		SO 13.1.B-3 COL	RCIC System Control Board Lineup	Revision 3
71111.05	Procedures	PF-127	Unit 2 Turbine Building, Emergency Battery Switchgear Rooms, Elevation 135'0"	Revision 11
		PF-132	Diesel Generator Building, General Area, Elevation 127'0"	Revision 9
		PF-78H	Turbine Building Common, Cable Spreading and Computer Rooms	Revision 9
71111.06	Corrective Action Documents	4424351		
	Miscellaneous	EC 634394		
	Procedures	AO 20A.1	Temporary Removal and Installation of Flood Barriers on the Reactor Building Drainage System	Revision 18
71111.15	Corrective Action Documents	4422956		
		4416510		
		4422713 4422952		
	Corrective Action Documents Resulting from Inspection	04412724		
	Miscellaneous	EC 634268		
71111.18	Calculations	87RE/CONN	Design of Additional Support for the Connection Between Block Wall and Concrete Wall	Revision 3
	Corrective Action Documents	4422138		
	Miscellaneous	EC 624203, XJ-70133D/134D	Exp. Joint Misalign Measurement Out of Tolerance	Revision 0
71111.19	Corrective Action	4431791		

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
	Documents	4431792 4431794		
	Procedures	RT-O-052-253-2	'E-3' Diesel Generator Inspection Post-Maintenance Functional Test	Revision 44
	Work Orders	04325592 05120294 05124552 05126927		
		5136414		
71111.22	Work Orders	5128336		
71114.06	Procedures	EP-AA-1007, Addendum 3	Emergency Action Levels for Peach Bottom Atomic Power Station	Revision 9
71152	Corrective Action Documents	04195447		
		04310888		
		04312522		
		04378513		
	Corrective Action Documents Resulting from Inspection	04421136		
		04422880		
		4395106 4399049 4401017 4405806 4410976 4413233 4420472 4420571 4426780 4428531 4428772 4430092		
	Miscellaneous	Technical Evaluation 1291255-01	Justification for not Inspecting Inaccessible Penetration Seals	01

September 1, 2021 – Letter from Jonathan E. Greives, Chief Projects Branch 4 Division of Operating Reactor Safety to David P. Rhoades Senior Vice President Exelon Generation Company, LLC President and Chief Nuclear Officer Exelon Nuclear with subject of UPDATED INSPECTION PLAN FOR THE PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 (REPORT 05000277/2021005 AND 05000278/2021005)

The enclosed inspection plan lists the inspections scheduled through June 30, 2023, for Peach Bottom Atomic Power Station, Units 2 and 3. The U.S. Nuclear Regulatory Commission (NRC) provides the inspection plan to allow for the resolution of any scheduling conflicts and personnel availability issues. Routine inspections performed by resident inspectors are not included in the inspection plan. The inspections listed during the last twelve months of the inspection plan are tentative and may be revised. The NRC will contact you as soon as possible to discuss changes to the inspection plan should circumstances warrant any changes.

In response to the COVID-19 public health emergency (PHE), the NRC is adjusting inspection plans and schedules in order to safeguard the health and safety of both NRC and licensee staff while still effectively implementing the Reactor Oversight Process (ROP). Each planned inspection is being carefully reviewed in order to determine if any portions of the inspection can be performed remotely, determine how best to perform on-site portions to minimize personnel health risks, and adjust inspection schedules if needed. This is done in accordance with guidance contained in the February 1, 2021 memo, “Calendar Year 2021 Inspection Guidance During COVID-19 Telework Restrictions” (ML21027A274). For inspections requiring extensive coordination with offsite organizations, such as evaluated emergency preparedness exercises, NRC guidance and frequently asked questions for security and emergency preparedness can be found here: https://www.nrc.gov/about-nrc/covid-19/security-ep/. Similarly, the NRC has developed guidance if force-on-force inspections cannot be completed as scheduled due to an emergency, such as the COVID-19 PHE. These changes help ensure the health and safety of both NRC and licensee staff while maintaining the NRC’s important safety and security mission during the COVID-19 PHE. The attached inspection plan is accurate on the date of issuance but remains subject to change based on approval of potential exemption requests or other changes needed due to changing conditions in the COVID-19 PHE. NRC staff will contact your appropriate regulatory affairs staff in order to coordinate inspection planning and scheduling.

The NRC will also be performing baseline inspections of licensee cyber security programs. The specific schedule and procedure to be used for these inspections is being developed by the NRC staff. The staff expects to communicate the schedule for these inspections to each utility separately.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390 of the NRC’s “Rules of Practice,” a copy of this letter will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Please contact me at 610-337-5337 with any questions you have regarding this letter.

IP 22b Site Inspection Activity Plan

Peach Bottom

07/01/2021 - 06/30/2023

Unit	Start	End	Activity	CAC	Title	Staff Count
Annual PIR Sample: 2B RHR MOV Anti-rotation Key Failure						1
2	7/19/2021	7/23/2021	IP 71152	000748	Problem Identification and Resolution	
Radwaste (71124.08)						1
2, 3	7/26/2021	7/30/2021	IP 71124.08	000732	Radioactive Solid Waste Processing & Radioactive Material Handling, Storage, & Transportation	
PI&R BIENNIAL - PB						4
2, 3	7/26/2021	7/30/2021	IP 71152B	000747	Problem Identification and Resolution	
2, 3	8/2/2021	8/6/2021	IP 71152B	000747	Problem Identification and Resolution	
2, 3	8/9/2021	8/13/2021	IP 71152B	000747	Problem Identification and Resolution	
HP Instrumentation (71124.05)						1
2, 3	9/13/2021	9/17/2021	IP 71124.05	000729	Radiation Monitoring Instrumentation	
INSERVICE INSPECTION - UNIT 3						1
3	10/31/2021	11/6/2021	IP 71111.08G	000701	Inservice Inspection Activities (BWR)	
Rad Hazards (71124.01) and PIs (71151)						1
2, 3	11/1/2021	11/5/2021	IP 71124.01	000725	Radiological Hazard Assessment and Exposure Controls	
2, 3	11/1/2021	11/5/2021	IP 71151	000746	Performance Indicator Verification	
ALARA (71124.02)						1
2, 3	11/29/2021	12/3/2021	IP 71124.02	000726	Occupational ALARA Planning and Controls	
Design Basis Assurance Inspection - Power Operated Valves - Peach Bottom Units 2 and 3						3
2, 3	3/21/2022	4/8/2022	IP 71111.21N.02	001645	Design-Basis Capability of Power-Operated Valves Under 10 CFR 50.55a Requirements	
Resp Protection & Dose Assessment (71124.03/71124.04)						1
2, 3	4/18/2022	4/22/2022	IP 71124.03	000727	In-Plant Airborne Radioactivity Control and Mitigation	
2, 3	4/18/2022	4/22/2022	IP 71124.04	000728	Occupational Dose Assessment	
EP EXERCISE INSPECTION - PEACH BOTTOM						4
2, 3	4/25/2022	4/29/2022	IP 71114.07	000722	Exercise Evaluation - Hostile Action (HA) Event	
2, 3	4/25/2022	4/29/2022	IP 71151	001397	Performance Indicator Verification	
Access Control, Protective Strat, TSR, SPR, PI-PB						4
2, 3	5/16/2022	5/20/2022	IP 71130.02	000734	Access Control	
2, 3	5/16/2022	5/20/2022	IP 71130.05	000737	Protective Strategy Evaluation	
2, 3	5/16/2022	5/20/2022	IP 71130.09	001656	Security Plan Changes	
2, 3	5/16/2022	5/20/2022	IP 71130.14	000743	Review of Power Reactor Target Sets	
2, 3	5/16/2022	5/20/2022	IP 71151	001338	Performance Indicator Verification	
FY22 Peach Bottom Initial License Examination						4

IP 22b Site Inspection Activity Plan

2, 3	4/24/2022	4/29/2022	OV	000956	FB-OR-ONSITE VALIDATION OF INITIAL LICENSE EXAMINATION (OV)	
2, 3	5/22/2022	5/27/2022	EXAD	000500	FB-OR-INITIAL LICENSE EXAM ADMINISTRATION (EXAD)	
50.59 Inspection - Peach Bottom Units 2 and 3						3
2, 3	6/13/2022	6/17/2022	IP 71111.17T	000709	Evaluations of Changes, Tests, and Experiments	
RETS (71124.06)						1
2, 3	7/18/2022	7/22/2022	IP 71124.06	000730	Radioactive Gaseous and Liquid Effluent Treatment	
Material Control and Accountability - Peach Bottom Units 2 and 3						1
2, 3	8/22/2022	8/26/2022	IP 71130.11	000742	Material Control and Accounting (MC&A)	
Triennial Heat Sink - Peach Bottom Units 2 and 3						1
2, 3	9/19/2022	9/23/2022	IP 71111.07T	000700	Heat Sink Performance	
Inservice Inspection - Peach Bottom Unit 2						1
2	10/24/2022	10/28/2022	IP 71111.08G	000701	Inservice Inspection Activities (BWR)	
Rad Hazards (71124.01) and PIs (71151)						1
2, 3	10/24/2022	10/28/2022	IP 71124.01	000725	Radiological Hazard Assessment and Exposure Controls	
2, 3	10/24/2022	10/28/2022	IP 71151	000746	Performance Indicator Verification	
EP PROGRAM INSPECTION - PEACH BOTTOM						1
2, 3	2/27/2023	3/3/2023	IP 71114.02	000717	Alert and Notification System Testing	
2, 3	2/27/2023	3/3/2023	IP 71114.03	000718	Emergency Response Organization Staffing and Augmentation System	
2, 3	2/27/2023	3/3/2023	IP 71114.04	000719	Emergency Action Level and Emergency Plan Changes	
2, 3	2/27/2023	3/3/2023	IP 71114.05	000720	Maintenance of Emergency Preparedness	
2, 3	2/27/2023	3/3/2023	IP 71151	001397	Performance Indicator Verification	
PB Requal Inspection with P/F Results						2
2, 3	3/5/2023	3/10/2023	IP 71111.11A	000703	Licensed Operator Requalification Program and Licensed Operator Performance	
2, 3	3/5/2023	3/10/2023	IP 71111.11B	000704	Licensed Operator Requalification Program and Licensed Operator Performance	
AA, FFD-PB						2
2, 3	3/6/2023	3/10/2023	IP 71130.01	000733	Access Authorization	
2, 3	3/6/2023	3/10/2023	IP 71130.08	000740	Fitness For Duty Program	
Design Basis Assurance Inspection Teams - Peach Bottom						6
2, 3	4/10/2023	4/28/2023	IP 71111.21M	000713	Design Bases Assurance Inspection (Teams)	
HP Instrumentation (71124.05)						1
2, 3	5/15/2023	5/19/2023	IP 71124.05	000729	Radiation Monitoring Instrumentation	
FY23 Peach Bottom Initial License Exam						4
2, 3	4/30/2023	5/5/2023	OV	000956	FB-OR-ONSITE VALIDATION OF INITIAL LICENSE EXAMINATION (OV)	
2, 3	5/28/2023	6/9/2023	EXAD	000500	FB-OR-INITIAL LICENSE EXAM ADMINISTRATION (EXAD)	

<u>September 3, 2021</u> – Letter from Nancy L. Salgado, Chief Plant Licensing Branch III Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to David P. Rhoades Senior Vice President Exelon Generation Company, LLC President and Chief Nuclear Officer (CNO) Exelon Nuclear with subject of BRAIDWOOD STATION, UNITS 1 AND 2; CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 AND 2;

CLINTON POWER STATION, UNIT NO. 1; LIMERICK GENERATING STATION, UNITS 1 AND 2; NINE MILE POINT NUCLEAR STATION, UNITS 1 AND 2; PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3; AND R. E. GINNA NUCLEAR POWER PLANT — PROPOSED ALTERNATIVE TO USE ASME OM CODE CASE OMN-28 (EPID L-2021-LLR-0056)

By application dated August 5, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21217A117), Exelon Generation Company, LLC (Exelon, the licensee) submitted a request in accordance with paragraph 50.55a(z)(1) of Title 10 of the *Code of Federal Regulations* (10 CFR) for a proposed alternative to certain requirements of

10 CFR 50.55a, "Codes and standards," for Braidwood Station (Braidwood), Units 1 and 2; Calvert Cliffs Nuclear Power Plant (Calvert Cliffs), Units 1 and 2; Clinton Power Station (Clinton), Unit No. 1; Limerick Generating Station (Limerick), Units 1 and 2; Nine Mile Point Nuclear Station (NMP), Units 1 and 2; Peach Bottom Atomic Power Station (Peach Bottom), Units 2 and 3; and R. E. Ginna Nuclear Power Plant (Ginna) (collectively, the facilities).

The American Society of Mechanical Engineers (ASME), *Operation and Maintenance of Nuclear Power Plants*, Division 1, Section IST (OM Code), as incorporated by reference in 10 CFR 50.55a, specifies requirements for the inservice testing (IST) of nuclear power plant components. Exelon requests to use the ASME OM Code Case OMN-28, "Alternative Valve Position Verification Approach to Satisfy ISTC-3700 for Valves Not Susceptible to Stem-Disk Separation," as an alternative to the IST requirements in the 2012 Edition of the ASME OM Code, as supplemented by 10 CFR 50.55a, for certain specified valves at its facilities.

The regulations in 10 CFR 50.55a(z) state, in part, that alternatives to the requirements in paragraphs (b) through (h) of 10 CFR 50.55a may be authorized by the U.S. Nuclear Regulatory Commission (NRC) if the licensee demonstrates that: (1) the proposed alternative provides an acceptable level of quality and safety, or (2) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The NRC staff has reviewed Exelon's application and concludes, as set forth in the enclosed safety evaluation, that the licensee has adequately addressed the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC staff authorizes the licensee to use the proposed alternative to implement the ASME OM Code Case OMN-28 in its entirety, as specified in its August 5, 2021, application, for the verification of valve position indication for valves identified as having position indication requirements (referred to as PI requirements) in the updated IST Program Plans referenced below that are not susceptible to stem-disk separation, in lieu of the requirements in the ASME OM Code (2012 Edition), Subsection ISTC, paragraph ISTC-3700, as incorporated by reference in 10 CFR 50.55a and supplemented by 10 CFR 50.55a(b)(3)(xi). This authorization is for the remainder of the current 10-year IST intervals at Braidwood, Units 1 and 2; Calvert Cliffs, Units 1 and 2; Clinton, Unit No. 1; Limerick, Units 1 and 2; NMP, Units 1 and 2; Peach Bottom, Units 2 and 3; and Ginna.

The updated IST Program Plans applicable to the current 10-year IST intervals for each facility are the following:

1. Braidwood, Units 1 and 2, “Inservice Testing Program, Fourth Ten Year Interval,” Revision 4, submitted by letter dated April 28, 2021 (ADAMS Package Accession No. ML21118A009);
2. Calvert Cliffs, Units 1 and 2, “Inservice Testing (IST) Program Plan, Fifth Ten-Year Interval,” Revision 00, submitted by letter dated July 6, 2018 (ADAMS Accession No. ML18192B990);
3. Clinton, Unit No. 1, “Inservice Testing (IST) Program Plan, 4th Ten-Year Interval,” Revision 1, submitted by letter dated May 23, 2021 (ADAMS Accession No. ML21225A189);
4. Limerick, Units 1 and 2, “Inservice Testing (IST) Program Plan, Fourth Ten-Year Interval,” Revision 28, submitted by letter dated March 3, 2021 (ADAMS Accession No. ML21062A050);
5. NMP, Units 1 and 2 “Inservice Testing (IST) Program Plan, Unit 1 Fifth 10-Year Interval, Unit 2 Fourth 10-Year Interval,” Revision 09, submitted by letter dated March 13, 2019 (ADAMS Accession No. ML19072A182);
6. Peach Bottom, Units 2 and 3, “Inservice Testing (IST) Program Plan, 5th Ten-Year Interval,” Revision 006, submitted by letter dated November 29, 2018 (ADAMS Accession No. ML18337A196); and
7. Ginna, “Inservice Testing (IST) Program Plan, Sixth 10-Year Interval,” Revision 0, submitted by letter dated February 5, 2020 (ADAMS Accession No. ML20036C593).

All other ASME OM Code requirements, as incorporated by reference in 10 CFR 50.55a, for which relief or an alternative was not specifically requested, and granted or authorized (as appropriate), in the subject request remain applicable.

If you have any questions, please contact Blake Purnell at 301-415-1380 or via e-mail at

Blake.Purnell@nrc.gov.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
PROPOSED ALTERNATIVE TO USE ASME OM CODE CASE OMN-28 BRAIDWOOD STATION, UNITS 1 AND 2
CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 AND 2 CLINTON POWER STATION, UNIT NO. 1
LIMERICK GENERATING STATION, UNITS 1 AND 2
NINE MILE POINT NUCLEAR STATION, UNITS 1 AND 2
PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3
R.E. GINNA NUCLEAR POWER PLANT
EXELON GENERATION COMPANY, LLC
DOCKET NOS. STN 50-456, STN 50-457, 50-317, 50-318, 50-461
50-352, 50-353, 50-220, 50-410, 50-277, 50-278, AND 50-244

INTRODUCTION

By application dated August 5, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21217A117), Exelon Generation Company, LLC

(Exelon, the licensee) submitted a request in accordance with paragraph 50.55a(z)(1) of Title 10 of the *Code of Federal Regulations* (10 CFR) for a proposed alternative to certain requirements of

10 CFR 50.55a, “Codes and standards,” for Braidwood Station (Braidwood), Units 1 and 2; Calvert Cliffs Nuclear Power Plant (Calvert Cliffs), Units 1 and 2; Clinton Power Station (Clinton), Unit No. 1; Limerick Generating Station (Limerick), Units 1 and 2; Nine Mile Point Nuclear Station (NMP), Units 1 and 2; Peach Bottom Atomic Power Station (Peach Bottom), Units 2 and 3; and R. E. Ginna Nuclear Power Plant (Ginna) (collectively, the facilities).

The American Society of Mechanical Engineers (ASME), *Operation and Maintenance of Nuclear Power Plants*, Division 1, Section IST (OM Code), as incorporated by reference in 10 CFR 50.55a, specifies requirements for the inservice testing (IST) of nuclear power plant components. Exelon requests to use the ASME OM Code Case OMN-28, “Alternative Valve Position Verification Approach to Satisfy ISTC-3700 for Valves Not Susceptible to Stem-Disk Separation,” as an alternative to the IST requirements in the 2012 Edition of the ASME OM Code, as supplemented by 10 CFR 50.55a, for certain specified valves at its facilities.

2.0 REGULATORY EVALUATION

The regulations in 10 CFR 50.55a(f)(4) state, in part, that throughout the service life of a boiling or pressurized water-cooled nuclear power facility, pumps and valves that are within the scope of the ASME OM Code must meet the IST requirements (except design and access provisions) set forth in the ASME OM Code and addenda that become effective subsequent to editions and addenda specified in 10 CFR 50.55a(f)(2) and (3) and that are incorporated by reference in

10 CFR 50.55a(a)(1)(iv), to the extent practical within the limitations of design, geometry, and materials of construction of the components. The 2012 edition of the ASME OM Code, as incorporated by reference in 10 CFR 50.55a with conditions, is applicable to the current 10-year IST intervals at the facilities.

The NRC regulations in 10 CFR 50.55a(b)(3)(xi), “OM condition: Valve Position Indication,” state the following:

When implementing paragraph ISTC-3700, “Position Verification Testing,” in the ASME OM Code, 2012 Edition through the latest edition and addenda of the ASME OM Code incorporated by reference in paragraph (a)(1)(iv) of this section [10 CFR 50.55a], licensees shall verify that valve operation is accurately indicated by supplementing valve position indicating lights with other indications, such as flow meters or other suitable instrumentation to provide assurance of proper obturator position for valves with remote position indication within the scope of Subsection ISTC including its mandatory appendices and their verification methods and frequencies.

The regulations in 10 CFR 50.55a(z) state, in part, that alternatives to the requirements in paragraphs (b) through (h) of 10 CFR 50.55a may be authorized by the NRC if the licensee demonstrates that: (1) the proposed alternative provides an acceptable level of quality and safety, or (2) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

1. 3.0 TECHNICAL EVALUATION
2. 3.1 Licensee's Request

3.1.1 ASME Code Components Affected

In its application, the licensee states that the valves covered by ASME OM Code Case OMN-28 are those stem-disk separation non-susceptible valves with remote position indication within the scope of Subsection ISTC of the ASME OM Code (2012 Edition) including its mandatory appendices and their verification methods and frequencies, in accordance with regulatory requirements. The licensee notes that a listing of the valves requiring position indication and testing in accordance with ASME OM Code, Subsection ISTC, paragraph ISTC-3700, was submitted with the IST Program update performed as part of the interval update and latest revisions for each facility. The latest revision of the updated IST Program Plans submitted to the NRC for the current 10-year IST intervals at each facility are referenced in Table 1 below.

3.1.2 Applicable Code Edition and Addenda

The 2012 edition of the ASME OM Code, as incorporated by reference in 10 CFR 50.55a with conditions, is applicable to the current 10-year IST intervals at the facilities. The current IST interval, including the start and end dates, and latest revision of the updated IST Program Plan for each plant is provided in Table 1 below.

Table 1: Current IST Interval and Program Plan Information.

Plant	IST Interval	Start	End	IST Program Plan Submittal Date and ADAMS Accession No.
Braidwood, Units 1 and 2	4th	7/29/2018	7/28/2028	4/28/2021 ML21118A009 (Package)
Calvert Cliffs, Units 1 and 2	5th	7/1/2018	6/30/2028	7/6/2018 ML18192B990
Clinton, Unit No. 1	4th	7/1/2020	6/30/2030	5/23/2021 ML21225A189
Limerick, Units 1 and 2	4th	1/8/2020	1/7/2030	3/3/2021 ML21062A050
NMP, Unit 1	5th	1/1/2019	12/31/2028	3/13/2019 ML19072A182
NMP, Unit 2	4th	1/1/2019	12/31/2028	3/13/2019 ML19072A182
Peach Bottom, Units 2 and 3	5th	11/16/2018	8/14/2028	11/29/2018 ML18337A196
Ginna	6th	1/1/2020	12/31/2029	2/5/2020 ML20036C593

3.1.3 Applicable Code Requirements

Paragraph ISTC-3700, "Position Verification Testing," of the ASME OM Code (2012 Edition) states:

> Valves with remote position indicators shall be observed locally at least once every 2 yr [years] to verify that valve operation is accurately indicated. Where practicable, this local observation should be supplemented by other indications such as use of flow meters or other suitable instrumentation to verify obturator position. These observations need not be concurrent. Where local observation is not possible, other indications shall be used for verification of valve operation.
>
> Position verification for active MOVs [motor-operated valves] shall be tested in accordance with Mandatory Appendix III of this Division.

As noted in Section 2.0 of this safety evaluation (SE), when implementing this paragraph, "licensees shall verify that valve operation is accurately indicated by supplementing valve position indicating lights with other indications, such as flow meters or other suitable instrumentation to provide assurance of proper obturator position for valves with remote position indication within the scope of Subsection ISTC [of the ASME OM Code] including its mandatory appendices and their verification methods and frequencies."

3.1.4 Licensee's Proposed Alternative, Reason for Request, and Basis for Use

The licensee's proposed alternative is to use ASME OM Code Case OMN-28 (approved for use by ASME on March 4, 2021) in lieu of the requirements in paragraph ISTC-3700 of the ASME OM Code for the specific valves described in Section 3.1.1 of this SE. The licensee did not propose any deviations from the code case. The licensee stated that implementation of Code Case OMN-28 would provide an acceptable level of quality and safety in accordance with

10 CFR 50.55a(z)(1).

The licensee stated: “The position verification with Supplemental Position Indication (SPI) requires the valves to be exercised in the open and closed direction and the valve’s position verified by other indications such as use of flow meters or other suitable instrumentation to verify obturator position.” The licensee also stated that Code Case OMN-28 “has been determined to satisfy the valve position verification requirements in ASME OM Code, Subsection ISTC, paragraph ISTC-3700, for valves that are not susceptible to stem-disk separation.”

Valves with remote position indication within the scope of ASME OM Code, Subsection ISTA, paragraph ISTA-1100, “Scope,” not satisfying the scope and provisions of Code Case OMN-28 shall meet the valve position verification requirements in ASME OM Code, Subsection ISTC, paragraph ISTC-3700, in accordance with the regulatory requirements.

3.2 NRC Staff’s Evaluation

The NRC staff reviewed the provisions in the ASME OM Code Case OMN-28 used to demonstrate that the remote position indicators for valves that are not susceptible to stem-disk separation accurately represent valve operation (open and closed). The code case requires remote position verification for valves that are not susceptible to stem-disk separation to include: (a) observation of evidence, such as changes in system pressure, flow rate, level, or temperature, that represent valve operation; (b) local observation of valve operation where practicable; and (c) stem-disk separation evaluation shall be documented and available for regulatory review demonstrating that the stem-disk connection is not susceptible to separation. For active valves not susceptible to stem-disk separation, the code case states that these observations shall be performed at least once every 12 years. For passive valves not susceptible to stem-disk separation, the code case states that these observations shall be performed whenever the valve is stroked from its passive position or every 12 years, whichever is greater.

The licensee proposes to implement the ASME OM Code Case OMN-28 in its entirety, without any deviations, for the specific valves described Section 3.1.1 of this SE. Based on the review of the provisions in the code case, the NRC staff has reasonable assurance that the remote position indicators for these specific valves will be properly verified to accurately represent valve operation (open and closed). Therefore, the NRC staff finds that, for these specific valves, the implementation of the proposed alternative at each facility provides an acceptable level of quality and safety in accordance with 10 CFR 50.55a(z)(1).

4.0 CONCLUSION

As set forth above, the NRC staff determined that the licensee’s proposed alternative to implement the ASME OM Code Case OMN-28 in its entirety, without any deviations, for the specified valves provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC staff authorizes the licensee to use the proposed alternative to implement the ASME OM Code Case OMN-28 in its entirety, as specified in its August 5, 2021, application, for the verification of valve position indication for valves identified as having position indication requirements (referred to as PI requirements) in the IST Program Plans listed in Table 1 that are not susceptible to stem-disk separation, in lieu of the requirements in the ASME OM Code (2012 Edition), Subsection ISTC, paragraph ISTC-3700, as incorporated by reference in 10 CFR 50.55a and supplemented by 10 CFR 50.55a(b)(3)(xi). This authorization is for the remainder of the current 10-year IST intervals at Braidwood, Units 1 and 2; Calvert Cliffs, Units 1 and 2; Clinton, Unit No. 1; Limerick, Units 1 and 2; NMP, Units 1 and 2; Peach Bottom, Units 2 and 3; and Ginna.

All other ASME OM Code requirements, as incorporated by reference in 10 CFR 50.55a, for which relief or an alternative was not specifically requested, and granted or authorized (as appropriate), in the subject request remain applicable.

Principal Contributors: Thomas G. Scarbrough, NRR Date of issuance: September 3, 2021

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2; CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 AND 2; CLINTON POWER STATION, UNIT NO. 1; LIMERICK GENERATING STATION, UNITS 1 AND 2; NINE MILE POINT NUCLEAR STATION, UNITS 1 AND 2; PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3; AND R. E. GINNA NUCLEAR POWER PLANT — PROPOSED ALTERNATIVE TO USE ASME OM CODE CASE OMN-28 (EPID L-2021-LLR-0056) DATED SEPTEMBER 3, 2021

DISTRIBUTION:
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RidsNrrPMBraidwood Resource
RidsNrrPMCalvertCliffs Resource
RidsNrrPMClinton Resource
RidsNrrPMLimerick Resource
RidsNrrPMNineMilePoint Resource
RidsNrrPMPeachBottom Resource
RidsNrrPMREGinna Resource
RidsNrrDexEmib Resource
RRichardson, EDO
TScarbrough, NRR
BWolfgang, NRR

ADAMS Accession No. ML21230A206

OFFICE	NRR/DORL/LPL3/PM	NRR/DORL/LPL3/LA	NRR/DEX/EMIB/BC(A)
NAME	BPurnell	SRohrer	ITseng
DATE	08/30/2021	08/19/2021	08/30/2021
OFFICE	NRR/DORL/LPL3/BC		
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DATE	09/03/2021		

OFFICIAL RECORD COPY

<u>September 7, 2021</u> – Letter from Jason C. Paige, Project Manager Plant Licensing Branch I Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to David P. Rhoades Senior Vice President Exelon Generation Company, LLC President and Chief Nuclear Officer Exelon Nuclear with subject of PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 – CLOSEOUT OF BULLETIN 2012-01, "DESIGN VULNERABILITY IN ELECTRIC POWER SYSTEM"

The purpose of this letter is to inform you that the U.S. Nuclear Regulatory Commission (NRC) staff has verified that Exelon Generation Company, LLC (the licensee) has provided the necessary information requested in NRC's Bulletin (BL) 2012-01, "Design Vulnerability in Electric Power System," dated July 27, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12074A115), for Peach Bottom Atomic Power Station. The NRC staff has completed its review of this information and has closed out BL 2012-01 for this facility.

The NRC issued BL 2012-01 on July 27, 2012, to all holders of operating licenses and combined licenses for nuclear power reactors, except those who have permanently ceased operation and have certified that fuel has been removed from the reactor vessel. The Bulletin requested information about each facility's electric power system designs that would allow the NRC staff to verify the system's capability to address open phase conditions. Specifically, the NRC requested licensees to provide the following information:

- • A description of how the protection scheme for engineered safety features buses
  (Class 1E for current operating plants or non-Class 1E for passive plants) is designed to detect and automatically respond to a single-phase open circuit condition or high impedance ground fault condition on offsite power circuits or another power source; and
- • A description of the operating configuration of engineered safety features buses (Class 1E for current operating plants or non-Class 1E for passive plants) at power (i.e., normal operating condition).

  By letter dated October 25, 2012 (ADAMS Accession No. ML12300A106), the licensee provided its response to BL 2012-01 for Peach Bottom Atomic Power Station. By letter dated February 3, 2014 (ADAMS Accession No. ML14034A179), the license provided supplemental information for this facility in response to an NRC staff request for additional information issued on

  December 20, 2013 (ADAMS Accession No. ML13351A314).

By letters dated October 9, 2013, and March 16, 2015 (ADAMS Accession Nos. ML13333A147 and ML15075A454, respectively), the Nuclear Energy Institute (NEI) submitted a voluntary industry initiative to address open phase conditions at nuclear power plants. The NEI letter dated March 16, 2015, stated, in part: "The initiative is a formal commitment by the companies that operate nuclear power plants to follow a specific policy or plan of action. The initiative calls for a proactive plan and schedule for addressing potential design vulnerabilities to the open phase condition."

To evaluate the adequacy of the open phase isolation systems designs, the NRC staff inspected four nuclear power plants with four distinct open phase isolation system designs using the NRC Temporary Instruction (TI) 2515/194, "Inspection of the Licensees' Implementation of Industry Initiative Associated with the Open Phase Condition Design Vulnerabilities in Electric Power Systems (NRC Bulletin 2012-01)," dated October 31, 2017 (ADAMS Accession

No. ML17137A416). A summary of the NRC staff's preliminary observations and issues needing additional clarity were discussed with industry representatives in two public meetings conducted on September 19, 2018, and October 17, 2018. The meeting summaries can be found in ADAMS under Package Accession Nos. ML18268A342 and ML18309A226, respectively.

By letter dated June 6, 2019 (ADAMS Accession No. ML19163A176), NEI submitted Revision 3 of the voluntary industry initiative to include an option for plants to perform a risk evaluation under certain boundary conditions to support manual response to an open phase condition. NEI also submitted NEI 19-02, "Guidance for Assessing Open Phase Condition Implementation Using Risk Insights" on June 20, 2019 (ADAMS Accession No. ML19172A086).

In March 2021, the NRC staff performed an inspection using TI 2515/194, Revision 3, at Peach Bottom Atomic Power Station to verify the licensee's implementation of the voluntary industry initiative at this facility. To address the open phase condition design vulnerability issue at this facility, the licensee implemented open phase isolation system plant modifications, which provide detection and alarm in the control room, and necessary plant procedures that allow operators to diagnose and take manual action to mitigate an open phase condition. The NRC inspection report listed below documents the results of this TI 2515/194 inspection.

- Peach Bottom Atomic Power Station, Units 2 and 3 – Temporary Instruction 2515/194 Inspection Report 05000277/2021011 and 05000278/2021011, dated April 1, 2021 (ADAMS Accession No. ML21091A149)

The NRC staff reviewed the information submitted by the licensee and the results of the TI 2515/194 inspection for Peach Bottom Atomic Power Station. The deviations/exceptions regarding the voluntary industry initiative implementation at this facility were considered as minor or insignificant. Based on this review, the NRC staff concludes that the licensee provided the necessary information requested in BL 2012-01 and has completed the implementation of its open phase isolation system. Therefore, the NRC staff has closed BL 2012-01 for Peach Bottom Atomic Power Station.

If you have any questions, please contact me at 301-415-1474 or by e-mail to

Jason.Paige@nrc.gov

September 22, 2021 – Letter from Jonathan E. Greives, Chief Projects Branch 4 Division of Operating Reactor Safety to David P. Rhoades Senior Vice President Exelon Generation Company, LLC President and Chief Nuclear Officer Exelon Nuclear with subject of PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 –

BIENNIAL PROBLEM IDENTIFICATION AND RESOLUTION INSPECTION REPORT 05000277/2021012 AND 05000278/2021012

On August 12, 2021, the U.S. Nuclear Regulatory Commission (NRC) completed a problem identification and resolution inspection at your Peach Bottom Atomic Power Station, Units 2 and 3 and discussed the results of this inspection with Mr. David Henry and other members of your staff. The results of this inspection are documented in the enclosed report.

The NRC inspection team reviewed the station's corrective action program and the station's implementation of the program to evaluate its effectiveness in identifying, prioritizing, evaluating, and correcting problems, and to confirm that the station was complying with NRC regulations and licensee standards for corrective action programs. Based on the samples reviewed, the team determined that your staff's performance in each of these areas adequately supported nuclear safety.

The team also evaluated the station's processes for use of industry and NRC operating experience information and the effectiveness of the station's audits and self-assessments. Based on the samples reviewed, the team determined that your staff's performance in each of these areas adequately supported nuclear safety.

Finally the team reviewed the station's programs to establish and maintain a safety conscious work environment, and interviewed station personnel to evaluate the effectiveness of these programs. Based on the team's observations and the results of these interviews the team found no evidence of challenges to your organization's safety-conscious work environment. Your employees appeared willing to raise nuclear safety concerns through at least one of the several means available.

One finding of very low safety significance (Green) is documented in this report. This finding involved a violation of NRC requirements. We are treating this violation as a non-cited violation (NCV) consistent with Section 2.3.2 of the Enforcement Policy.

If you contest the violation or the significance or severity of the violation documented in this inspection report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement; and the NRC Resident Inspector at Peach Bottom Atomic Power Station, Units 2 and 3.

If you disagree with a cross-cutting aspect assignment in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; and the NRC Resident Inspector at Peach Bottom Atomic Power Station, Units 2 and 3.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with Title 10 of the *Code of Federal Regulations* 2.390, "Public Inspections, Exemptions, Requests for Withholding."

U.S. NUCLEAR REGULATORY COMMISSION Inspection Report

Docket numbers: 05000277 and 05000278
License numbers: DPR-44 and DPR-56
Report numbers: 05000277/2021012 and 05000278/2021012

Enterprise Identifier: I-2021-012-0029
Licensee: Exelon Generation Company, LLC
Facility: Peach Bottom Atomic Power Station, Units 2 and 3

Location: Delta, PA 17314
Inspection dates: July 26, 2021 to August 12, 2021

Inspectors: L. Casey, Senior Project Engineer
N. Floyd, Senior Reactor Inspector
S. Rutenkroger, Senior Resident Inspector N. Warnek, Senior Allegations Coordinator

Approved by: Jonathan E. Greives, Chief Projects Branch 4 Division of Operating Reactor Safety

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting a biennial problem identification and resolution inspection at Peach Bottom Atomic Power Station, Units 2 and 3, in accordance with the Reactor Oversight Process. The Reactor Oversight Process is the NRC's program for overseeing the safe operation of commercial nuclear power reactors. Refer to https://www.nrc.gov/reactors/operating/oversight.html for more information.

List of Findings and Violations

Failure to Correct Erosion of Structural Backfill Material in the Pipe Trench			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000277,05000278/2021012-01 Open/Closed	[P.2] - Evaluation	71152B
The inspectors identified a Green finding and associated non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for Exelon's failure to adequately correct a condition adverse to quality associated with the erosion of structural backfill material in the pipe trench on the west side of the site. Specifically, continued erosion of backfill and settlement of the buried safety-related piping can lead to increased loading on the piping not intended by the design.			

Additional Tracking Items

None.

INSPECTION SCOPES

Inspections were conducted using the appropriate portions of the inspection procedures (IPs) in effect at the beginning of the inspection unless otherwise noted. Currently approved IPs with their attached revision histories are located on the public website at http://www.nrc.gov/reading-rm/doc-collections/insp-manual/inspection-procedure/index.html. Samples were declared complete when the IP requirements most appropriate to the inspection activity were met consistent with Inspection Manual Chapter (IMC) 2515, "Light-Water Reactor Inspection Program - Operations Phase." The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel to assess licensee performance and compliance with Commission rules and regulations, license conditions, site procedures, and standards. Starting on March 20, 2020, in response to the National Emergency declared by the President of the United States on the public health risks of the coronavirus (COVID-19), inspectors were directed to begin telework. In addition, regional baseline inspections were evaluated to determine if all or a portion of the objectives and requirements stated in the IP could be performed remotely. If the inspections could be performed remotely, they were conducted per the applicable IP. In some cases, portions of an IP were completed remotely and on site. The inspections documented below met the objectives and requirements for completion of the IP.

OTHER ACTIVITIES – BASELINE

71152B - Problem Identification and Resolution

Biennial Team Inspection (IP Section 02.04) (1 Sample)

(1) The inspectors performed a biennial assessment of the licensee's corrective action program, use of operating experience, self-assessments and audits, and safety conscious work environment.

- Corrective Action Program Effectiveness: The inspectors assessed the corrective action program's effectiveness in identifying, prioritizing, evaluating, and correcting problems. The inspectors also conducted a five-year review on cable degradation, piping and hanger issues, emergency diesel generators, and FLEX equipment.

- Operating Experience, Self-Assessments and Audits: The inspectors assessed the effectiveness of the station's processes for use of operating experience, audits and self-assessments.

- Safety Conscious Work Environment: The inspectors assessed the effectiveness of the station's programs to establish and maintain a safety conscious work environment.

INSPECTION RESULTS

Assessment	71152B
Problem Identification: The team determined that, in general, the licensee identified issues and entered them into the corrective action program at a low threshold.	

However, the team identified one minor performance deficiency regarding the licensee not initiating an issue report (IR) for a breach of foreign material exclusion (FME) controls when fiberglass fibers were identified on the contacts of a switch.

Problem Prioritization and Evaluation: Based on the samples reviewed, the team determined that, in general, the licensee appropriately prioritized and evaluated issues commensurate with the safety significance of the identified problem. The licensee appropriately screened IRs for operability and reportability, categorized IRs by significance, and assigned actions to the appropriate department for evaluation and resolution.

However, the team identified one Green finding with an associated non-cited violation and one minor performance deficiency. The Green finding was due to the licensee's failure to correct the erosion of structural backfill in the pipe trench due to not properly recognizing the condition when prioritizing and evaluating issues caused by the erosion. The minor performance deficiency was due to the licensee's failure to appropriately evaluate issues with the suppression chamber to reactor building pressure gauges in that sticking affected both indication and actuation signals.

Corrective Actions: The team determined that the overall corrective action program performance related to resolving problems was effective. In most cases, the licensee implemented corrective actions to resolve problems in a timely manner.

However, the team identified two minor performance deficiencies in the area of corrective actions. First, the team identified that the licensee did not establish and implement a corrective action to revise the maintenance procedure for the installation of emergency diesel generator piston rings after identifying oil scraper rings that were installed backwards. Second, the team identified that the licensee did not implement corrective actions to correct design control issues related to lead shielding packages and did not effectively accomplish corrective actions related to the control of lead shielding blankets.

Additional details on these findings and minor performance deficiencies are included later in this report.

Assessment	71152B
Use of Operating Experience: The team determined that the licensee appropriately evaluated industry operating experience for its relevance to the facility. The licensee appropriately incorporated both internal and external operating experience into plant procedures and processes, as well as lessons learned for training and pre-job briefs. Self-Assessments and Audits: The team reviewed a sample of self-assessments and audits to assess whether the licensee was identifying and addressing performance trends. The team concluded that the licensee had an effective self-assessment and audit process.	

Assessment	71152B
Safety Conscious Work Environment: The team interviewed approximately 20 individuals. The purpose of these interviews was to evaluate the willingness of the licensee staff to raise nuclear safety issues; to evaluate the perceived effectiveness of the corrective action program at resolving identified problems; and to evaluate the licensee's safety conscious work environment. The personnel interviewed were randomly selected by the inspectors from the Operations, Engineering, Maintenance, Security, and Radiation Protection work	

groups. To supplement these discussions, the team interviewed the Employee Concerns Program (ECP) representative to assess his perception of the site employees' willingness to raise nuclear safety concerns. The team also reviewed the ECP case log and select case files.

All individuals interviewed indicated that they would raise safety concerns. All individuals felt that their management was receptive to receiving safety concerns and generally addressed them promptly, commensurate with the significance of the concern. Most interviewees indicated they were adequately trained and proficient on initiating condition reports. All interviewees were aware of the licensee's ECP, stated they would use the program if necessary, and expressed confidence that their confidentiality would be maintained if they brought issues to the ECP. When asked whether there have been any instances where individuals experienced retaliation or other negative reaction for raising safety concerns, all individuals interviewed stated that they had neither experienced nor heard of an instance of retaliation at the site. The team determined that the processes in place to mitigate potential safety conscious work environment issues were adequately implemented.

Failure to Correct Erosion of Structural Backfill Material in the Pipe Trench			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000277,05000278/2021012-01 Open/Closed	[P.2] - Evaluation	71152B

The inspectors identified a Green finding and associated non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for Exelon's failure to adequately correct a condition adverse to quality associated with the erosion of structural backfill material in the pipe trench on the west side of the site. Specifically, continued erosion of backfill and settlement of the buried safety-related piping can lead to increased loading on the piping not intended by the design.

Description: During the week of May 10, 2021, while conducting visual inspections in the reactor buildings, Exelon staff found three safety-related piping hangers on the high pressure service water system for Units 2 and 3 and one piping hanger on the Unit 2 emergency service water system that were not supporting pipes as designed, as evidenced by a gap between the pipe and the support. These support issues resulted in a reasonable doubt of operability for the associated piping systems and required Exelon to perform emergent analysis. Exelon staff completed immediate corrective actions to restore the function of the supports by installing metal shims such that the bottom of the pipe was now in contact with the structural steel support. Exelon staff also evaluated the impact of the "degraded" supports and determined that the emergency service water and high pressure service water systems remained capable of performing their design safety functions.

The high pressure service water system is designed to provide a reliable supply of cooling water for the residual heat removal system under post-accident conditions (shutdown cooling and torus cooling). The emergency service water system is designed to provide a reliable supply of cooling water to the diesel generator coolers, emergency core cooling system and reactor core isolation cooling compartment air coolers, core spray pump motor oil coolers, and other selected equipment during various design basis events. The high pressure service water and emergency service water systems are designed as seismic class I, and portions of both systems are buried in a common pipe trench on the west side of the site and backfilled with select material. The structural backfill is designed to provide for drainage and support the buried components such as piping and electrical ducts.

Exelon staff performed a corrective action program evaluation under AR 04424065 and determined the cause of the pipe support gaps was due to settlement of the buried sections of pipe because of backfill soil erosion. Specifically, water is introduced and then removed from the backfill, smaller soil particles are carried away with the water resulting in a reduction in backfill volume, and then compaction/collapse of the lower density areas causes the buried pipes in the trench to settle and pivot about the wall penetration and results in an uplift for the pipes inside the building. Exelon staff identified that pipe support gaps was a historical issue occurring as far back as 1997 and referenced an equipment apparent cause evaluation under AR 00864304 from 2009 that was performed when a damaged underground electrical duct bank was discovered in the pipe trench. This evaluation documented multiple contributing causes to backfill erosion including the drainage ditch not maintained, road to reactor building transition not maintained sealed, and potential leakage of a buried pipe. Exelon determined the conclusion from this original cause evaluation remained valid.

Exelon procedure PI-AA-125, "Corrective Action Program (CAP) Procedure," Revision 7, defines a condition adverse to quality as an all-inclusive term used in reference to any of the following: failures, malfunctions, deficiencies, defective items, and non-conformances. The procedure defines a corrective action as an action taken or planned to restore a condition adverse to quality to an acceptable condition or capability.

The inspectors reviewed the cause evaluations from 2009 and 2021 to assess the effectiveness of the current and previous corrective actions for erosion of backfill. The inspectors also performed walkdowns of the roadway above the pipe trench to assess the general condition of the area. The inspectors observed conditions that contribute to water ingress into the pipe trench:

- Roadway is sunken to various depths in multiple areas and is not graded to drain water away from the plant. AR 4236628 from 2019 documented further sinking of the roadway west of the dewatering building approximately 4.5" under the steel plates. Drawing C-52, "Underground Piping Details, Revision 14, specifies a 2.0% grade from the wall of the reactor building away from the plant and states that the roadway is top finished.
- Ongoing excavation as part of the cathodic protection system modification where the length of the roadway above the pipe trench spanning between Units 2 and 3 has been excavated and filled with gravel (i.e., not sealed).
- Southern storm drain near the dewatering building has a corroded drainpipe that is missing a large section around the circumference. AR 4436966 documented an erosion cavity approximately 2 ft by 5 ft around the drain and excavated cathodic protection trench that formed after a large rainstorm. Borescope inspection of the drain below the surface identified disconnection of the vertical drain from the main drainage pipe buried in the pipe trench.

The inspectors observed that Exelon had not developed or planned corrective actions to address the continued erosion of backfill material in the pipe trench. The planned corrective action was to increase the inspection frequency of the pipe supports in the reactor building which, the inspectors concluded, would only address changes in pipe loading after it occurred and would not address the loss of structural backfill to pipe and electrical ducts underground. The inspectors noted that Exelon staff assigned several considerations and enhancements (characterized as ACITs under PI-AA-125) such as determining solutions to improve water drainage, investigating the use of soil injection, and evaluating the maximum settlement of the buried piping; however, the inspectors noted these items did not have clear, direct actions or controlled due dates to provide for planned resolution of this condition adverse to quality. The inspectors determined the failure to establish corrective actions was a performance deficiency.

Corrective Actions: Exelon staff entered the issue in their corrective action program under AR 04443058 to evaluate the overall condition of soil erosion and to assess the cumulative impact on the piping systems. Exelon staff also took immediate corrective actions to restore the function of the pipe supports. Commensurate with the very low safety-significance of the violation, NRC inspectors determined that Exelon's corrective actions to immediately restore the function of the pipe supports and maintain system operability by increasing the frequency of piping inspections is sufficient in the near term while long term corrective actions to address the soil erosion are developed under AR 04443058.

Corrective Action References: AR 04424065 and AR 04443058

Performance Assessment:

Performance Deficiency: The inspectors identified a performance deficiency for Exelon's failure to correct a condition adverse to quality associated with the erosion of structural backfill material in the pipe trench on the west side of the site

Screening: The inspectors determined the performance deficiency was more than minor because if left uncorrected, it would have the potential to lead to a more significant safety concern. Specifically, continued erosion of backfill material and settlement of the buried piping can lead to increased pipe loading not analyzed and pipe exterior corrosion protective coatings in contact with material that was not intended. The inspectors reviewed IMC 0612, Appendix E, and did not find any examples that were applicable to this performance deficiency.

Significance: The inspectors assessed the significance of the finding using Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." The inspectors determined the finding was of very low safety significance (Green) because the impacted piping and connected components have been currently demonstrated to maintain their operability and/or PRA functionality.

Cross-Cutting Aspect: P.2 - Evaluation: The organization thoroughly evaluates issues to ensure that resolutions address causes and extent of conditions commensurate with their safety significance. Exelon staff did not develop resolutions to address the causes of continued backfill erosion in the pipe trench, which contains buried safety-related components for both Units 2 and 3.

Enforcement:

Violation: 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures shall be established to assure that conditions adverse to quality, such as deficiencies, defective material, and non-conformances are promptly identified and corrected. PI-AA-125, "Corrective Action Program (CAP) Procedure," Revision 7, step 4.5.2 states, in part, to create a corrective action for any planned actions necessary to restore a condition adverse to quality.

Contrary to the above, from at least May 10, 2021 to present, Exelon staff did not establish measures to correct the condition adverse to quality associated with the erosion of structural backfill material in the pipe trench.

Enforcement Action: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.

Minor Performance Deficiency	71152B

Minor Performance Deficiency: The team identified a minor performance deficiency due to the licensee's inadequate corrective actions regarding the control of design modifications for permanent lead shielding and the control of lead shielding blankets in the facility. In 2015 Exelon initiated IR 2490285 which identified that the governing procedure for lead shielding, RP-PB-552, "Shielding Program," contained a category for non-permanent long term shielding exceeding one year which was contrary to the design specification NE-00048, "Use of Temporary Shielding." Shielding in place for greater than one year was required to be controlled in accordance with procedure CC-AA-103, "Configuration Change Control for Permanent Physical Plant Changes," which was not performed. Exelon revised RP-PB-552 and performed a 10 CFR 50.59 review in accordance with LS-AA-104 to accept the shielding on an interim basis. The associated evaluation credited previous annual inspections and long-term temporary shielding requirements but was not an installation performed in accordance with CC-AA-103.

In 2019 Exelon initiated IR 4265109 which identified that lead shielding blankets were draped over an existing lead curtain without being properly secured. The shielding was not installed in accordance with RP-PB-552 and NE-00048. Exelon removed the blankets improperly placed and created an action to perform lead shielding inspections more frequently on a quarterly basis rather than annual.

The team determined that Exelon personnel had stopped performing quarterly inspections in 2021 based on no additional issues being identified, but no approval was sought or documented for this change to the action. The team also performed a walkdown and identified lead shielding blankets laying on Unit 3 'D' core spray pump torus suction line piping without any knowledge or approval by Radiation Protection or Engineering staff or associated reviews and approvals. The team also identified multiple shielding packages with scaffold frameworks in close proximity to safety-related piping, including a shielding package in which supporting chains were routed in contact and across piping and piping insulation such that the chain pushed laterally against the piping. NE-00048 specifies for temporary shielding that all scaffold framework be installed with at least 12 inches clearance to nuclear safety-related equipment unless evaluated on a case-by-case basis by engineering. Finally, the team identified that the long term temporary shielding identified in 2015 was still categorized and tracked as such, and the required permanent design change process in accordance with CC-AA-103 was not performed.

The team determined that the shielding blankets placed on the Unit 3 'D' core spray piping was a performance deficiency since RP-PB-552 requires review and approval by Radiological Engineering which was not performed. The team determined that the long term temporary shielding packages installed for more than a year were a performance deficiency since the packages were required by design specification NE-00048 (and subsequently a revised RP-PB-552) to be installed in accordance with CC-AA-103 as permanent design change to the facility which was not performed.

Exelon initiated IR 4437733 to re-review the action to perform quarterly inspections and conduct a department clock reset. Exelon initiated IR 4440364 for the shielding blankets found on the piping and removed the blankets promptly. Exelon initiated IR 4440428 for the shielding package with chains in contact with piping and/or piping insulation and generated an action to review the package and recommend adjustments as required. Exelon initiated IR 4440269 for the long-term temporary shielding not installed per CC-AA-103 with recommended actions to verify all shielding meets RP-PB-552 installation criteria and develop engineering change packages per CC-AA-103. Lastly, Exelon reviewed the installed long-term shielding packages and determined that, while no permanent design change package was performed, they had been evaluated by Engineering for acceptability under 10 CFR 50.59, "Changes, tests, and experiments."

Screening: The inspectors determined the performance deficiency was minor. Specifically, the team concluded that for the specific lead shielding packages the team observed the weights, configurations, and good material conditions did not represent sufficient doubt of operability, or actual inoperability, to classify as more than minor.

Minor Performance Deficiency	71152B

Minor Performance Deficiency: The team identified a minor performance deficiency due to the licensee's inadequate corrective actions stemming from an adverse trend review that did not adequately consider design attributes and surveillance results. Specifically, the team identified an evaluation that was inaccurate for DPIS-3503A and DPIS-3503B, "Suppression Chamber – Reactor Building Pressure Instruments." The instruments provide open logic to the torus to reactor building vacuum breakers and annunciation in the control room. The instruments have a local gauge with a needle displaying vacuum and output signals with an alarm setpoint at 10.5 inches water vacuum and a vacuum breaker actuation setpoint at 13.8 inches water vacuum.

During normal operation the primary containment is maintained at a positive pressure relative to the reactor building. Due to the instruments' design, the indicators normally display below zero with the local needle gauge approaching or on the backstop. However, the instruments were not designed to remain routinely under-ranged, i.e. below zero, which places the linkages that move the cams under greater stress. The Unit 3 instruments, DPIS-3503A and DPIS-3503B, were found to be out of tolerance in 2005, 2011, 2014, 2016, 2019, and 2020. DPIS-3503A was found to actuate the vacuum breaker at a value less than the specified range on two occasions. DPIS-3503B was found initially stuck in the under-range position each surveillance, until releasing when applied pressure increased sufficiently: 7.5" in 2005, value not recorded in 2011, 8" in 2014, 8" in 2016, and 15" in 2019.

In 2014 Exelon initiated IR 2381544 and performed an evaluation of the instruments due to the surveillance results. The evaluation concluded that no adverse trend existed based on no historical issues being identified and stating that the pointer was for local indication only. When the sticking recurred in 2016 and 2019, Exelon initiated IR 2637387 and IR 4237955, respectively, and referenced the evaluation each time to conclude no additional actions were required. However, the team determined that the sticking condition was an historical issue that prevented the instrument from responding properly until sufficient pressure was applied. In particular, the condition was not solely a local pointer indication problem. Therefore, the adverse trend evaluation was inadequate since it did not properly consider the impact and potential for future degradation. In particular, the sticking in 2019 impacted the capability of the instrument to actuate the alarm and open the vacuum breaker. The instrument did not release until reaching 15" which exceeded the alarm acceptable range of 9.6" to 11.4". In

addition, this sticking impacted the as found range for the actuation of the opening of the vacuum breaker with an acceptable range of 12.4" to 15.2". However, the as found trip actuation was improperly recorded as 14.6" even though the instrument did not initially respond until reaching 15". As a result, Exelon did not establish and prioritize actions to correct the underlying condition adverse to quality in accordance with PI-AA-120, "Issue Identification and Screening Process," i.e. the instruments being routinely under-ranged during normal operation. However, notwithstanding the conclusion of the evaluation and IRs, Exelon created a work request from which a work order later replaced DPIS-3503B in 2019. By replacing the instrument, accumulated degradation was reset. However, the new instrument was also under-ranged with the indicating needle on the backstop, which the team observed during the inspection. Therefore, the underlying issue with the instrument design not being appropriate to the conditions was not corrected. The team also shared the observation that PI-AA-120 states that multiple examples of similar safety-related equipment problems where the equipment is operable, but has degraded, reflects a potential common failure mechanism and is a significance level 3 issue, and the previous IRs were coded level 4. Finally, the team noted that procedure CC-AA-309-101, "Engineering Technical Evaluations," provides the expectations for technical content and rigor for technical evaluations performed by Engineering, including input on conditions that are outside of expected ranges, or otherwise degraded, and requires that technical evaluations be complete, accurate, and technically adequate. In response to the PIR team observation, Exelon initiated IR 4439031 that documented the observation, created an action to consider revising the 2014 evaluation, and initiated an engineering change request to perform a design change to replace the instruments with a design appropriate to the conditions. Screening: The inspectors determined the performance deficiency was minor. Although the DPIS-3503B instrument exceeded the acceptable range for alarm actuation in 2019, and the as found breaker actuation setpoint was affected and an improper value was recorded, the instrument released at 15" which remained below the acceptable range for breaker actuation. Also, the team reviewed operator actions for the alarm and determined there would be no meaningful impact to nuclear safety had vacuum exceeded the alarm acceptable range but remained below the breaker actuation acceptable range. Finally, notwithstanding the use and re-use of the evaluation to conclude no action was required, Exelon later replaced the instrument.

Minor Performance Deficiency	71152B
Minor Performance Deficiency: The inspectors identified a minor performance deficiency related to Exelon's failure to implement a self-assigned corrective action. This matter was identified during the inspectors' review of the corrective action program evaluation (CAPE) performed under AR 04361516, which documented excessive lube oil usage during the E2 emergency diesel generator run in August 2020. The CAPE assigned one corrective action: to perform an E2 emergency diesel generator overhaul during the next available system outage window. Upon completion of the overhaul, in January 2021, the CAPE was revised to capture seven potential causes of the high lube oil consumption. One such cause was that two of the lower piston scraper rings were installed upside down, causing the rings to scrape oil toward the combustion chamber rather than away from it.	

The revised CAPE included a new corrective action, listed as Action 38, to revise the diesel maintenance procedure M-052-011 with current Fairbanks Morse guidance for inspection and installation of the upper and lower piston oil scraper rings. Exelon assigned a due date of June 11, 2021, for this corrective action. However, the inspectors identified that the corrective action was never generated in Exelon's action tracking system (PassPort), and was therefore not implemented by the due date.

The inspectors raised this matter to Exelon, who identified that there was an open action, under a related condition report, that was similar to the missed corrective action. Specifically, AR 04398998 had been written following the January 2021 E2 emergency diesel generator overhaul to document the cylinder liners/piston inspection results. Under this AR the station had created a procedure change request action (PCRA) to revise M-052-011 to add steps to ensure that rings are installed and verified in the correct orientation. The inspectors noted that, per Exelon procedures, PCRAs are not to be used for corrective actions; they can be extended with supervisor approval, whereas corrective actions require approval by the Management Review Committee (MRC). The original due date for the PCRA was March 25, 2021, but the due date was subsequently extended multiple times, most recently to August 13, 2021. As a result of the multiple extensions, the procedure was not revised in time to meet the 04361516-38 corrective action due date of June 11, 2021, and the station also missed the opportunity to revise the procedure prior to the next diesel overhaul, which was the E3 emergency diesel generator in June 2021.

The inspectors determined that Exelon had failed to implement corrective action 04361516-38 by the MRC-approved due date, and did not seek out an extension as required by their procedures. Specifically, Exelon procedure PI-AA-125, "Corrective Action Program (CAP) Procedure," Revision 7, provides that due dates for corrective actions can be extended, but that such "extensions for… CAs … shall be approved by the MRC or MRC Chairman (Plant Manager)." The inspectors determined that Exelon's failure to receive MRC approval to exceed a corrective action due date constituted a performance deficiency that was reasonably within their ability to foresee and correct and should have been prevented.

Exelon entered this performance deficiency into their corrective action program under AR 04439261. Corrective actions included locking in the due date for the PCRA, creating a new corrective action under AR 04361516 to track the PCRA completion, and revising the CAPE with the updated actions.

Screening: The inspectors determined the performance deficiency was minor. Specifically, even though the procedure was not updated prior to the next diesel overhaul, the inspectors determined through interviews that the piston scraper rings were most likely installed correctly and even if the piston scraper rings were installed incorrectly they would not, on their own, impact functionality of the diesel.

Minor Performance Deficiency	71152B

Minor Performance Deficiency: The team identified a minor performance deficiency due to the licensee not initiating an IR for a loss of FME integrity. In April of 2020, the 'B' emergency cooling tower fan tripped. Exelon initiated IR 4333925 and determined the cause of the trip was high vibration. However, the equipment operator in the field did not note any unusual noise or vibration. Exelon replaced the vibration microswitch, sent the failed switch to a laboratory for special testing, and performed a work group evaluation. The laboratory determined that the switch reset exhibited high resistance, and/or failed to actuate, and therefore failed contact continuity acceptance criteria in seven of the first ten tests. Then, the switch and contacts passed all testing acceptance criteria in the next ten documented tests, with further good performance during subsequent testing. The lab then examined the contacts, and although they found them mostly free of corrosion, they identified fiberglass-like particles on the contacts. The lab noted that such particles having dielectric properties could interrupt contact continuity. The lab report also noted that the special testing performed, in contrast to failure analysis, created a potential to dislodge and lose interfering material, if it was previously present. Exelon documented this information in the work group evaluation.

The team noted procedural requirements were not met regarding identification of foreign material (FM) within systems and components. Procedure MA-AA-716-008, "Foreign Material Exclusion Program," states that any material not part of the system or component as designed or modified, including unexpected dirt and debris is foreign material (FM) and to ensure an IR is initiated for a loss of FME integrity. It also defines a loss of FME integrity to include unexpected FM found in systems or components. Procedure MA-AA-716-008-1000, "Definitions and Measurements of FME Events," states that FM control practices that has resulted in a limiting condition for operation being entered is a significant event. It also states that FME discovered in a system or component due to failed preventive maintenance program, equipment design shortfall, or environmental conditions is equipment degradation. It also states that all IRs written to address FME should include FME in the title and requires tracking and trending FME issues. PI-AA-120, "Issue Identification and Screening Process," classifies an FME event with the potential to inhibit or has inhibited a safety-related function of a structure, system, or component as a significance level 3 IR.

Although the particles were small, Exelon did not provide information to the team to show that they were originally part of the system or component. The team therefore determined the material was akin to dirt and debris and not expected to be found in the component. The team further noted that the lab report and Exelon's work group evaluation documented that the particles were possibly interrupting the continuity of the contacts. The team shared an observation that initiating a specific IR for FM provides a means to track and trend FME issues. In addition, properly documenting such issues provides early indication of developing problems which can be investigated and resolved prior to becoming a more significant safety concern. Finally, an IR improves traceability of issues should FM be transported which allows the licensee to identify and correct the underlying cause of problems.

In response to the PIR team observation, Exelon initiated IR 4438511 to document that fiberglass fibers were identified on the contacts of the switch and that an evaluation of the PowerLabs results was not documented.

Screening: The inspectors determined the performance deficiency was minor. Specifically, Exelon replaced the microswitch, and the team did not conclude that an FME IR would reasonably have prevented an additional equipment failure in this situation.

EXIT MEETINGS AND DEBRIEFS

The inspectors verified no proprietary information was retained or documented in this report.

- On August 12, 2021, the inspectors presented the biennial problem identification and resolution inspection results to Mr. David Henry and other members of the licensee staff.

DOCUMENTS REVIEWED

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
71152B	Corrective Action Documents	AR 04252679	E1 EDG Cable Fault (NCV, Root Cause)	
		AR 04321794	E1 EDG Shutdown due to Intercooler Coolant Low Pressure (NCV, CAPE, LER)	
		AR 04359332	Station Battery Service Life (NCV, CAPE)	
		IR 00314158		
		IR 00493902		
		IR 02381544		
		IR 02490285		
		IR 02637387		
		IR 04217383		
		IR 04219191		
		IR 04219461		
		IR 04237955		
		IR 04254133		
		IR 04265109		
		IR 04268480		
		IR 04289548		
		IR 04291877		
		IR 04295183		
		IR 04311842		
		IR 04317675		
		IR 04333925		
		IR 04339030		
		IR 04345189		
		IR 04352704		
		IR 04358626		
		IR 04358666		
		IR 04379330		
		IR 04380273		
		IR 04381492		
		IR 04387917		

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		IR 04390615		
		IR 04395311		
		IR 04397083		
		IR 04403652		
		IR 04408187		
		IR 04408199		
		IR 04416673		
		IR 04422713		
		IR 04422952		
		IR 04422965		
		IR 04424708		
		IR 04425304		
		IR 04429399		
		IR 04429484		
	Corrective Action Documents Resulting from Inspection	IR 04437488		
		IR 04437490		
		IR 04437554		
		IR 04437733		
		IR 04437986		
		IR 04438511		
		IR 04439031		
		IR 04440269		
		IR 04440364		
		IR 04440428		
	Drawings	C-51	Underground Piping South Area	Revision 37
	Miscellaneous	C-16	Specification for Installation of Underground Piping	Revision 1
		C-32	Specification for Structural Backfilling	Revision 1
		NE-00048	Use of Temporary Shielding	Revision 3
	Procedures	CC-AA-103	Configuration Change Control for Permanent Physical Plant Changes	Revision 33
		CC-AA-309-101	Engineering Technical Evaluations	Revision 16
		LS-AA-104	Exelon 50.59 Review Process	Revision 12
		MA-AA-716-008	Foreign Material Exclusion Program	Revision 16

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		MA-AA-716-008-1000	Definitions and Measurements of FME Events	Revision 6
		PI-AA-120	Issue Identification and Screening Process	Revision 11
		PI-AA-125	Corrective Action Program (CAP) Procedure	Revision 7
		PI-AA-125-1003	Corrective Action Program Evaluation Manual	Revision 6
		RP-PB-552	Shielding Program	Revision 6
	Work Orders	WO 04705293		
		WO 04893232		
		WO 04976510		
		WO R1250450		
		WO R1299879		
		WR 01495053		

October 13, 2021 – Letter from Caroline L. Carusone, Deputy Director Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to David P. Rhoades Senior Vice President Exelon Generation Company, LLC President and Chief Nuclear Officer Exelon Nuclear with subject of PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 – EXEMPTION FROM SPECIFIC REQUIREMENTS OF 10 CFR PART 26 (EPID L-2021-LLE- 0041 [COVID-19])

The U.S. Nuclear Regulatory Commission (NRC) has approved the requested temporary exemption from specific requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 26, "Fitness for Duty Programs," Section 26.205, "Work hours," for Peach Bottom Atomic Power Station (Peach Bottom), Units 2 and 3. This action is in response to Exelon Generation Company, LLC (Exelon, the licensee) application dated September 17, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession
No. ML21260A162). This application cited the March 28, 2020 (ADAMS Accession No. ML20087P237) and November 10, 2020 (ADAMS Accession No. ML20261H515), letters from Mr. Ho Nieh describing a process to request expedited review of certain exemptions from 10 CFR Part 26 during the Coronavirus Disease 2019 (COVID-19) public health emergency (PHE).

The application provided the following information:

- • A statement that explains how, and for which covered groups at Peach Bottom, the COVID-19 PHE impacts the licensee's ability to meet the work-hour control requirements of 10 CFR 26.205(d)(1)-(d)(7);
- • A statement that describes how the licensee would use an exemption from the 10 CFR 26.205(d)(1)-(d)(7) work-hour control requirements to manage the impact of the COVID-19 PHE on maintaining plant operational safety and security at Peach Bottom;
- • A list of positions for which the licensee may implement alternative work-hour controls at Peach Bottom upon the NRC granting the requested exemption;
- • The date and time when the licensee will begin implementing site-specific COVID-19 PHE fatigue-management controls at Peach Bottom for personnel specified in
  10 CFR 26.4(a);
- • A statement that the licensee's site-specific COVID-19 fatigue-management controls at Peach Bottom are consistent with the constraints outlined in the November 10, 2020, letter; and
- • A statement that the licensee will establish alternative controls at Peach Bottom for the management of fatigue during the period of the exemption and that, at a minimum, the controls ensure the following for individuals subject to these alternative controls:

  ○ Individuals will not work more than 16 work hours in any 24-hour period and not more than 86 work hours in any 7-day period, excluding shift turnover;

  ○ A minimum 10-hour break is provided between successive work periods;

  ○ 12-hour shifts are limited to not more than 14 consecutive days;

  ○ A minimum of 6 days off is provided in any 30-day period; and

  ○ Requirements have been established for behavioral observation and self-declaration during the period of the exemption.

Therefore, the NRC finds that the technical basis for an exemption, described in the March 28, 2020, letter from Mr. Ho Nieh, is applicable to the licensee's application.

Furthermore, although not explicitly stated as part of the licensee's application, the November 10, 2020, letter also states that the controls should ensure that the calculation of work hours and days off includes all work hours and days off during the applicable calculation periods, including those work hours and days off preceding initiation of the exemption period.

The NRC previously approved an exemption request for Peach Bottom, with the preceding exemption period ending on December 11, 2020 (ADAMS Accession No. ML20247J620).

> Because the requested exemption will not begin within 14 days of the end of the most recent exemption for Peach Bottom, the licensee has had sufficient time to manage the potential for cumulative fatigue by implementing the standard work-hour controls administered in accordance with 10 CFR Part 26. Therefore, the NRC did not request that Exelon provide additional information regarding the technical basis for its request, as discussed in the November 10, 2020, letter.

Section 26.9, "Specific exemptions," of 10 CFR allows the NRC to grant exemptions from the requirements of 10 CFR Part 26, as it determines are authorized by law, will not endanger life or property or the common defense and security, and are otherwise in the public interest.

The NRC determined that the requested exemption is permissible under the Atomic Energy Act of 1954, as amended, and other regulatory requirements. Therefore, the NRC finds that the requested exemption is authorized by law.

The underlying purpose of 10 CFR 26.205(d) is to prevent impairment from fatigue due to duration, frequency, or sequencing of successive shifts. Based on the evaluation provided in the NRC's March 28 and November 10, 2020, letters, along with the criteria discussed above, no new accident precursors are created by using whatever licensee staff resources may be necessary or available during the term of this exemption to respond to a plant emergency and to ensure that the plant maintains a safe and secure status. Therefore, the probability of postulated accidents is not increased. Also, the consequences of postulated accidents are not increased because there is no change in the types of accidents previously evaluated. The requested exemption would allow the use of licensee staff resources as may be necessary to maintain safe operation of the plant and to respond to a plant emergency. Therefore, the NRC finds that the requested exemption will not endanger life or property.

The requested exemption would allow the use of licensee security staff resources as may be necessary to ensure the common defense and security. Therefore, the NRC finds that the requested exemption will not endanger the common defense and security.

Due to the impacts that the COVID-19 PHE has had on the licensee's ability to comply with the work-hour controls of 10 CFR 26.205(d), the importance of maintaining the operations of Peach Bottom, and the controls the licensee has established, the NRC finds that the requested exemption is in the public interest.

Granting the requested exemption from the requirements of 10 CFR 26.205 is categorically excluded under 10 CFR 51.22(c)(25), and there are no extraordinary circumstances present that would preclude reliance on this exclusion. The NRC staff determined, per
10 CFR 51.22(c)(25)(vi)(I), that the requirements from which the exemption is sought involve other requirements of an administrative, managerial, or organizational nature.

The NRC staff also determined that approval of this exemption involves no significant hazards consideration because it does not authorize any physical changes to the facility or any of its safety systems, does not authorize changes to any of the assumptions or limits used in the licensee's safety analyses, and does not introduce any new failure modes. There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite because this exemption does not affect any effluent release limits as provided in the licensee's technical specifications or by the regulations in 10 CFR Part 20, "Standards for Protection Against Radiation." There is no significant increase in individual or cumulative public or occupational radiation exposure because this exemption does not affect the limits on the release of any radioactive material or the limits provided in 10 CFR Part 20 for radiation exposure to workers or members of the public. There is no significant construction impact because this exemption does not involve any changes to a construction permit. There is no significant increase in the potential for or consequences from radiological accidents because the exemption does not alter any of the assumptions or limits in the licensee's safety analysis. In addition, the NRC staff determined that there would be no significant impacts to biota, water resources, historic properties, cultural resources, or socioeconomic conditions in the region. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the approval of this exemption request.

Based on the above, the NRC finds that: (1) the exemption is authorized by law, (2) the exemption will not endanger life or property or the common defense and security, and (3) the exemption is otherwise in the public interest.

This exemption is effective starting October 18, 2021, for a period of 60 days.

November 9, 2021 – Letter from Jonathan E. Greives, Chief Projects Branch 4 Division of Operating Reactor Safety to David P. Rhoades Senior Vice President Exelon Generation Company, LLC President and Chief Nuclear Officer Exelon Nuclear with subject of PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 – INTEGRATED INSPECTION REPORT 05000277/2021003 AND 05000278/2021003

On September 30, 2021, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Peach Bottom Atomic Power Station, Units 2 and 3. On October 15, 2021, the NRC inspectors discussed the results of this inspection with Mr. Matthew Herr, Site Vice President, and other members of your staff. The results of this inspection are documented in the enclosed report.

One finding of very low safety significance (Green) is documented in this report. This finding involved a violation of NRC requirements. We are treating this violation as a non-cited violation (NCV) consistent with Section 2.3.2 of the Enforcement Policy.

If you contest the violation or the significance or severity of the violation documented in this inspection report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement; and the NRC Resident Inspector at Peach Bottom Atomic Power Station, Units 2 and 3.

If you disagree with a cross-cutting aspect assignment in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; and the NRC Resident Inspector at Peach Bottom Atomic Power Station, Units 2 and 3.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with Title 10 of the *Code of Federal Regulations* 2.390, "Public Inspections, Exemptions, Requests for Withholding."

U.S. NUCLEAR REGULATORY COMMISSION
Inspection Report

Docket Numbers:	05000277 and 05000278
License Numbers:	DPR-44 and DPR-56
Report Numbers:	05000277/2021003 and 05000278/2021003
Enterprise Identifier:	I-2021-003-0023
Licensee:	Exelon Generation Company, LLC
Facility:	Peach Bottom Atomic Power Station, Units 2 and 3
Location:	Delta, PA 17314
Inspection Dates:	July 1, 2021 to September 30, 2021
Inspectors:	S. Rutenkroger, Senior Resident Inspector P. Boguszewski, Resident Inspector E. Andrews, Health Physicist T. Corcoran, Project Engineer J. DeBoer, Senior Emergency Preparedness Inspector N. Mentzer, Reactor Inspector C. Swisher, Project Engineer A. Turilin, Reactor Inspector
Approved By:	Jonathan E. Greives, Chief Projects Branch 4 Division of Operating Reactor Safety

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting an integrated inspection at Peach Bottom Atomic Power Station, Units 2 and 3, in accordance with the Reactor Oversight Process. The Reactor Oversight Process is the NRC's program for overseeing the safe operation of commercial nuclear power reactors. Refer to https://www.nrc.gov/reactors/operating/oversight.html for more information.

List of Findings and Violations

Safety Relief Valve Inoperability Due to Nitrogen Leakage from Braided Hose Wear			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000277/2021003-01 Open/Closed	[H.7] - Documentation	71153
The inspectors identified a self-revealing Green finding and associated non-cited violation (NCV) of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." Specifically, stainless steel (SS) braided flexible hoses associated with safety relief valve (SRV) '71K' failed due to a dislodged support clip that was not installed properly because the work instructions lacked sufficient detail and therefore were not appropriate to the circumstances.			

Additional Tracking Items

Type	Issue Number	Title	Report Section	Status
LER	05000277/2-2021-02	LER 2-2021-002-00 for Peach Bottom Atomic Power Station, Unit 2, SRV Inoperability Due to Nitrogen Leakage from Braided Hose Wear	71153	Closed

PLANT STATUS

Unit 2 began the inspection period at rated thermal power (RTP). On September 17, 2021, the unit was down powered to 70 percent for a planned control rod sequence exchange and friction testing, turbine valve testing, and waterbox inlet valve cycling. The unit was returned to RTP the following day. The unit remained at or near RTP for the remainder of the inspection period.

Unit 3 began the inspection period at RTP. On July 9, 2021, the unit was down powered to 65 percent for a planned control rod pattern adjustment for all rods out end of cycle operation and was returned to RTP the following day. Other than end of cycle coast down, the unit remained at or near RTP for the remainder of the inspection period.

INSPECTION SCOPES

Inspections were conducted using the appropriate portions of the inspection procedures (IPs) in effect at the beginning of the inspection unless otherwise noted. Currently approved IPs with their attached revision histories are located on the public website at http://www.nrc.gov/reading-rm/doc-collections/insp-manual/inspection-procedure/index.html. Samples were declared complete when the IP requirements most appropriate to the inspection activity were met consistent with Inspection Manual Chapter (IMC) 2515, "Light-Water Reactor Inspection Program - Operations Phase." The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel to assess licensee performance and compliance with Commission rules and regulations, license conditions, site procedures, and standards. Starting on March 20, 2020, in response to the National Emergency declared by the President of the United States on the public health risks of the coronavirus (COVID-19), resident and regional inspectors were directed to begin telework and to remotely access licensee information using available technology. During this time, the resident inspectors performed periodic site visits each week, increasing the amount of time on site as local COVID-19 conditions permitted. As part of their onsite activities, resident inspectors conducted plant status activities as described in IMC 2515, Appendix D; observed risk significant activities; and completed on site portions of IPs. In addition, resident and regional baseline inspections were evaluated to determine if all or a portion of the objectives and requirements stated in the IP could be performed remotely. If the inspections could be performed remotely, they were conducted per the applicable IP. In some cases, portions of an IP were completed remotely and on site. The inspections documented below met the objectives and requirements for completion of the IP.

REACTOR SAFETY

71111.04 - Equipment Alignment

Partial Walkdown Sample (IP Section 03.01) (4 Samples)

The inspectors evaluated system configurations during partial walkdowns of the following systems/trains:

(1) Unit 3 'A' and 'C' division of high-pressure service water (HPSW) during planned work on 'B' and 'D' HPSW on August 24, 2021
(2) Unit 3 'C' residual heat removal (RHR) system during planned work on 'B' and 'D' HPSW on August 31, 2021
(3) Unit 3 'B' and 'D' division of HPSW after the completion of planned work on 'B' and 'D' HPSW on September 2, 2021

(4) Unit common 'E-1' emergency diesel generator (EDG) during 'E-3' EDG maintenance on September 30, 2021

71111.05 - Fire Protection

Fire Area Walkdown and Inspection Sample (IP Section 03.01) (5 Samples)

The inspectors evaluated the implementation of the fire protection program by conducting a walkdown and performing a review to verify program compliance, equipment functionality, material condition, and operational readiness of the following fire areas:

(1) PF-132 and PF-132A, Unit common diesel generator building, general area, and upper level on July 8, 2021
(2) PF-144, Units 2 and 3 circulating water pump structure, general area on August 10, 2021
(3) PF-117 and PF-127, Units 2 and 3 turbine building emergency battery switchgear rooms on August 16, 2021
(4) PF-12B, Unit 3 reactor building closed loop cooling water room on August 31, 2021
(5) PF-5A, Unit 2 'A' and 'C' core spray rooms on September 24, 2021

71111.06 - Flood Protection Measures

Cable Degradation (IP Section 03.02) (1 Sample)

The inspectors evaluated cable submergence protection in:

(1) Manholes (MH) MH-065, MH-25B, and MH-26C on August 20, 2021

71111.07A - Heat Sink Performance

Annual Review (IP Section 03.01) (1 Sample)

The inspectors evaluated readiness and performance of:

(1) Unit common, 'E-4' EDG air coolant heat exchanger annual summer cleaning on July 21, 2021

71111.11Q - Licensed Operator Requalification Program and Licensed Operator Performance

Licensed Operator Performance in the Actual Plant/Main Control Room (IP Section 03.01) (1 Sample)

(1) The inspectors observed and evaluated licensed operator performance in the control room during a scheduled downpower and subsequent power ascension of Unit 3 for removal of feedwater heaters from service for end of cycle coastdown on August 11, 2021

Licensed Operator Requalification Training/Examinations (IP Section 03.02) (1 Sample)

(1) The inspectors observed and evaluated licensed operator requalification training in the simulator on September 27, 2021

71111.12 - Maintenance Effectiveness

Maintenance Effectiveness (IP Section 03.01) (1 Sample)

The inspectors evaluated the effectiveness of maintenance to ensure the following structures, systems, and components (SSCs) remain capable of performing their intended function:

(1) Unit 3 HPSW on September 9, 2021

71111.13 - Maintenance Risk Assessments and Emergent Work Control

Risk Assessment and Management Sample (IP Section 03.01) (5 Samples)

The inspectors evaluated the accuracy and completeness of risk assessments for the following planned and emergent work activities to ensure configuration changes and appropriate work controls were addressed:

(1) Unit 3 'A' RHR planned testing on July 7, 2021
(2) Unit 3 'B' and 'D' HPSW planned modification outage on August 24, 2021
(3) Unit 3 'B' and 'D' HPSW risk informed completion time for planned maintenance to upgrade the 'B' and 'D' division of HPSW on August 26, 2021
(4) Unit common, 'E-3' EDG planned maintenance with a focus on Unit 2 action green risk and systems on September 27, 2021
(5) Unit common, 'E-3' EDG planned maintenance with a focus on Unit 3 action green risk and systems on September 30, 2021

71111.15 - Operability Determinations and Functionality Assessments

Operability Determination or Functionality Assessment (IP Section 03.01) (4 Samples)

The inspectors evaluated the licensee's justifications and actions associated with the following operability determinations and functionality assessments:

(1) Unit common, EDG 'E-3' unsatisfactory backlash readings on governor drive gears on June 23, 2021
(2) Notification from transmission system operator that inadequate transmission facility trip contingency voltage is predicted on July 7, 2021
(3) Unit 3 HPSW pipe hanger found with a gap between the support and the pipe on August 25, 2021
(4) Unit common, standby gas treatment exhaust fan '0CV020' backdraft damper was found open on August 31, 2021

71111.18 - Plant Modifications

Temporary Modifications and/or Permanent Modifications (IP Section 03.01 and/or 03.02) (1 Sample)

The inspectors evaluated the following temporary or permanent modification:

(1) Unit 3 temporary configuration change for the reactor water clean-up pump trip logic on closure of the suction valve on August 26, 2021

71111.19 - Post-Maintenance Testing

Post-Maintenance Test Sample (IP Section 03.01) (3 Samples)

The inspectors evaluated the following post-maintenance test activities to verify system operability and functionality:

(1) Unit 2 ‘A’ battery charger '2AD003-1' after internal components replacement on July 27, 2021
(2) Unit common diesel driven fire pump following planned vendor overhaul on August 31, 2021
(3) Unit 3 HPSW pumps 'B' and 'D' after restoring the system from planned maintenance on September 1, 2021

71111.22 - Surveillance Testing

The inspectors evaluated the following surveillance tests:

Surveillance Tests (other) (IP Section 03.01) (3 Samples)

(1) Unit 3 RHR loop ‘A’ logic system functional test performed on July 7, 2021
(2) Unit common, 'E-3' EDG monthly surveillance performed on August 3, 2021
(3) Unit 2 response time test of the main steam isolation valve closure on scram signal with ‘A’ channel of the reactor protection system on August 17, 2021

RCS Leakage Detection Testing (IP Section 03.01) (1 Sample)

(1) Unit 3 monitored for increased drywell unidentified leakage on August 9, 2021

FLEX Testing (IP Section 03.02) (1 Sample)

(1) Unit common FLEX EDG 3-year preventive maintenance full load runs on August 4, 2021

71114.06 - Drill Evaluation

Select Emergency Preparedness Drills and/or Training for Observation (IP Section 03.01) (1 Sample)

(1) The inspectors observed a focused area emergency preparedness drill conducted on July 27, 2021

Drill/Training Evolution Observation (IP Section 03.02) (1 Sample)

The inspectors evaluated:

(1) The inspectors observed a full scope emergency preparedness drill conducted on August 3, 2021

RADIATION SAFETY

71124.05 - Radiation Monitoring Instrumentation

Walkdowns and Observations (IP Section 03.01) (8 Samples)

The inspectors evaluated the following radiation detection instrumentation during plant walkdowns:

(1) Area radiation monitors throughout Unit 2 and Unit 3
(2) Argos-5 personal contamination monitors
(3) Continuous air monitors throughout Unit 2 and Unit 3
(4) Drywell high-range radiation monitors
(5) PM-12 gamma portal monitors
(6) Portable instruments in a "ready for use" state
(7) Small article monitors
(8) Whole body counters

Calibration and Testing Program (IP Section 03.02) (13 Samples)

The inspectors evaluated the calibration and testing of the following radiation detection instruments:

(1) Eberline AMS-4, SN 334568
(2) Eberline RO20AA, SN 11963
(3) Eberline RO20AA, SN 12594
(4) Eberline RO-20, SN 005178
(5) Ludlum-3, SN 333597
(6) Ludlum-3, SN 320418
(7) Ludlum-3, SN 269079
(8) Ludlum 3030P, SN 264772
(9) Ludlum-177, SN 273161
(10) Ludlum-177, SN 319270
(11) Ludlum-177, SN 319273
(12) MPG BAK-2270, SN 6612-111
(13) MPG BAK-2270, SN 6618-069

Effluent Monitoring Calibration and Testing Program Sample (IP Sample 03.03) (2 Samples)

The inspectors evaluated the calibration and maintenance of the following radioactive effluent monitoring and measurement instrumentation:

(1) Unit 2 reactor building vent exhaust radiation monitors, RIS-2-17-425A, RIS-2-17-425B, RIS-2-17-425C, and RIS-2-17-425D

(2) Unit 3 refuel floor vent exhaust radiation monitors, RIS-3-17-458A, RIS-3-17-458B, RIS-3-17-458C, and RIS-3-17-458D

71124.08 - Radioactive Solid Waste Processing & Radioactive Material Handling, Storage, & Transportation

Radioactive Material Storage (IP Section 03.01) (2 Samples)

(1) Inspectors evaluated the licensee's performance in controlling, labeling and securing radioactive materials in the Low Level Radioactive Waste Storage Facility
(2) Inspectors evaluated the licensee's performance in controlling, labeling and securing radioactive materials in the Radiological Controlled Area South Yard

Radioactive Waste System Walkdown (IP Section 03.02) (2 Samples)

(1) Inspectors walked down accessible portions of the solid radioactive waste systems and evaluated system configuration and functionality
(2) Inspectors walked down accessible portions of the liquid radioactive waste systems and evaluated system configuration and functionality

Waste Characterization and Classification (IP Section 03.03) (2 Samples)

(1) The inspectors evaluated the licensee's characterization and classification of radioactive waste associated with waste shipment number PW-21-0022
(2) The inspectors evaluated the licensee's characterization and classification of radioactive waste associated with waste shipment number PW-20-0016

Shipment Preparation (IP Section 03.04) (1 Sample)

(1) The sample was not able to be completed due to a radioactive shipment not being available at the time of the inspection

Shipping Records (IP Section 03.05) (4 Samples)

The inspectors evaluated the following non-excepted radioactive material shipments through a record review:

(1) Radioactive waste shipment number PW-19-0040, UN 3321, Radioactive material 7, low specific activity (LSA-II)
(2) Radioactive waste shipment number PW-20-0016, UN 3321, Radioactive material 7, low specific activity (LSA-II)
(3) Radioactive waste shipment number PW-21-0020, UN 3321, Radioactive material 7, low specific activity (LSA-II)
(4) Radioactive waste shipment number PW-21-0022, UN 3321, Radioactive material 7, low specific activity (LSA-II)

OTHER ACTIVITIES – BASELINE

71151 - Performance Indicator Verification

The inspectors verified Exelon's performance indicator submittals listed below for the period October 1, 2020 through September 30, 2021:

MS07: High-Pressure Injection Systems (IP Section 02.06) (2 Samples)

(1) Unit 2 high-pressure injection systems
(2) Unit 3 high-pressure injection systems

MS08: Heat Removal Systems (IP Section 02.07) (2 Samples)

(1) Unit 2 heat removal systems
(2) Unit 3 heat removal systems

MS09: Residual Heat Removal Systems (IP Section 02.08) (2 Samples)

(1) Unit 2 RHR systems
(2) Unit 3 RHR systems

MS10: Cooling Water Support Systems (IP Section 02.09) (2 Samples)

(1) Unit 2 cooling water support systems
(2) Unit 3 cooling water support systems

71152 - Problem Identification and Resolution

Annual Follow-up of Selected Issues (IP Section 02.03) (2 Samples)

The inspectors reviewed the licensee's implementation of its corrective action program (CAP) related to the following issues:

(1) Unit 2 RHR motor-operated valve (MOV) MO-2-10-034B material failure of anti-rotation key
(2) Unit 2 Inboard Isolation Valve (AO-2-07B-2511) Failed Rendering Hardened Containment Vent System Non-Functional on April 19, 2020

71153 - Follow-Up of Events and Notices of Enforcement Discretion

Event Report (IP Section 03.02) (1 Sample)

The inspectors evaluated the following licensee event report (LER):

(1) LER 05000277/2021-002-00, SRV Inoperability Due to Nitrogen Leakage (ADAMS Accession No.: ML21196A485)

The inspection conclusions associated with this LER are documented in this report under Inspection Results Section.

INSPECTION RESULTS

Observation: Unit 2 Inboard Isolation Valve (AO-2-07B-2511) Failed Rendering Hardened Containment Vent System Non-Functional	71152
The inspectors reviewed Exelon's evaluation and corrective actions associated with the April 19, 2020, control room "Torus Vent Valves Open" alarm, as documented under AR 04339435. The inspectors concluded that Exelon had taken timely and appropriate actions in accordance with Exelon procedure PI-AA-125, "CAP Procedure," and 10 CFR Part 50, Appendix B. The inspectors determined that Exelon's associated failure analysis, cause evaluations (CAPE), extent of condition, operating experience review, and past functionality assessment based on equipment bench testing were sufficiently thorough to support their conclusions. Exelon's assigned corrective actions addressed the underlying cause (dimensional tolerance of woodruff key and keyways), were aligned with engineering evaluations, adequately tracked, appropriately documented, and completed as scheduled. Based on the documents reviewed, redundant system walkdowns with this valve type, and discussions with engineering personnel, the inspectors noted that Exelon staff identified problems and entered them into the CAP at an appropriate threshold in accordance with their procedures.	

Observation: Unit 2 RHR Motor-operated Valve (MOV) MO-2-10-034B Material Failure of Anti-rotation Key	71152
The inspectors reviewed a problem related to the Unit 2 RHR MOV MO-2-10-034B involving material failure of its anti-rotation key. Exelon staff identified the problem on October 5, 2020, during pump testing when the valve did not open. The inspectors determined that Exelon staff entered the problem into their CAP under issue report (IR) 04374639, promptly repaired and retested the valve returning it to service, and performed a CAPE (04374639-17). Exelon engineering staff concluded the problem resulted from excessive stem to plug clearance which allowed the valve plug to tilt and "catch" the seat guide on closure and then damage the anti-rotation key, which subsequently prevented valve opening during testing. Exelon staff concluded the non-vertical installation of this valve also contributed to the problem. Corrective actions involved valve stem-to-disc design clearance changes, the installation of a replacement stem and disc, and a recommendation for additional action to consider modifying the torque switch bypass setting. Exelon staff identified three other similar non-vertical installed valves could be affected and examined their anti-rotation keys to verify satisfactory condition. The inspectors observed Exelon's corrective actions were aligned with their engineering evaluation, appropriately documented, and were being tracked. Operability reviews declared the valve inoperable and returned to service within the applicable technical specification (TS) action statement. The inspectors concluded that, in general, Exelon staff completed timely and appropriate actions in accordance with their Exelon procedures PI-AA-120, "Issue Identification and Screening" and PI-AA-125, "CAP Procedure." Exelon procedure PI-AA-125-1003, "Corrective Action Program Evaluation Manual," notes an effective Extent of Condition Evaluation will ensure the right actions and organizational focus are in place to "minimize other similar conditions." The inspectors concluded that while Exelon staff promptly confirmed the anti-rotation keys for other similar valves were functional to address the "extent of condition" and they identified longer term corrective actions, a key verification check or other suitable short term action was not being completed to minimize this condition from recurring in the interim.	

Based on the documents reviewed and discussions with engineering personnel, the inspectors noted that, in general, Exelon personnel identified problems and entered them into the CAP at a low threshold.

Safety Relief Valve Inoperability Due to Nitrogen Leakage from Braided Hose Wear			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000277/2021003-01 Open/Closed	[H.7] - Documentation	71153
The inspectors identified a self-revealing Green finding and associated non-cited violation (NCV) of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." Specifically, stainless steel (SS) braided flexible hoses associated with safety relief valve (SRV) '71K' failed due to a dislodged support clip that was not installed properly because the work instructions lacked sufficient detail and therefore were not appropriate to the circumstances.			

Description: PBAPS Unit 2 has 11 SRVs that provide reactor pressure vessel overpressure protection, of which 5 are part of the automatic depressurization system (ADS). The ADS is an emergency core cooling system that is automatically initiated as a back up to high-pressure coolant injection (HPCI) to depressurize the reactor pressure vessel so that low pressure coolant injection systems can inject to cool the core and limit excessive fuel temperatures. The Unit 2 ADS consists of five air/nitrogen accumulators, associated isolation check valves, and air/nitrogen piping from the check valves to the accumulators and from the accumulators to the five ADS SRV's solenoid valves. The Unit 2 '71K' SRV was modified on November 5 to 11, 2016, to relocate its solenoid valve 'SV-071K' due to vibration-caused issues which added two SS braided flexible hoses routing the nitrogen supply to the solenoid valve and from the solenoid valve to the SRV.

On March 21, 2021, Exelon identified an increase in runtime hours for the Unit 2 nitrogen compressors which was later determined to be caused by a leak located within the drywell. On May 18, 2021, Exelon conducted a drywell entry and identified that the ADS SRV '71K' flexible hoses upstream and downstream of the solenoid valve were fretted with through-wall holes. Exelon determined that a nitrogen supply tubing clamp had failed which allowed the two SS braided hoses to move into contact with each other and fail from wear and abrasion caused by system induced vibration. Exelon determined the through-wall condition was present and statistically significant on March 5, 2021, based on an analysis of compressor runtimes.

As a result, Exelon submitted LER 2-2021-002-00, "SRV Inoperability Due to Nitrogen Leakage from Braided Hose Wear," on July 16, 2021, for a condition not allowed by TSs and a loss of safety function for the ADS system, based on subsequent analysis of the leakage. Exelon also performed a cause evaluation. Exelon determined that engineering change request (ECR) 16-00350, to relocate 'SV-071K' to isolate it from system vibration, was inadequate with respect to analysis and design of the tubing and supports, which was performed in November 2016. Specifically, not all drawings and design standards were included as required, and the drawings that were included lacked sufficient detail for how to construct and install the tubing support and stated, "support tubing from accumulator and at SRV per engineering direction."

On October 20, 2018, SRV '71K' and its associated solenoid valve were replaced as part of corrective maintenance for a suspected bellows leak. Finally, on October 25, 2020, SRV '71K' and its associated solenoid valve were replaced again as part of the scheduled preventive maintenance. Exelon reviewed the work orders performed in 2018 and 2020 and found no details or installation instructions for routing or supporting the tubing and no references to tubing support specifications or drawings. In addition, there was no record indicating the hoses were replaced and no details about disconnecting or reconnecting the nitrogen lines or supports associated with them. However, Exelon concluded that based on three distinct wear marks on the Unistrut, that it is likely that the clamp for the supply tubing of 'SV-2016-071K' was located/re-installed further and further away from the supported end of the cantilevered Unistrut support in 2018 and 2020. As a result, the angle that the clamp was installed became larger (exceeding design specifications), and the clamp was subjected to higher and higher operational vibration levels which resulted in the clamp loosening and then falling off the Unistrut within six months of operation.

The inspectors reviewed Exelon's records and the cause evaluation and determined the timelines and conclusions were reasonable. The inspectors also concluded that the engineering design change was inadequate, the work order instructions were inadequate, and the documentation of work performed was inadequate.

Corrective Actions: Exelon's corrective actions on May 18, 2021, included replacing the two SS braided flexible hoses and modifying the flexible hose connections and routing such that the hoses will remain apart even if not restrained by the clamp.

Corrective Action References: IR 4424199

Performance Assessment:

Performance Deficiency: The inspectors determined that Exelon's failure to establish and implement maintenance work instructions appropriate to the circumstances by not providing sufficient detail to route, clamp, and install SS braided flexible hoses associated with SRV '71K' was reasonably within their ability to foresee and correct and should have been prevented and therefore was a performance deficiency.

Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the flexible hoses fretting through-wall prevented SRV '71K' from operating to reduce reactor pressure vessel pressure during events.

Significance: The inspectors assessed the significance of the finding using Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." The inspectors screened this finding for safety significance and determined that the finding was of very low safety significance (Green). Specifically, although the function of the '71K' SRV was lost for a time that exceeded the allowed TS outage time, the probabilistic risk analysis (PRA) function of ADS was not lost with four remaining ADS SRVs and six remaining non-ADS SRVs. Therefore, the degraded condition did not represent a loss of the PRA function for a train or a system and screens to Green.

Cross-Cutting Aspect: H.7 - Documentation: The organization creates and maintains complete, accurate and up-to-date documentation. The inspectors determined the finding has a cross-cutting aspect in the area of Human Performance, Documentation. Specifically, the referenced common language in NUREG-2165, "Safety Culture Common Language," includes the example "Design documentation, procedures, and work packages are complete, thorough, accurate, and current," and Exelon failed to create and maintain sufficient documentation throughout this process, first with the design documentation in the ECR in 2016, then in the development and implementation of the associated work orders in 2018 and 2020, and finally with the documentation of work performed in 2016, 2018, and 2020.

Enforcement:

Violation: 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed and implemented by documented instructions, procedures, or drawings, of a type appropriate to the circumstances. Contrary to the above, from November 5, 2016, to May 18, 2021, activities affecting quality were not prescribed and implemented by documented instructions, procedures, and drawings, of a type appropriate to the circumstances. Specifically, the work instructions for the fabrication, routing, and installation of the tubing, clamps, and supports for the SS braided flexible hoses for SRV '71K' and its associated solenoid valve were not appropriate to the circumstances since they lacked sufficient detail to accomplish proper installation and were activities affecting quality. In addition, TS 3.5.1 requires for one ADS valve inoperable that the valve be restored to an operable status within 14 days, or the unit be placed in Mode 3 in the next 12 hours. Contrary to this, the ADS valve '71K' was not operable for a time that exceeded 14 days and the unit was not placed in Mode 3 since the valve was inoperable from about March 5, 2021, through May 18, 2021.

Enforcement Action: This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy.

EXIT MEETINGS AND DEBRIEFS

The inspectors verified no proprietary information was retained or documented in this report.

- On July 16, 2021, the inspectors presented the PI&R Hardened Torus Vent Valve Debrief inspection results to Ms. Amy Huber, Sr. Regulatory Engineer, and other members of the licensee staff.
- On July 23, 2021, the inspectors presented the PI&R 2B RHR MOV anti-rotation key failure inspection results to Mr. Matthew Rector, Regulatory Assurance Manager, and other members of the licensee staff.
- On July 30, 2021, the inspectors presented the radioactive solid waste processing and radioactive material handling, storage and transportation inspection results to Mr. Matthew Rector, Regulatory Assurance Manager, and other members of the licensee staff.
- On September 17, 2021, the inspectors presented the radiation monitoring instrumentation inspection results to Mr. Matthew Herr, Site Vice President, and other members of the licensee staff.
- On October 15, 2021, the inspectors presented the integrated inspection results to Mr. Matthew Herr, Site Vice President, and other members of the licensee staff.

THIRD PARTY REVIEWS

Inspectors reviewed Institute of Nuclear Power Operations reports that were issued during the inspection period.

DOCUMENTS REVIEWED

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
71111.04	Procedures	COL 10.1.A-3A	"RHR System Setup for Automatic Operation Loop A"	Revision 21
71111.05	Procedures	PF-117	Unit 2 Turbine Building, Emergency Battery Switchgear Rooms, Elevation 135'-0"	Revision 11
		PF-127	Unit 3 Turbine Building, Emergency Battery Switchgear Rooms, Elevation 135'-0"	Revision 11
		PF-132	Diesel Generator Building, General Area – Elevation 127'-0"	Revision 9
		PF-132A	Diesel Generator Building, General Area (Upper Level)	Revision 4
		PF-144	Circulating Water Pump Structure – General Area	Revision 7
71111.06	Corrective Action Documents	04438979, 04441542, 04441547		
	Work Orders	01497874		
71111.07A	Corrective Action Documents	AR 02717440-02		
	Procedures	ER-AA-340-1002	"Service Water Heat Exchanger Inspection Guide"	Revision 10
	Work Orders	05062088-01, 05062088-02, 05062088-08		
71111.11Q	Procedures	AO 1E.4-3	Planned Removal of the Fifth or Fourth Stage Feedwater Heaters from Service During End of Cycle Coastdown	Revision 28
		GP-5-3	Power Operations	Revision 15
		OP-AA-20	Conduct of Operations Process Description	Revision 1
		SO 2A.1.D-3	Operation of the Recirc Pump Speed Control System	Revision 20
71111.12	Corrective Action Documents	04422952, 04442169, 04442523, 04442622		
	Miscellaneous		Engage Health SR Service Water	
	Work Orders	05183208		
71111.13	Corrective Action Documents	IR 4442598		
	Miscellaneous	Paragon RMTS	PB3_PRD_Z_007_RICT	

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		Report	WW2134 – 3B/D HPSW RICT Review	
	Procedures	ER-PB-600-2001	"Peach Bottom RICT System Guidelines"	Revision 1
71111.15	Corrective Action Documents	04437761		
		AR 04442523		
		IR 4431014		
		IR 4443607		
	Drawings	HSO-3260	HP Service Water	Revision 5
	Miscellaneous	EC 634956		
	Procedures	CC-AA-309-101	Engineering Technical Evaluation	Revision 16
		ER-AA-600-1042	On-line Risk Management	Revision 13
		M-052-002	"Diesel Engine Maintenance"	Revision 56
		SE-16	Grid Emergency	Revision 18
		SE-16	Attachment D: Contingency Issues	Revision 3
		WC-AA-101	On-Line Work Control Process	Revision 31
		WC-AA-101-1006	On-line Risk Management and Assessment	Revision 4
	Work Orders	05181736		
71111.18	Corrective Action Documents	AR 04426127		
	Miscellaneous	ECN	634522	
		TCC	21-0048	
	Procedures	HU-AA-1212	"Technical Task Risk/Rigor Assessment, Pre-Job Brief, Independent Third-Party Review, and Post-Job Review"	Review 11
		IP-ENG-001	"Standard Design Process"	Revision 2
		SO 12.7.B-3	"RWCU System Restoration After Isolation"	Revision 2
71111.19	Corrective Action Documents	04437494		
		AR 04443036		
	Procedures	M-057-014	Cyberex 125 Volt Battery Charger Maintenance	Revision 18
		ST-M-57B-761-2	Battery Charger 2AD003-1 and 2AD003-2 Capability Test	Revision 8
		ST-O-032-301-3	"HPSW Pump, Valve, and Flow Functional and Inservice Test"	Revision 40
		ST-O-37D-340-2	"Diesel Driven Fire Pump Flow Rate Test"	
	Work Orders	04234158, 05108396,		

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		05177990		
		4861874-01, 4861874-02, 4861874-03		
71111.22	Corrective Action Documents	04439594		
		AR 2648765		
	Miscellaneous		Chemistry and Gamma Spec. Analysis	
	Procedures	RT-O-020-100-3		Revision 2
		SI2M-60F-RT23-A4M2	"Response Time Test of MSIV Closure Scram Channel A"	Revision 1
		SO 39.1.A	Flex Generator Startup and Shutdown	Revision 4
		ST-I-010-100-3	RHR Loop 'A' Logic System Functional Test	Revision 25
	Work Orders	5014569		
		5131909, 5131910, 5131911		
		5168509		
71124.05	Corrective Action Documents	4415109		
	Procedures	RP-AA-700-1235	Operation and Calibration of the PM-12 Gamma Portal Monitor	5
		RP-AA-700-1239	Operation and Calibration of the Model SAM-12 Small Articles Monitor	6
		RP-AA-800	Control, Inventory, and Leak Testing of Radioactive Sources	10
71124.08	Corrective Action Documents	04269107		
		04329294		
	Corrective Action Documents Resulting from Inspection	04437507		
	Procedures	RP-AA-600	Radioactive Material/Waste Shipments	Revision 18
71152	Corrective Action Documents	01206317	Surveillance Frequency Control Program, Assign 10, Inflatable Seal Valves from 8Y to 12Y - SR 3.6.1.3.16 ST-M-007-440-2(3) Current frequency is 96 months. Proposed	04/22/2011

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
			frequency is 144 months.	
		01410761	AO-2511 Exceeded Stroke Time Limit	09/09/2012
		4339435	Received the "Torus Vent Valves Open' Alarm"	04/29/2020
		A0127590	PMC-13-089919, U/2 Boot Seal Valve PMCR - 4R to 6R	
	Engineering Changes	EC-632926	Confirm new torque key dimensions are adequate	Revision 1
	Miscellaneous	6280-M117-64-1	Vendor Manual - Type 9200 T-Ring Butterfly Valve	10/01/1985
	Procedures	CC-PB-118	Peach Bottom Implementation of Diverse and Flexible Coping Strategies (FLEX) and Spent Fuel Pool Instrumentation Program	Revision 13
		M-040-002	Fisher Type 9200 T-Ring Butterfly Valve Maintenance	Revision 12
		M-040-002	Fisher Type 9200 T-Ring Butterfly Valve Maintenance	Revision 13
	Work Orders	01470811	Torus 18-inch Vent Inboard Isolation Valve to SBGT/ATMOS-OPER	07/14/2020
		04247914	AO-2-07B-2511-OP: PM Replace T-Ring/Access's/Booster Relay	10/08/2020
		04842474	ST-O-007-440-2, PCIS Containment Atmospheric Control and Drywell Ventilation Valves Inservice Test	10/14/2018
		5035700	AO-2-07B-2511 PFIN Support Resting / Repair COV Arm	06/16/2020
		C0219014	MO-2-10-034B Inbody Repair	05/01/2007
		C0244596	Investigate/Rework/Replace Assoc. Components on A0-2-07B-2511	01/03/2013
		R0922217	AO-2-07B-2511-OP: OP PM, Replace T-Ring/Access's	11/15/2010

December 14, 2021

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
2100 RENAISSANCE BLVD., SUITE 100
KING OF PRUSSIA, PENNSYLVANIA 19406-2713

December 14, 2021

Mr. David P. Rhoades
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 – INFORMATION REQUEST TO SUPPORT TRIENNIAL BASELINE DESIGN-BASIS CAPABILITY OF POWER-OPERATED VALVES INSPECTION; INSPECTION REPORT 05000277/2022011 AND 05000278/2022011

Dear Mr. Rhoades:

The purpose of this letter is to notify you that the U.S. Nuclear Regulatory Commission (NRC) Region I staff will conduct a team inspection at Peach Bottom Atomic Power Station, Units 2 and 3. David Kern, a Senior Reactor Inspector from the NRC's Region I Office, will lead the inspection team. The inspection will be conducted in accordance with Inspection Procedure 71111.21N.02, "Design-Basis Capability of Power-Operated Valves Under 10 CFR 50.55a Requirements," dated October 9, 2020 (ADAMS Accession No. ML20220A667).

The inspection will assess the reliability, functional capability, and design bases of risk-important power-operated valves (POVs) as required by Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a, and Appendix A and B requirements. The inspectors will select a sample of POVs based on risk insights, safety significance, and operating margin.

During a telephone conversation on December 7, 2021, with Ms. Amy Huber, Senior Regulatory Engineer, we confirmed arrangements for an information gathering visit and the two-week onsite inspection. Depending on site access conditions, the information gathering visit may be onsite or may be performed remotely. The schedule is as follows:

- Information gathering visit: Week of January 3, 2022
- Onsite weeks: Weeks of March 21 and April 4, 2022

The purpose of the information gathering visit is to meet with members of your staff and to become familiar with your programs and procedures intended to ensure compliance with 10 CFR 50.55a for POVs. The lead inspector will discuss aspects of the programs including any specific applicable regulatory commitments made by your facility and your use of NRC regulatory guides or industry standards.

Experience with previous design basis team inspections of similar depth and length has shown this type of inspection is resource intensive, both for NRC inspectors and licensee staff. In order to minimize the inspection impact on the site and to ensure a productive inspection for both parties, we have enclosed a request for information needed for the inspection.

It is important that all of these documents are up-to-date and complete in order to minimize the number of additional documents requested during the preparation and onsite portions of the inspection. Insofar as possible, this information should be provided electronically to the lead inspector at the NRC Region I Office by January 3, 2022. Recognizing the timeframe, my staff will work with your staff to prioritize our document requests so these activities can be accomplished, as much as possible, in the normal course of your activities. Particularly considering the end of year timeframe, please do not hesitate to contact Mr. Dave Kern if there are challenges in document retrieval. We will work with your staff. Additional documents may be requested during the information gathering visit and/or during team preparation week (the week prior to the first onsite inspection week). The inspectors will minimize your administrative burden by specifically identifying only those documents required for the inspection.

If there are any questions about the inspection or the material requested in the enclosure, please contact the lead inspector at 610-337-6931 or via e-mail at David.Kern@nrc.gov.

This letter does not contain new or amended information collection requirements subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). Existing information collection requirements were approved by the Office of Management and Budget, Control Number 3150-0011. The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid Office of Management and Budget Control Number.

This letter and its enclosure will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with 10 CFR 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Sincerely,

Melvin K. Gray Digitally signed by Melvin K. Gray Date: 2021.12.14 12:55:34 -05'00'

Mel Gray, Chief
Engineering Branch 1
Division of Operating Reactor Safety

Docket Nos. 05000277 and 05000278
License Nos. DPR-44 and DPR-56

Enclosure:
Document Request for Design Bases Assurance Inspection

cc: Distribution via ListServ ®

DOCUMENT REQUEST FOR DESIGN BASES ASSURANCE INSPECTION

Inspection Report:	05000277/2022011 and 05000278/2022011
Onsite Inspection Dates:	March 21 through March 25, 2022; and April 4 through 8, 2022
Inspection Procedure:	Inspection Procedure 71111.21N.02, Design-Basis Capability of Power-Operated Valves Under 10 CFR 50.55a Requirements
Lead Inspector:	David Kern, Senior Reactor Inspector 610-337-6931 David.Kern@nrc.gov

I. Information Gathering Visit

During this visit, we plan to obtain sufficient insights to finalize power-operated valve (POV) samples for this inspection. We would like to meet with POV specialists to discuss the upcoming inspection and our sample selection process. The primary valve types to be reviewed for this inspection include motor-operated valves (MOVs) and air-operated valves (AOVs); and additional valve types include hydraulic-operated valves (HOVs), solenoid-operated valves (SOVs), and pyrotechnic-actuated (squib) valves. During this visit, the lead inspector will: (a) discuss the scope of the planned inspection; (b) identify additional information needed to review in preparation for the inspection; (c) ensure that the information to be reviewed is available at the beginning of the inspection; and (d) verify that logistical issues will be identified and addressed prior to the team's arrival. Depending on the local COVID environment and potential travel restrictions, this visit may be either onsite or performed remotely through a series of skype video calls. If performed onsite, please reserve a room during the site visit with a telephone, wireless internet access, and a licensee computer with access to procedures, corrective action program documents, and a printer.

II. Information Requested for Selection of Power-Operated Valves

The following information is requested by January 3, 2022, to facilitate inspection preparation. Feel free to contact the lead inspector if you have any questions regarding this information request. Please provide the information electronically in "pdf" files, Excel, or other searchable formats. The files should contain descriptive names, and be indexed and hyperlinked to facilitate ease of use. Information in "lists" should contain enough information to be easily understood by someone who has knowledge of light water reactor technology and POVs.

1. A word-searchable Updated Final Safety Analysis Report. If not available in a single file for each unit, please ensure a collective table of contents is provided.

2. Site (and corporate if applicable) procedures associated with implementation of the MOV program required by 10 CFR 50.55a(b)(3)(ii) and/or ASME OM Code Mandatory Appendix III; and site (corporate) procedure for AOV program.

Enclosure

3. Site response(s) to NRC Generic Letter (GL) 95-07, Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves.

4. Site response(s) to NRC GL 96-05, Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves.

5. Site evaluation of NRC Information Notice 2012-14, MOV Inoperable due to Stem-Disc Separation.

6. List of corrective action documents related to the MOV and AOV programs since January 1, 2017 (include document No., title/short description, date).

7. List of corrective action documents related to each of the 30 POVs listed below since January 1, 2017 (include document No., title/short description, date).

8. List of significant modifications, repairs, or replacement of safety-related POVs completed since January 1, 2017, including date completed (include document No., title, date completed).

9. List of POVs removed from the In-Service Test program since January 1, 1990.

10. Any self-assessments or quality assurance type assessments of the MOV/AOV programs (performed since January 1, 2017).

11. Most recent POV (e.g., MOV, AOV, SOV) program health report(s).

12. List and electronic copy of all Emergency Operating Procedures.

13. List of Abnormal Operating Procedures.

14. Identify the edition of the ASME Operation and Maintenance of Nuclear Power Plants (OM Code) that is the Code of Record for the current 10-year Inservice Test Program interval, as well as any standards to which the station has committed with respect to POV capability and testing.

15. Identify which of the valves listed in items #16 and #17 are located in harsh environment areas and subject to Environmental Qualification (EQ) requirements.

16. For each of the following MOVs, provide the information listed in the table below.

- MO-0-33-0498 ESW Return to Discharge Pond
- MO-2-12-15 RWCU Inlet Inboard Isolation Valve
- MO-2-10-017 RHR Shutdown Cooling Suction Outboard Isolation Valve
- MO-2-10-026B RHR Loop 'B', Drywell Spray Outboard Isolation Valve
- MO-2-13-015 RCIC Steam Line Inboard Isolation Valve
- MO-2-14-012B Core Spray Loop 'B', Inboard Discharge Isolation Valve
- MO-2-23-016 HPCI Turbine Steam Line Outboard Isolation Valve
- MO-2-23-020 HPCI Pump Discharge Valve
- MO-3-02-053B Recirculation Pump Discharge Isolation Valve
- MO-3-10-013A RHR Pump 3AP035 Torus Suction Isolation Valve

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- MO-3-10-018 RHR Pump Shutdown Cooling Suction Inboard Isolation Valve
- MO-3-10-025A RHR Loop 'A', Inboard Discharge Valve
- MO-3-10-039A RHR Loop 'A', Outer Block for Torus Cooling Spray
- MO-3-10-089A RHR HX 3AE024 HPSW Outlet Valve
- MO-3-13-016 RCIC Steam Line Outboard Isolation Valve
- MO-3-13-021 RCIC Pump Discharge to 'B' Feedwater Line
- MO-3-14-005D 'D' Core Spray Pump Minimum Flow Valve
- MO-3-23-014 HPCI Turbine Steam Supply Valve
- MO-3-23-015 HPCI Turbine Steam Line Inboard Isolation Valve

Item	Parameter/Information*
1	MOV Identification
2	Safety Function
3	Valve manufacturer, type, and size
4	Actuator manufacturer, type, and size
5	Motor manufacturer, type (AC/DC), and size
6	Valve ASME Class
7	Risk Significance
8	Control Switch Trip (CST) Application (Close/Open)
9	Design-Basis Differential Pressure (DBDP) and Flow (Close/Open)
10	Rising-Stem Valve: Assumed Valve Factor (VF)
11	Quarter-Turn Valve: Assumed bearing torque coefficient
12	Assumed Stem Friction Coefficient (SFC)
13	Assumed Load Sensitive Behavior (LSB) (%)
14	% Uncertainties (e.g., diagnostic equipment, CST repeatability, etc.)
15	Calculated Required Thrust/Torque (Close/Open)
16	Least Available Output (e.g., actuator, CST, rating, spring pack, weak link)
17	Test Conditions (e.g., fluid differential pressure (DP), system pressure, flow, and temperature; ambient temperature; and motor voltage) (Close/Open)
18	Thrust and torque required to overcome dynamic conditions (Close/Open)
19	Rising-Stem Valve: Measured VF (Close/Open)
20	Rising-Stem Valve: Available VF (Close/Open)
21	Measured SFC (Close/Open)
22	Measured LSB (%)
23	Quarter-Turn Valve: Measured bearing torque coefficient (Close/Open)
24	Determined % Margin (Close/Open)
25	*Basis for Design-Basis Capability:*
25.a	Dynamic test performed at design-basis DP/flow conditions
25.b	Extrapolation of dynamic test data
25.c	Justification from normal operation at or above design-basis conditions
25.d	Industry dynamic test methodology (such as EPRI MOV PPM)
25.e	Grouped with similar valves dynamically tested at plant
25.f	Grouped with similar valves dynamically tested at other plants
25.g	Valve qualification testing (such as ASME QME-1-2007)
25.h	Other (such as large calculated margin)
	**Specify Not Applicable (NA) as appropriate*

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16. For each of the following AOVs/SOVs/HOVs, provide the information listed in the table below.

    - AO-0-33-0241B ESW Outlet Block Valve from E2 DG Coolers
    - AO-2-01A-080D Inboard 'D' MSIV
    - AO-2-01A-086D Outboard 'D' MSIV
    - AO-2-03-35B Scram Discharge Volume Outboard Isolation Vent Valve
    - AO-2-03-36 Scram Discharge Volume Outboard Isolation Drain Valve
    - AO-2-07B-2511 Torus 18" Vent Inboard Isolation Valve to SBGT/ATMOS
    - AO-2-20-83 Drywell Floor Drain Sump Outboard Isolation Valve
    - HO-2-13C-4495 Steam Supply to RCIC Turbine Governor Valve
    - AO-3-01A-080A Inboard 'A' MSIV
    - AO-3-16-5235 Instrument Nitrogen System Suction Isolation Valve
    - HO-3-23C-5513 HPCI Turbine Stop Valve

Item	Parameter/Information*
1	AOV Identification
2	Safety Function
3	Fail safe position (open/close)
4	Valve manufacturer, type, and size
5	Actuator manufacturer, type, and size
6	Valve ASME Class
7	Risk Significance
8	Design-Basis Differential Pressure (DBDP) and Flow (Close/Open)
9	Rising-Stem Valve: Assumed Valve Factor (VF)
10	Quarter-Turn Valve: Assumed bearing torque coefficient
11	% Uncertainties (e.g., diagnostic equipment, CST repeatability, etc.)
12	Calculated Required Thrust/Torque (Close/Open)
13	Minimum allowable air pressure (Beginning/End Stroke)
14	Maximum allowable air pressure (Beginning/End Stroke)
15	Minimum allowable spring preload (Beginning/End Stroke)
16	Maximum allowable spring preload (Beginning/End Stroke)
17	Least Available Actuator Output (e.g., actuator capability, actuator limit, valve weak link limitation)
18	Test Conditions (e.g., fluid differential pressure (DP), system pressure, flow, and temperature; and ambient temperature) (Close/Open)
19	Thrust and torque required to overcome dynamic conditions (Close/Open)
20	Rising-Stem Valve: Measured VF (Close/Open)
21	Quarter-Turn Valve: Measured bearing torque coefficient (Close/Open)
22	Determined Margin (%) (Least margin for air stroke operation, spring stroke operation, maximum spring load, and structural capability)
23	*Basis for Design-Basis Capability:*
24.a	Dynamic test performed at design-basis DP/flow conditions
24.b	Extrapolation of dynamic test data
24.c	Justification from normal operation at or above design-basis conditions
24.d	Industry dynamic test methodology
24.e	Grouped with similar valves dynamically tested at plant
24.f	Grouped with similar valves dynamically tested at other plants

24.g	Valve qualification testing (such as ASME QME-1-2007)
24.h	Other (such as large calculated margin)
**Specify Not Applicable (NA) as appropriate*	