Incident Chronology at Susquehanna Steam Electric Station in Berwick: 1982- 2021

CHRONOLOGY of PROBLEMS at the
SUSQUEHANNA STEAM ELECTRIC STATION

This chronology does not include the cost to the rate payer to build Susquehanna-1 and -2. PP&L asked the Public Utility Commission (PUC) for $315 million to recover the cost of building Unit-1. The PUC granted $203 million on August 22, 1983, or a 16% increase to the customer. The company asked for $330 million for Unit-2 but was allowed $121 million in April, 1985; an 8% increase to rate payers. In addition, PP&L consumers have "contributed" approximately $4.6 million annually (since 1985) to the decommissioning fund.
(Also, refer to May 15 and August 13, 1998, for information on "stranded costs" passed on to "hostage" PP&L rate payers.)
Moreover, in the Winter 1999/2000, PPL unilaterally devaluated the combined PURTA and Real Estate tax assessments for the SSES. Prior to the Negotiated Settlement, the nuclear power generating stations were assessed by PP&L at approximately $1 billion. PPL is now claiming that the the SSES is only worth $74 million or the same amount as the valuation of the Columbia Hospital. If PPL prevails, the Berwick School District and Luzerne County will experience revenue shock. PPL is not paying or escrowing any moneys they owe to Luzerne County and the Berwick School District.
(See April 23, 2001 and July 13, 2003, for related development).
i The Susquehanna Steam Electric Station is owned by PP&L (90%) and the Allegheny Electric Cooperative (10%). The Allegheny Electric Cooperative (AEC) is responsible for 10% of the cost of decommissioning. PP&L's consultant, TLG, estimated PP&L's decommissioning share to be $724 million. Therefore, the AEC is responsible for the remaining 10%, or $79 million, of the $804 million projected funding "target" for nuclear decommissioning.
At the Susquehanna Steam Electric Station, projected costs for decommissioning have increased by 553% since 1981-1993. In 1981, PP&L engineer Alvin Weinstein predicted that PP&L's share to decommission SSES would fall between $135 and $191 million. By 1985, the cost estimate had climbed to $285 million, and by 1991 the cost in 1988 dollars for the "radioactive portion" of decommissioning was $350 million. The Company then contracted out for a site-specific study which projected that the cost of immediate decommissioning [DECON] would be $725 million in 1993 dollars. The 1994 cost estimate remained steady at $724 million, but the market value of securities held and accrued in income in the trust funds declined, and thus the estimate reflected another increase in decommissioning costs.

PPL's share to decommission the SSES is projected to be $936 million in 2002 dollars (2002, Annual Report).
ii - September 22, 1982 - An emergency was declared at the plant. (UPI, September 22, 1982.)

August 6, 1982 - UPI reported PP&L announced it was investigating nuclear plant allegations; however, the utility initially denied the complaints on December 29, 1981. (UPI, December 29, 1981.)

January 21, 1983 - UPI reported, "Another spill at the Susquehanna nuclear plant."

March 29, 1983 - UPI reported, "Nuclear plant workers evacuated, Berwick, Pa."

June 9, 1983 - Unit-1 went commercial. The plant was at 100% power in February, and has been operating at full-power since May 23, 1983. (AP, June 9, 1983).

June 14, 1983 - Susquehanna was forced to shut down. The incident was termed "minor." (UPI, June 14, 1983.) However, the Company later admitted "the reactor shut down when an usually high degree of radiation was detected..." (AP, June 25, 1983).

June 25, 1983 - Susquehanna automatically to shut down due to an electrical problem inside a transformer.

"Eight hours after the shut down, workers were still trying to determine the nature of the malfunction, spokesman Ira Kaplan said. He said the plant would not be restarted until the transformer is repaired." (UPI, June 14, 1983.)
(Please reference the following dates for a list of chronic electrical problems at the SSES: "1986"; September, 1988; February 6, 1990; July 23, 1997; June 8-16, 1999; April 8, 2004; and, April 12, 2005.)
- The SSES provides 20% of the commercial power PP&L supplies to its customers. (See September 5, 1989, for new figures.)- April 26, 1984 - "Nuclear plant water discharges studied" (UPI, April 26, 1984.)

July 26, 1984 - An "unusual event" was declared. (UPI, July 26, 1984.)

August 9, 1983 - The New Jersey Public Utilities Board refused to pass on excess costs to rate payers as a result Atlantic City Electric's purchase of 125 megawatts (almost 6% of the SSES output) from PP&L. ACE has refused to to take any power from the Susquehanna Steam Electric Station. The power agreement was valued at $30 million.

1985 - 1994 - PP&L cut 1,600 jobs over this period.
(Please refer to November 14, 1995 and June 19, 2002, for more terminations.)

1986 - PP&L reported safety violations to the NRC "after it discovered that a number of cable splices and electrical terminals did not meet new standards passed in 1985. We did have some of those terminal blocks and splices in service beyond the date were were supposed to be in compliance" according to PP&L spokesman, Herb Woodeshick. (UPI, September, 1988.
(See September, 1988, for information on a $50,000 fine.)
(Please reference the following dates for a list of chronic electrical problems at the SSES: June 25, 1983; September, 1988; February 6, 1990; July 23, 1997; June 8-16, 1999; April 8, 2004; and April 12, 2005).

September 23, 1987 - A "low-level emergency " was declared when an "800-pound steel plug fell out of steam line during a test." (AP.)

October 1, 1987 - Prior to the contamination of four PPL employees (See below), "a relief valve opened in Unit 1 pump room, allowing about 1,300 gallons of contaminated water to spill onto the floor." Company spokesman Ira Kaplan quipped, "We're no precisely sure what happened. The valve opened and when it did the water spilled out on the floor" (UPI, October 1, 1987.) - October 1, 1987 - "Four workers contaminated, Berwick, Pa." (UPI, October 1, 1987.) After the workers were decontaminated, PPL spokesman Ira Kaplan observed, "It is not unusual to have people contaminated, especially during an outage. (AP.) (See August, 1989 and January 19, 1992, for related incidents.)

September, 1988 - The NRC leveled a $50,000 fine against Pennsylvania Power & Light for not properly testing electrical equipment. (See "1986" for background information).
(Please reference the following dates for a list of chronic electrical problems at the SSES: June 25, 1983; "1986"; February 6, 1990; July 23, 1997; June 8-16, 1999; April 8, 2004; and April 12,

2 0 0 5) .

August, 1989 - The NRC reported that a contracted employee received "a significant exposure" to radiation. NRC Inspector Jim Stair stated that the Commission is reviewing the incident and levy a fine. (Patriot News, September 15, 1989.) (See October 1, 1987 and January 19, 1992, related incidents).

September 5 , 1989 - The SSES provides about 30% of the commercial power PP&L supplies to its customers. (See June 25, 1983, for initial figures.)

April 11, 1989 - An "unusual event" was declared at the plant. (UPI, April 11, 1989.)

February 6, 1990 - "A short circuit Saturday that temporarily cut off cooling water to the Unit 1 reactor at the Susquehanna Nuclear plant...has been traced to a failed insulator, according to the unclear Regulatory Commission." ("Patriot News", February 6, 1990.)

(Please reference the following dates for a list of chronic electrical problems at the SSES: June 25, 1983; "1986"; September, 1988; July 23, 1997; June 8-16, 1999; April 8, 2004; and April 12, 2 0 0 5) .

November 28, 1990 - "The Nuclear Regulatory Commission Wednesday fined Pennsylvania Power & Light $25,000 for failing to promptly certify that components at its Susquehanna nuclear power plant would continue to function during an accident. The Allentown-based utility said it would not contest the fine." (UPI, November 28, 1990.)

March 5 and 9, 1992 - PP&L received $55 million in a settlement with General Electric over the Mark II containment structure. ("Electric Utility Week" and "Nucleonics Week.") The rate payers received a $55 million amortized rebate over five years beginning on April 1, 1992 and ending March 31, 1997. The arrangement was approved by the PUC as part of a Special Base Rate Credit Adjustment. (Docket # P91052). Customers rates decreased by .59%.

July 30, 1992 - Federal regulators say that a safety mechanism used by three Pennsylvania nuclear power plants [including Susquehanna] might fail to alert operators about a drop in the water level -- a condition which could lead to a nuclear accident." (States News Service, July 30, 1992.)

January 19, 1992 - PP&L Shareowners' Newsletter, February 3, 1992: "One of our employees was injured in a small hydrogen explosion and contaminated with radioactive material. He suffered burns to his chest and face...A second employee was examined and released after complaining of ringing in the ears after the explosion."
"The accident occurred in the basement of the plant's turbine building during work on an out-of-service recombiner -- equipment that combines hydrogen and oxygen to make water. A review team has found that a leak in a valve on the system allowed the hydrogen gas to build up in the pipe where the employee was working with a grinding wheel. New work procedures have been put in place to more clearly label hazards, and to institute safeguards aimed at preventing such incidents in the future." (See October 1, 1987 & August, 1989, for related incidents.)- December 31, 1992 - Two PP&L engineers charged that Susquehanna's highly radioactive spent fuel pools are unsafe and that if emergency cooling systems fail, a meltdown of spent fuel elements could occur. They told the NRC they reported their concerns to PP&L in March, 1992, and the company dismissed the matter and then tried to fire the engineers. The engineers, Donald Prevatte and David Lochbaum, are consultants for several companies. PP&L's spent fuel pool design is utilized by 1/3 of the nation's 109 nuclear power plants. (See October 1, 1993 for follow-up, February 9, 1996 and 1998 for similar patters of harassment.)

March 7, 1993 - PP&L backed a reduction in nuclear power plant drug testing. According to the Times-Leader, "Only four employees at the Susquehanna nuclear power plant tested positive for drugs and alcohol in 1992, fewer than the previous year."

May 26, 1993 - PP&L "determined that the 'C' EDG level indicating instrument had drifted in a nonconservative direction." (LER, 93-003.)

July 1, 1993 - An INPO inspection "pointed out some areas for improvement at the plant, and we're taking appropriate action." (Shareowners' Newsletter, July 1, 1993.)

July 12, 1993 - While Unit -1 was operating at 100%

power, a reactor scram occurred when the Main Turbine tripped. (LER, 93-008.)

July 12 to August 1, 1993 - Mechanical problems forced Unit-1 out of service for seven weeks. “The unit shut down automatically July 12 when vibrations caused two large turbine blades to break loose, damaging the turbine and other nonnuclear components of the unit.” (PPL, Shareowners’ Newsletter, October 1, 1993.) (Refer to July 1- 15, 1999, for related problems). - September 10, 1993 - Power at Unit-2 was reduced to 40% for “control rod sequence” and “reactor recirc motor generator set brush change outs.”

September 24, 1993 - A power reduction was initiated at Unit-1 due to the inoperability of RHR instrumentation; power was held at 26%. (Refer to February 28 and August, 1999, for related problems).

October 1, 1993 - During an NRC presentation, David Lochbaum and Donald Prevatte postulated that failure in spent fuel pool cooling could possibly lead to safety-related equipment failure and a full core meltdown. (See July 30, 1992.)

October 28, 1993 - At Unit-1, “PP&L suspended [fuel] loading after experiencing three fuel-loading problems in a 36 hour period” (”Patriot,” February 2, 1994.) Unit-1 was due to be back on line by November but not return to service until January 22, 1994; four days after a record demand for electric. (See July 1 and August 1994 for follow-up.)

January 1, 1994 - “Unit-1 at our Susquehanna nuclear plant, out of service since Sept. 25 for refueling and maintenance, is expected to resume operation in early January. Its return was delayed by a series of problems with our fuelloading operations...In an unrelated development, we further extended the refueling outage to replace metal support beams for pumps that circulate water inside the reactor. We took the action after problems developed with the components at a similar nuclear plant in Mississippi [Grand Gulf]” (PPL, Shareowners’ Newsletter, January 1, 1994.)

January 22, 1994 - Unit-2 tripped and created further

problems for the PJM depleted grid. (Refer to June 28, 2000, for reliability related problems at the SSES.)
(Also, see May 9, 2000 & January through March, 2001, for PJM problems related to PPL. Refer to June 14, 2002, October 19, 2002, and June 19, 2003, for incidents involving PPL's manipulation of the PJM grid). - July 1, 1994 - "The extended refueling outage at Unit-1 last October resulted in two citations from the NRC, but the agency decided that a fine was not appropriate, noting the prompt and effective actions we took to prevent future fuelhandling problems...The citations dealt with violations of certain NRC requirements during portions of the refueling outage" (PPL Shareowners Newsletter, July 1, 1994.) (See October 28, 1993 and August 1994 for related incidents.)

August, 1994 - "Safety is our first priority at Susquehanna, and the NRC evaluation [SALP] reflects our continuing emphasis on it. It also points out some areas where we can improve, including refueling activities and corrective action programs" (PPL, Connect, August 1994.) (See October 28, 1993, and July 1, 1994 for related incidents.)

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 material used in more than 70 nuclear reactors [including Susquehanna]" (The Nuclear Monitor, October 17, 1994.)
(For related incidents, see April 14, 1995 and October 1, 1996.)

December 1994 - PP&L joined a consortium of 33 nuclear utilities actively pressuring the Mescalero Apaches to accept high-level radioactive waste.

January 1 through December 31, 1995 - Unit-1 complied 18 Licensee Event Reports (LER) and one Severity Level III violation. Susquehanna 2 listed 17 LERs and one Severity Level III and IV violation. (Nuclear Regulatory Commission.)- March 16, 1995 - PP&L agreed to pay the PUC $300,000 to settle alleged violations of customer service requirements. The Settlement is the result of an informal PUC Bureau of Consumer Services investigation concluded in October, 1994. (See June 28, 1999, for related behavior.)

April 14, 1995 - “Documents obtained by NIRS under the Freedom of Information Act indicate that Pennsylvania Power & Light (PPL) conducted its own tests of Thermo-Lag in 1981 prior to its installation at Susquehanna. Under standard testing criteria, the Thermo-Lag failed the tests. But PP&L used it anyway. (For related developments see September 29, 1994 and October 1, 1996.)
“The Problem was discovered by the NRC’s Office of Inspector General in 1992, and the NRC staff investigated the issue. The staff found other fire protection violations as well, but issued no fines and did not even cite PP&L for the ThermoLag violation.” (The Nuclear Monitor,” April 10, 1995.) (See September 29, 1994.)

April 15, 1995 - Unit-2 scrammed. The uninterruptible power supply failed during recovery. (See June 6, 1995 for related incident.)

June 6, 1995 - Unit-2 was at 100% power when a loss of instrument AC at panels 2Y218 and 2Y219 occurred due to the failure of uninterruptible power supply (UPS) 2D240.” NRC, MR Number 1-95-0081. Dockets: 50-238, BWR/GE-4.) (See April 15, 1995 for related incident.)

August 22, 1995 “...while performing a fuel shuffle from the Unit 2 fuel vault to the fuel preparation machine, a new fuel bundle fell into the fuel preparation machine in the spent fuel pool when the grapple separated from the hoist cable. The bundle was being lowered into the machine at the time of the event and the bundle fell approximately 15-20 feet through water until it impacted the lower carriage support plate.” Morning Report-Region I, August 23, 1995.)
(See February 1, 1999 & August 5, 2002, for related events). - November, 1995 - PPL rebuffs two efforts by PECO to acquire PP&L in a hostile acquisition.

November 14, 1995 - PPL cut 300 jobs or 4.5% of its work force in an attempt to cut $671 million in operating costs. (See “From, 1985 - 1994” and June 19, 2002, for more job cuts.)

December 11, 1995 - A nonconservative error was reported in core thermal power calculations for both units. As a result, “Both units were reduced in power by 2 MWe to account for the discrepancy.” (“Licensee 24 Hour Report,” December 11, 1995.)

- 1996 - New Accounting Standards, SFAS 121 adopted on January 1, 1996. Previous standards relied on SFAS 71. (Refer to 2002 for a related development.)

January 1 through May 31, 1996 - Susquehanna 1 listed nine Licensee Event Reports (LER) and two Severity Level IV violations. Unit 2 compiled two LER’s and and three Level IV violations. (Nuclear Regulatory Commission.)

February 9, 1996 - The NRC informed PP&L that the Company would be fined $100,000 for disciplining a security officer for raising safety concerns in 1992. In October, 1995, the United States Department of Labor found that the security officer was “subjected to adverse action” for raising concerns about the the administration of security requalification exam. (See October 1, 1993, February 9, 1996 and 1998 for similar patters of harassment.)

June 12, 1996 - “A third alleged violation which was cited but for which no fine has been proposed involved a non-licensed operator’s failure to follow administrative procedures for controlling the status of equipment associated with the Standby Liquid Control System. The system’s purpose is to shutdown the reactor during an emergency by injecting a neutron-absorbing
Continued on the following page...solution into it via the core spray system. On June 12, 1996, the operator repositioned a breaker switch, resulting in the deenergization of heat tracing for an operable standby liquid control pump for 34 hours.” (NRC Press Release, July 23, 1997.)

July 30, 1996 - “...a containment isolation valve valve was opened and deactivated for 24 hours, rendering the valve inoperable. The valve had been deactivated for preventive maintenance work but without the proper actions taken to comply with the plant’s technical specification requirements. “The problem was significant because PP&L’s incorrect interpretation of requirements would have allowed the valve to remain inoperable and open indefinitely. A fine of $50,00 has been proposed for that alleged violation.” (NRC Press Release, July 23, 1993. (See July 23, 1993 for more complete date from the NRC.)

September 5, 1996 - The Company joined a consortium of electric utilities exploring the use of MOX, or weapons grade plutonium left over from the Cold War, as a fuel source.

October 1, 1996 - The Nuclear Regulatory Commission fined Thermal Sciences, Inc., $900,000 for "deliberately providing inaccurate or incomplete information to the NRC concerning TSI's fire endurance and ampacity testing programs." James Lieberman, NRC, Director of Enforcement.
The fine was the largest assessed against a nuclear contractor, and the second highest in NRC history. In 1992, the NRC declared TSI's fire barrier, Thermo-Lag, "inoperable." (For background data please refer to September 29, 1994 & April 15, 1995.)

November 5, 1996 - The Class 1E 4160 VAC Switch gear failed to pass seismic qualification testing at Unit-1 & Unit-2. PP&L reported an "outside design basis" (#31279) event. (See August, 1999, for more information.)- July, 1997 - The NRC "found that the load limit setting on
one of the [emergency diesel] generators had been positioned at approximately 35 percent, when it should have remained at 100 percent. The misalignment, which was subsequently determined to have occurred sometime between June 16 and July 11, could have resulted in the governor not starting within the required time and not being able to provide sufficient emergency backup power during an accident. Furthermore, the operation of the generator at a lower-than-normal speed could have damaged emergency core cooling system motors." (See January 12, 1998, for information on the NRC's enforcement actions.)

July 23, 1997 - "The Nuclear Regulatory Commission has proposed a $210,000 fine against Pennsylvania Power & Light Co. for several alleged violations of agency guidelines at the utility's Susquehanna nuclear power plant in Berwick, Pa. The alleged infractions fall into two major areas: the misalignment of a circuit breaker for an emergency diesel generator that left in operable, and plant operators' repeated failure to detect this problem; and the improper deactivation of a containment isolation valve:
"...All told, the generator was out of service for almost three weeks. However, in their equipment test records, the operators incorrectly reported that the circuit breaker was inn the appropriate position.
"Further, alarm tests that were supposed to have been done during rounds by the non-licensed operators were listed as

having been performed when in many cases that did not occur. The operators failed to perform the required panel tests on approximately 157 occasions between January and June 1996.

"Given the number of individuals involved, the actual and potential impact in equipment, the duration of the problem and the lack of management and supervisory oversight that resulted in the failure to detect this widespread condition, the NRC is classifying these alleged violations in the aggregate as a Severity Level II problem, which constitutes a very significant regulatory concern. ...Continued on the following page... "According to the NRC, "[t]his case represents particularly poor license performance, as evidenced by 1.) the nature of the violations associated with the Severity Level II problem, including the inoperability of the diesel generator for almost three weeks and the number of employees involved; 2.) the extensiveness of the problem with inaccurate records; and 3.) the management and supervisory failures demonstrated by these violations." (NRC Press Release, July 23, 1997.)
(See June 12, 1996 and July 30, 1996; April 8, 2004; and April 12, 2005 for other incidents cited in this violation.)

(Please reference the following dates for a list of chronic electrical problems at the SSES: June 25, 1983; "1986"; September, 1988; February 6, 1990; and, June 8-16, 1999.)

September, 1997 - "...Reported earnings for the quarter and year-to-date were influenced by several one time adjustments. First, a windfall profits tax in the United Kingdom based on PP&L Global's equity interest in a U.K. utility reduced earnings by about $40 million or 24 cents per share."
("Quarterly Review: PP&L Resources, Inc.", September 1997).
(Please refer to February 4, 2000, 2002: PPL kills expansion; earnings projections slashed and, April 26, 2003, for related developments).

October 22, 1997 - Unit-1 and Unit-2's suppression pools were identified as having the potential for bypass during a lossof-coolant-accident. PP&L reported an "outside design basis" (#33131) event. (See August, 1999, for more information.)

January 12, 1998 - "The Nuclear Regulatory Commission staff has proposed a $55,000 fine against the operator of the Susquehanna nuclear power plant for a violation of agency requirements involving a misaligned emergency diesel generator at the facility...
"In a letter to PP&L announcing the enforcement action,

NRC Region I Administrator Hubert J. Miller said that the failure caused ‘important safety-related equipment to be inoperable for an indeterminate period, thus degrading the plant’s capability to respond to accidents.

Continued on the following page... “Further, the NRC is concerned that you failed to implement effective controls for the alignment of the Woodward governor controls despite the fact that multiple events involving the functioning of the Woodward governors have been identified in the industry between 1985 and the present,’ including three at Susquehanna.”

Mr. Miller also noted that the “NRC is concerned that your investigation of the event could not preclude tampering as a cause and that the investigations revealed at least two other recent instances of unexplained misalignment of out-of-service EDG’s (emergency diesel generators) similar to the misalignment of the ‘A” EDG.” (NRC Press release, January 12, 1998.) (See July 11, 1997 for more on this incident.)

March 13, 1998 - “Earnings for 1997 were $296 million, or $1.80 per share of common stock, compared with $329 million, or $2.05 per share in 1996.” (PP&L Resources, Inc., A Common Sense Guide to Competition, 1997 Summary Annual Report.)

April 5, 1998 - Unit-2 was shut down manually due to a leak on the non-nuclear side of the water cooling system. (Lancaster Sunday News, April 5, 1998.)

May 15, 1998 - The PUC gave tentative approval, by a 5-0 vote, to a plan for PP&L’s restructuring that could save rate payers 10% on monthly bills. The Commission slashed the amount of stranded costs PP&L may recover to $2.864 billion. The company had sought $4.5 billion and PUC administrative law judge [Kashi] suggested $4 billion.” (“The Patriot News”, May 15, 1998.)

August 13, 1998 - The Pennsylvania Public Utility Commission adopted a tentative order approving PP&L’s restructuring case. Provisions include a 4% rate decrease for all customers in 1999, allows PP&L to recover $2.97 billion in “stranded expenses” over 11 years, and grants PP&L the opportunity to “securitize up to $2.97 billion in transition costs with 75% of the associated savings returned to rate payers. - September 4, 1998 - “Standard & Poor’s last week assigned its Triple B-plus rating to PP&L Inc.” (Dean Witter Reynolds Inc. and Standard & Poors Value Line, September 4, 1998.)

- 1998 - The Company was forced by the U.S. Department of Labor to rehire Donald Ranft, manager of the nuclear system engineering department. PP&L paid Mr. Ranft over $100,000 in back pay and legal fees. Mr. Ranft was forced out of his job after safety concerns he raised were not addressed. PP&L also pressured Mr. Ranft, a ten year veteran of the nuclear industry, not to report his safety concerns to the NRC. (See February 9, 1996, for a similar incident.
(See October 1, 1993, February 9, 1996 and 1998 for similar patterns of h a r a s s m e n t.)

December 27, 1998 - "For the 12 months that ended Sept. 30, PP&L reported a net loss of $3.51 a share, compared to earnings of $1.81 a share the year before." (Patriot News from Dean Witter Inc. and Standard & Poors Value Line.)
(See April 1999, for related development.)

February 1, 1999 - PP&L announced the arrival of dry storage casks designed by Trans Nuclear (Vectra) for spent fuel storage. The NRC approved the license and design of the casks scheduled to be operational by in the summer of 1999. Construction for this project resumed after a cessation of activity in fall 1998. PP&L has moved the scheduled operational date back to "late 1999." (PP&L, May 12, 1999.)
(See August 22, 1995 & August 5, 2002, for related events).

February 28, 1999 - The Company reported an "outside design basis" event (#35423) relating to a valve stem in the RHR. (See August, 1999, for more information. Refer to September 24, 1993, for a related incident).
- Mid-March until the end of April, 1999 - Extended refueling outage for Unit-2. However, the potential for problems with the main transformers were not discovered. (See June 7-8, 1999.) - April 1999 - "PP&L Resources reported a 1998 loss of $3.46 per share, reflecting $948 million of charges to net income related to the settlement of PP&L, Inc.'s restructuring case before the Pennsylvania Public Utility Commission and another other competition-related case before the Federal Energy Regulatory Commission." (PP&L Resources, Inc., Shareowner News.)
"The utility's dividend payout ratio was 64 percent on Dec. 31, 1998, compared with 82 percent on Dec. 31, 1997." (Patriot News from Dean Witter Reynolds Inc. and Standard and Poors Value Line.) (See December 27, 1998, for earlier announcement.)

March 13 to April 28, 1999 - Unit-2 was shut down for a planned refueling outage.

May 29-June 5, 1999 - Unit-1 was manually shut down. A change out celluloid valve in one of the steam lines was the root cause of the problem. Unit-1 was put back on-line from June 5-6, 1999.

June 7-8, 1999 - Unit-2 tripped due to a problem with one of the main transformers. PP&L plans to replace the troubled unit. (See "Summer 2000.")

June 8-16, 1999 - Unit-2 was shut down to replace "three main electrical transformers..." ("News Release(s)", PPL, June 8 & 16, 1999.)

(Please reference the following dates for a list of chronic electrical problems at the SSES: June 25, 1983; "1986"; September, 1988; February 6, 1990; and, July 23, 1997.)

June 28, 1999 - PP&L was assessed a $125,000 fine by the Attorney General relating to the Company's electric competition advertising and bill-stuffing. (See March 16, 1995, for related behavior).- July 1- 15, 1999 - Unit-1 was shut down automatically after one of the four main steam valves failed." The line carries steam from the reactor to the turbines..." ("News Release(s), PPL, July 1 & 15, 2000.) (Refer to July 12 to August 1, 1993, for related problems).

August, 1999 - "If a utility has operated the 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 reactor, the less certain that the reactor and its safety systems will operate as deigned." (James Riccio, Public Citizen, August 1999, Executive Summary.) (Refer to November 5, 1996; October 22, 1997; and, February 28, 1999.)

August 26, 1999 - Both Units were operating at 100% power, "with the 'B' loop of emergency service water (ESW) out of service for scheduled maintenance. During testing on the ESW system, with all ESW pumps in service, it was identified that the 'C' and 'D' ESW pumps' discharge check valves were closed. The ESW flow surveillance was performed, and the 'C' and 'D' ESW pumps failed to achieve the required flow and were declared inoperable. Concurrently, the 'B' loop of ESW was returned to

service.
"During the time the 'B' ESW loop was inoperable, the 'A' ESW pump was the only one operable ESW pump. This constitutes a serious degradation of the plant in that it is a condition which is outside of a design basis and, therefore, reportable...requiring a 1-hour notification." (PP&L facsimile.)

September 6, 1999 - PPL "planned to initiate the first fuel transfer to the storage location the week of September 6, 1999, but problems developed and the transfer has been delayed for a few weeks." (Office of Nuclear Reactor Regulation).- December 19-24, 1999 - Unit-2 was shutdown to make
"repairs [replace] to a pipe" connected to the "water pressure on a recirculation water pump". this system is part of the plant's primary containment structure. (News Release, PPL, December 24, 1999).
(See August 17-25, 2000, for a related problem at Unit-1).

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.)
- Winter 1999 - 2000 - PPL unilaterally devaluated the combined PURTA and Real Estate tax assessments for the SSES. Prior to the Negotiated Settlement, the nuclear power generating stations were assessed by PP&L at approximately $1 billion. PPL is now claiming that the the SSES is only worth $74 million or the same amount as the valuation of the Columbia Hospital. If PPL prevails, the Berwick School District and Luzerne County will experience revenue shock. PPL is not paying or escrowing any moneys they owe to Luzerne County and the Berwick School district.
(See April 23, 2001 and July 13, 2003, for related developments).

February 4, 2000 - "PP&L Capital Funding Inc.'s new $500 million 7 3/4% issue of medium term notes (MTN) due April 15, 2005 is rated /BBB+' by Fitch IBCA, Inc. PP&L Capital is a wholly-owned subsidiary of PP&L Resources, Inc. (Resources) and the funding conduit for Resources and its nonregulated subsidiaries, which invest in domestic and

international energy projects...Resources has investments and Continued on the following page...
commitments to invest about $2.6 billion in distribution, transmission and generation facilities in the US, UK, Bolivia, Peru, Argentina, Peru, Spain, Portugal, Chile, and El Salvador. Resources also plans to add about 8,000 megawatts (MW) of merchant generation over the next four to five years through acquisitions and/or new construction. The growing exposure to emerging markets and merchant generation will increase business risk." (PP&L, Company Press Release, February 4, 2000.)
(Please refer to September, 1997, 2002: PPL kills expansion; earnings projections slashed May 4, 2000, and March 4 & 18, September 23 & October 24, 2001, January 6, 2002, and April 26, 2003, for related de v e lopment s) .

May 4, 2000 - "One thing cushioning the blow to stockholders is GPU's annual dividend, raised this year to $2.18 a share. That is considerably higher than Allentown-based PPL Corp.'s dividend, which was raised last week to a $1.06 share. PPL stock is trading less than GPU shares." (Patriot News, Business, B9, May 5, 2000.)
(Please refer to February 4, 2000, and March 4 & 18, September 23 & October 24, 2001, for related developments).

May 5, 2000 - Unit-1 returned to service after a planned outage.

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.
(See January 22, 1994 and January through March, 2001, for PJM problems related to PPL. Refer to June 14 & October 19, 2002 and June 19, 2003 , for PPL's manipulation of the PJM grid).

"The action was taken to avoid emergency rolling blackouts where power is interrupted for short durations - typically 20 to 30 minutes." (Update, 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 this summer for consumers stretching from Virginia Beach to Detroit.
"The all-time maximum PJM demand of 51,700 MW occurred on July 6, 1999." (PECO Energy Company, Form 10-K/A, p.7). (Refer to June 14 & October 19, 2002, for PPL's manipulation

of the PJM grid).

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." (Irwin Popowsky, Consumer Advocate, Investor's Business Daily).
(See July 12 to August 1, 1993, January 1,1994, January 22,1994, July 1, 1994, April 15, 1995, Mid-March until the end of April, 1999, May 29-June 5, 1999, December 19-24, 1999, and August 17-25, 2000 for data relating to SSES's reliability. Refer to June 14, 2002, and June 19, 2003 for PPL's manipulation of the "Grid").

August 17-25, 2000 - Unit-2 was shut down to make repairs on a "small leak in the instrument line [inside the primary containment area]...on a large water pump". ("News Release(s)," PPL, August 17 & 25, 2000.) (See December 19-24, 1999, for a related problem at Unit-1.)

October 30, 2000 - PPL petitioned the NRC to increase the capacity of SSES by 100 megawatts. (See April 23, 2001, for follow-up.) - January through March, 2001 - PPL manipulated the Installed Capacity Market (ICAP) of the Pennsylvania-JerseyMaryland (PJM) Grid. PPL, identified as "E 1" in PJM and PUC investigations, manipulated the ICAP market during the first quarter of the 2001, but ICAP prices remain volatile. PPL's exercise of unilateral and documented abuses of its market power in the PJM capacity credit market during the first quarter of 2001 dramatically and artificially increased credit capacity markets to the economic detriment of Pennsylvania consumers.
(Refer to November 30, 2001, for a follow-up investigation.Also see June 14 & October 19, 2002, and June 19, 2003, for PPL's manipulation of the PJM grid.)

March 4, 2001 - "PPL stock was raised from 'hold' to "buy' by...Argus Research Corp." (See March 18, 2001, for a related development). (Sunday Patriot News, March 4, 2001).
(Please refer to February 4 & May 4, 2000, and March 18, September 23 & October 24, 2001, and January 6, 2002, for related developments).

March 18, 2001- "PPL stock was downgraded from 'strong buy’ to ‘buy’ by analyst Paul Patterson at Credit Suisse First Boston.” (See March 4, 2001, for a related development) (Sunday Patriot News, Business, March 18, 2001). (Please refer to February 4 & May 4, 2000, September 23 & October 24, 2001, and January 6, 2002, for related developments).

April 23, 2001 - PPL announced it would petition the NRC to increase the capacity of SSES by 100 megawatts, while decreasing the properly value of the plant. “The $120 million of improvements at the Susquehanna plant are expected to add to earnings as soon as they go into operation” (Reuters, April 23, 2001).(Please refer to Winter 1999 - Winter 2000, for background information).
(Please see July 17, 2001, for follow-up data.) - July 17, 2001 - The NRC approved PPL’s capacity expansion request. Unit 1 will be increased this month while the upgrade at Unit 2 is planned for Spring, 2002, after the planned refueling outage. (See October 30, 2000 & April 23, 2001, for background information).

August 23, 2001 - An “unusual event” was declared “after plant security apprehended a man inside a vehicle access area at one of the plant’s gates.” The man was not armed, but scaled one security fence. (PPL Susquehanna LLC, Press Release, August 23, 2001).

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, PPL’s “voice in Washington, “recommended” that the Petition be “denied.”

September 23, 2001 - After trading resumed on September 17, 2001, PPL closed down -$5.10 at $37.00 ABN Amro rated the stock as “hold” and the “target price range is $49 to $50. a share.” (“Sunday Patriot News”, Business, September 23, 2001. (Please refer to February 4 & May 4, 2000, and March 4 & 18, October 24, 2001, and January 6, 2002, for related developments).

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, for a related incident.) - 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, for a related event.)
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 24, 2001 - Wachovia downgraded PPL Resources from "strong buy" to "market perform." (Also see March 18, & September 23, 2001.)
(Please refer to February 4 & May 4, 2000, and March 4, and October 24, 2001, and January 6, 2002 for related developments).

November, 2001 - PPL filed a pre-notification letter with the NRC announcing plans to extend Susquehanna's operating licensees for Units 1 & 2. To date, the NRC has approved every license extension before the agency. A similar affirmation at the SSES would extend the license for Unit-1 from 2022 to 2042 and Unit-2 from 2024 to 2044.- 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 detachments deployed to protect nuclear facilities against terrorist attacks (See October 6 & 17, 2001, January 30, 2002, and May 22, 2003 for related incidents).

November 30, 2001 - The PUC ordered an Investigation into PJM's ICAP market manipulation. (See January to March, 2001, for data relating to ICAP market manipulation. See December 6, 2001, for "market response", and PUC follow-up on June 16, 2002. Also, refer to January 6, 2002 & October 19, 2002, for plant cancellations and a revised earnings forecast.)

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.)

Earlier, on 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.]

(Please refer to February 4 & May 4, 2000, and March 4 & 18, September 23 & October 24, 2001, for related developments).

PPL's stock fell by 3% due to events surrounding PPL's ICAP market manipulation.

(See January to March, 2001, for data relating to ICAP market manipulation. Also, please refer to November 30, 2001, January 6, 2002 and June 19, 2003)

January 6, 2002 - "PPL lowered its 2002 earnings forecast a second time and canceled plans for six new power plants, citing a continuing drop in wholesale energy profit margins and fallout form the Enron bankruptcy." (Sunday Patriot News, Business, January 6, 2002).

PPL's stock closed at $32.34 on Friday, January 4, 2002.

Its 52-week high was $62.36. (Please refer to November 30 and December 1, 2001, for related developments.)

(Please refer to February 4 & May 4, 2000, and March 4 & 18, September 23 & October 24, 2001, for related developments).

2002: PPL kills expansion; earnings projections slashed

Citing Enron Corp.'s bankruptcy and plans to cancel construction of six new power plants, PPL Corp. slashed its earnings forecasts for 2001 and 2002. In a filing with the Securities and Exchange Commission, the Allentown, Pa.-based utility said it's scaling back its generation-expansion program as a result of continuing declines in wholesale energy prices. PPL previously announced plans to develop an additional 4,605 megawatts of generating capacity. It cut projects that would have produced 2,100 megawatts of power. (One megawatt heats about 600 homes.) Though PPL said it still sees a need for new generating capacity, market prices and regulatory conditions deterred it from building six new power plants, five in Pennsylvania and one in Washington state. The cancellations of $1.3 billion worth of projects will cause PPL to take its biggest charge in its 2001 earnings. In addition, Houston-based Enron's bankruptcy filing caused some PPL subsidiaries to end electricity and gas agreements. PPL now expects 2001 earnings per share of $3.35 to $3.45, down from an initial projection of more than $4 per share, with flat growth for 2001.

Market researcher Thomson Financial/First Call had released a consensus estimate of $4.13 for 2001 and $4.16 for 2002. PPL's earnings estimate includes a 60-cent charge for canceling its order of 22 turbines from General Electric Co. for the nixed power plants. PPL's revised estimate also includes a 14-cent charge from the Enron-related write-off of Western Power Distribution, its United Kingdom affiliate, and a 6-cent charge from other Enron-related items. PPL had a 51 percent interest in Western Power. Brazil's drought and poor economic conditions also will hurt the earnings from PPL's Latin American operations, the company said.

In addition, a change in the accounting rules for goodwill could hurt PPL's earnings, though the company said it can't yet quantify such an impact, if any.

(Refer to 1996 for a related development.)

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 29, 2002 - PPL notified the Nuclear Regulatory
Commission (NRC) that it intended to file for renewal of the operating licenses for SSES Units 2 and 3. If approved, Unit' 1's license would be extended from 2022 and Unit 3's from 2024 for an additional 20 year period.

The Nuclear Regulatory Commission is expected to take two years to review the license renewal application. 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.

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.)

March 28, 2002 - The NRC admitted that and the the SSES and the nation's 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 6 & 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 October 6, 2001 & October 17, January, 9 and 30, 2002 for related incidents.)

April 29, 2002 - At PPL's annual shareholder meeting, Bill Hecht told the audience the Company is “agile and robust” and predicted above average earnings. Hecht noted that he was navigating PPL through the most volatile period in the history of the electric industry.” (Restructuring Today, April 29, 2002.

May 5, 2002 - PPL stock was rated ‘hold’ by UBS Warburg. ’ (Sunday Patriot News, May 5, 2005).

May 8, 2002 - The NRC found PPL’s emergency preparedness plan for the SSES lacked adequate staffing. In 2001 the Commission documented under staffing on several different occasions. PPL submitted a compliance plan on May 13, 2002. (Philadelphia Inquirer, May 8, 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.)

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

June 14, 2002 - The Pennsylvania Public Utility Commission accused PPL of gaming the capacity market in the PJM grid in early 2001, but asked state regulators and federal authorities to investigate.
“The Pennsylvania PUC has evidence that allegedly shows PPL withheld electricity to create an artificial power shortage in the market for extra capacity where utilities buy credits to meet PJM reserve requirements.
“Such alleged activity drove up prices when the capacity price shot up from $5/mwh to a $177/mwh on average for more than three months.” PPL denies the charges. (“Restructuring Today”, Friday June 14, 2002.) Refer to January through March, 2001 background information, and further October 19, 2002, for additional legal action. Also, see January 22, 1994, for PJM-related problems. Refer to June 19, 2003, for results from the PA AG’s investigation.)

June 17, 2002 - PPL traveled to Wall Street to assure investors the Company “has long-standing policies to ensure that, across the company, the actions of our marketing operation are ethical and legal, John R. Biggar, CFO, (Philadelphia Inquirer, June 18, 2002.)- June 19, 2002 - PPL cut its work force by 7%. On June 1, 2002, “Public Utilities Fortnightly” published a list of highest paid electric CEO’s. PPL’s William Hecht was ranked 31 at $1,197,500. (See “From, 1985 - 1994” and November 14, 1995, for more on job cuts.)

August 5, 2002 - The NRC issued a Severity III Violation for a “mix- up of gases in a spent fuel storage cask at Susquehanna last summer, and the company said it would not contest finding...”, and pay the $15,000 base civil penalty. PPL spokesman Herbert Woodeshick said: “We have cooperated with the NRC throughout its investigation of this matter, and we respect the commission's decision in determining that the incident constituted a level III violation” (Nuclear Fuel, February 3, 2003).
(See also August 22, 1995 and February 1, 1999).

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 9 , 2002 - Standard & Poor’s downgraded PPL’s rating.

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.)

October 3, 2002 BERWICK, Pa. (AP) - A fire broke out early Thursday at PPL's Susquehanna nuclear power plant and was quickly put out, officials said.
The fire, detected at around 2:30 a.m., was confined to a startup transformer on Unit 2, according to a company news release. An automatic system extinguished the flames, and the transformer will be replaced with a spare on site, PPL said.
Continued on the following page...
The fire apparently was caused by an internal failure, company spokesman Herbert Woodeshick said. He could not give a monetary estimate of the damage.
The incident was classified as an "unusual event," the least serious of four federal classifications of power plant emergencies.
PPL Corp. is a global energy company based in Allentown. The plant is in east-central Pennsylvania. (http://www.pplweb.com)

October 19 , 2002 - Fourteen boroughs brought suit against PPL

for alleged market manipulation. The boroughs include: Blakely, Catawissa, Duncannon, Haven, Kutztown, Landsdale, Lehighton, Mifflinburg, Olyphant, Perkasie, Quakerton, Saint Claire, Schuylkill, and Watsontown.
(See January 22, 1994 and January through March, 2001, for PJM problems related to PPL. Refer to September 9 & June 14, 2002, and June 19, 2003 for PPL's manipulation of the PJM grid).

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.)
(See October 6 & 17, 2001, January 30, 2002, November 2, 2002 and May 22, 2003 for related incidents).

December 13, 2002 - "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 13, 2002 - A security challenge occurred at the SSES nuclear facility 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.

"At 1158 EST a delivery truck at the owner controlled entrance gate was identified to have a saddle tank leak which resulted in a spill of approximately 10 gallons. The diesel fuel was contained by site personnel, and is in the process of being cleaned by site personnel. None of the oil reached a waterway, and therefore does not meet the requirements for a reportable spill. The delivery company contacted their contracted spill response team, and they responded to the site. They were subsequently released without performing any of the cleanup activities. The minor spill was not due to sabotage. This information has been provided to PEMA.
"This report is being issued due to the involvement of other government agencies, and reportable under 10CFR50.72(b)(2)(xi)." (US NRC).

January 29, 2003 -An Unusual Event was declared due to an airborne release containing Cesium-138. An hour later, monitor readings returned to normal. (See March 4 and , 2003 for related radioactive events.)

February 23, 2003 "PPL Corp. stock is rated "overweight/neutral" in new coverage by Daniel F. Ford at Lehman Brothers. The target price is $39 a share."

February 29, 2003 - "PPL reported 2002 earnings from core operations of $3.54 a share, compared with $4.22 a share in 2001. "Sunday Patriot News").

Radioactivity found on two GE workers at Pa. nuke

March 4, 2003 " Two contract employees reported to the Susquehanna nuclear power plant in Pennsylvania with low levels of radioactive material on their clothing, owner PPL Corp. ((PPL.N)) said on Tuesday. Highly sensitive monitoring equipment at the plant detected the radioactivity on Monday as the General Electric Co. ((GE.N)) contractors were leaving an area inside a security fence, the company said in a statement.

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"The radioactive material is believed to have originated at another facility, and not at Susquehanna, the company said, and the level of radioactivity was very low. This type of event is rare but not unheard of at the nation's nuclear power reactors. But since the Sept. 11, 2001, attacks, all incidents of possible public exposure to radioactive materials receives increased scrutiny. PPL plant personnel began investigating and conducting additional radiological surveys immediately, said Joe Scopelliti, spokesman for the Susquehanna plant.

" 'At no time was the health and safety of the contractors, other Susquehanna workers or the general public affected because of this incident,' " Scopelliti said in a statement. "'The level of radioactivity on the clothing was slightly above what is seen in background radiation in the environment.'"

The contractors' previous job was at a nuclear power plant in Sweden, PPL said in its statement. Monday was their first day inside Susquehanna's security fence, however neither contractor had entered the part of the plant that contains radioactive materials, Scopelliti said. Routine radiological surveys found the areas outside that part to be free of radioactivity, PPL said. General Electric said it also was investigating. Federal regulators and state environmental officials have been notified, the company said. (See January 29, 2003 and March 25, for other releases.)

March 23, 2003 - "PPL is replacing all four steam turbines at its

Susquehanna nuclear plant near Berwick" ("Sunday Patriot News", March 23, 2003).

March 25, 2003 - An "unusual event" was declared when "contamination was taken off site" when "a worker "tripped on lead shielding blankets..." The event was "declared at 4:52 pm and ended at 7:15 pm. " (Platts Nuclear News Flashes, March 25, 2003)
(See January 29 and March 4, 2003, for related incidents).

April 26, 2003 - PPL defended its $314 million investment loss in a Brazilian electric distribution company, and plans to maintain its investments in similar companies located in El Salvador in the United Kingdom (Please refer to September, 1997, February 4, 2000, and 2002: PPL kills expansion; earnings projections slashed and for related developments).
Despite management's objections, shareholders approved a resolution that "recommended" the submittal of "poison pills " to shareholders for approval. "Two other shareholder resolutions failed. One would have set limits on bonuses for PPL executives, and the other would have required that the accounting firm that does the annual PPL audit not get other business from the company" (April 26, 2003).

May 16, 2003 - PPL issued a press release indicating that they will be filing a distribution rate case at the PUC in the Spring of 2004 with proposed new rates to take effect on January 1, 2005. The press release does not specify the anticipated amount of the increase. PPL's transmission and distribution rate cap expires on December 31, 2004. Company representatives previously had informally indicated that they would file in 2004.

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. 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).

June 19, 2003 - The Attorney General rejected the PUC's claim that

PPL manipulated whole sale electricity prices between January and April, 2001. Although prices spikes 3o times above normal seasonal rates, the AG "determined that that PPL did not violate antitrust in acquiring that market power." The Attorney General did admit held extra capacity in 2001. FERC did not act as is satisfied with subsequent PJM rule changes will prevent future spikes. However, as result of the price gauging several smaller electric retailers were permanently forced out of the market (See June 14, 2002, for background information).

June 29, 2003 - "More than 50 Montana residents have sued PPL Corp., alleging that the Colstrip power plant PPL operates and partially owns in Montana is polluting their drinking water. PPL says there is 'no merit' to the claim" (Sunday Patriot News, June 29, 2003).

July 13, 2003 - "Utilities save big as towns lose out: Tax bills on plants of major power companies in Pennsylvania have gone from $120 million annually to $20 million (Anthony R. Wood, Inquirer Staff Writer)

While homeowners are paying an average of 30 percent more than they did in 1997, Exelon, Pennsylvania Power & Light, and the other major electric utility companies in the state are paying 85 percent less in taxes on their plants, down from about $120 million annually to about $20 million, an Inquirer analysis has found.
Meantime, the utilities are passing on their real estate levies to their customers, based not on what the companies are currently taxed but on the far higher sums of six years ago....For the previous 25 years, the power companies' property taxes
were relatively cut-and-dried. Payments were calculated by the state and put into one important pot: the Pennsylvania Utility Realty Tax Act fund, or PURTA. For 1997, $167.5 million was paid in, the bulk of it by the two electric behemoths, Peco Energy Co. and Pennsylvania Power & Light.
...When the state loosened its grip on the electric industry, the commercial power plants - 25 major ones, 55 much smaller - were gradually released from PURTA. For the first two years, 1998 and 1999, the utilities were allowed to appraise their plants for tax purposes; the fund tumbled to $60 million.
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....On Jan. 1, 2000, the plants were removed from PURTA and put on the property rolls of the locales in which they sat, to be assessed and taxed like any hometown business.
....Susquehanna nuclear power plant. Although the facility was built at a cost of $4 billion and assessed at $3.8 billion, PP&L argued in its appeal that it was worth only a fraction of that. In December 2000, a Luzerne County judge agreed, fixing the assessment at $165.4 million.

PP&L now pays $3 million annually to the county, Salem Township and the Berwick Area School District - far less than the $30 million the plant used to add each year to the PURTA pot, according to court records.The Susquehanna appeal has been by far the biggest in the state. The Common Pleas Court ruling, which paralleled PP&L's arguments virtually point for point, could set the course for other cases in Pennsylvania and around the nation, said Epstein, the consumer activist.

"[Susquehanna] was the first nuke case to come in, and it was precedent-setting," Epstein said. Since then, he added, the strategy "of driving school districts off a cliff without a seat belt" has been applied in cases around the commonwealth.

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...From 2000 through 2009, PP&L is including in its customer billings $280 million in real estate levies, according to court records. In reality, the company pays only $3 million a year on the plant - an estimated 10-year windfall of $250 million.

Study Finds Utility Winners During Deregulation Are Companies That 'Stuck to Their Knitting'

August 4, 2003 - "From 1998 to 2002, U.S. utilities leapt into deregulation and created multiple strategies to compete. Because it takes time to determine how the strategies worked, we are just seeing results now. Winners among utility companies relied on traditional regulated utility assets," said Coyne and Hartshorne. "They are firms that stuck to their knitting rather than plunging into merchant power generation or purchasing foreign power plants.

" The top five companies in annualized shareholder return were Exelon Corp., Southern Company, Entergy Corp., Western Gas Resources and PPL Corp.

" The bottom five companies in total shareholder return for the fiveyear period were Aquila Inc., Dynegy Inc., The Williams Companies, Inc., The AES Corp. and El Paso Corp."

August 6, 2003 -The NRC released NUREG 1774 which documented a 60% increase in fuel load drop events from 1993 to 2002. The Report found half of the incidents involved moving fuel assemblies at spent fuel pools, and greater risks for heavy load drops were at Boiling Water Reactors like Susquehanna (The Report #ML033060160 can be accessed through ADAMs.) (For related events at he SSES please refer to December 31, 1992; September 10 and October 1 & 28, 1993; January 1, July 1 and August 1994; August 22, 1995; and, September 5, 1996.)

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 threequarters 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.

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.)

August 31, 2003 - "In the first half of the year, PPL posted earnings from core operations of $292 million, or $1.72 a share, compared with $262 million, or $1.77 a share, during the same period in 2002" (Sunday Patriot News, August 31, 2003).

September 11, 2003 - SUSQUEHANNA-1 WAS AT ABOUT 65% POWER TODAY AFTER A FIRE ON A FEEDWATEpump was extinguished last night. Joe Scopelliti, a spokesman for operatoPPL Susquehanna, said today that plant personnel were investigating the cause of the oil fire, which the plant fire brigade extinguished eight minutes after it started at 11:14 last night. He declined to estimate when the unit would return to full power. (Platts Nuclear News Flashes). - September 15, 2003 - SUSQUEHANNA-1 RETURNED TO FULL POWER THIS MORNING following repairs to one of the three pumps that provides water to the reactor. The repairs were necessary following a fire on Sept. 10 that was caused by a leak in a pump lubrication system. The unit was at 70% power following the fire. Herbert Woodeshick, spokesman for operator PPL Susquehanna, said the investigation into the root cause of the leak is still ongoing (Platts Nuclear News Flashes).

September 19, 2003 - "Critics say that the high electric prices and the subsequent failure of Montana Power were evident from the start: Montana Power, which once provided the sixth lowest electric rates in the country, consented to sell off its generating plants as part of the deal to allow it to diversify into unregulated businesses. But, one buyer PP&L came in and bought all the assets. So, instead of having a steady supply from one, regulated in-state supplier, there is now one, unregulated out-ofstate supplier...

"Concerns that rates may rise even higher have prompted a voter initiative in Montana to give the state the right to buy back the assets that were sold to PP&L. That vote failed in 2002, although supporters say that they will try again in 2004... (By Ken Silvestein , Director, Energy Industry Analysis). (See June 29, 2003, for related information). Power Reactor Event Number: 40196 Facility: SUSQUEHANNA

Event Text:

AUTOMATIC SCRAM AT SUSQUEHANNA ON LOW WATER LEVEL - "At 0053 hours on September 24, 2003 with Susquehanna Unit 1 operating at 100% power an automatic reactor scram occurred due to low water level. At the time of the scram, reactor feed pump testing was in progress and the 'C' reactor feed pump tripped. The reactor recirc pumps runback initiated as expected when water level reached 30" with the feed pump tripped. Level continued to drop and reached the Level 3 auto scram setpoint. Level continued to drop and reached a low level of approximately -48" wide range. Reactor Core Isolation Cooling and High Pressure Coolant Injection auto started at their initiation setpoints and injected to the vessel to recover level. All level 2 and 3 containment isolations occurred as expected. The reactor recirc pumps tripped as expected when level 2 was reached. Reactor Pressure was controlled with bypass valves, there were no Safety Relief Valve lifts. There are no challenges to containment.

"Unit 1 is currently stable in Mode 3 with both reactor recirc pumps restarted. A human performance error was the cause of the reactor feed pump trip. Investigation is continuing into the plant response to the reactor feed pump trip."

The NRC Resident Inspector was notified of this event. - NEW YORK, Sept 24 (Reuters) - PPL Corp. said on Wednesday that a unit at its Susquehanna nuclear power plant automatically shut down when one of three feedwater pumps that supply water to its reactor stopped working. The loss of the feedwater pump caused the water level in the Unit 1 reactor to drop, causing a full shutdown of the unit at 12:53 a.m. The plant is located in Luzerne County near..." (See November 13, 2003 for follow up inforamtion.)

The goal is for nuclear power plants to have

24-hour Coast Guard patrols

By SEAN ADKINS Daily Record staff Friday,

October 10, 2003 - The U.S. Coast Guard has proposed a permanent rule that would close off sections of the Susquehanna River adjacent to Three Mile Island and Peach Bottom Atomic Power Station.

Following the terrorist attacks, the Coast Guard began patrolling temporary circular security zones around the waters that both nuclear power plants use for producing electricity.

The temporary zones act as a barrier to vessel traffic in a specific areas and work to protect power plants from damage or terrorist attack, according to a public notice published in the Federal Register.

The proposed rule is part of a national plan to switch the status of the temporary zones to that of permanent, said Neil Sheehan, spokesman for the Nuclear Regulatory Commission. The goal of that national strategy is for each of the country's 68 nuclear power plants eventually to be subject to immutable 24-hour patrols by the Coast Guard with assistance from other federal state and local agencies, he said. "The concern here is to protect the critical and vital areas of the plant," Sheehan said.

Similar to the present temporary conditions, the permanent law would prevent people and boats from entering or lingering in the security zone without prior authorization. Pending public comment that could alter the rule, plans are for the temporary zones that surround Peach Bottom Atomic Power Station and TMI in Dauphin County to become permanent by early next year. The change in zone designation from temporary to permanent will not affect plant operations, said Dana Fallano, a spokeswoman for Exelon Generation. The company worked with both the NRC and the Coast Guard to establish the zone, she said. Exelon co-owns and operates both TMI and Peach Bottom Atomic Power Station.

The security zone is not expected to disrupt charter and recreational fishermen, since those boats will be allowed to pass safely around the area, according to the public notice.

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).- NUCLEAR NEWS FLASHES - Wednesday,

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 13 , 2003 - " Pennsylvania Power & Light's
Susquehanna-1 was forced [to] shut down 159 hours due to low reactor water level following an indervtent trip of a feed pump during feed pump testing" (Nucleoniocs Week, p. 17.) (See September 24 2003, for initiating event.)

Nuke fund falls short of target, report says
Owners required to set aside money to dismantle plants

December 05, 2003- BY GARRY LENTON, The Patriot-News
The owners of a third of the nation's nuclear plants, including the damaged reactor at Three Mile Island, aren't setting aside enough money to dismantle the plants when they close, according to a new federal study. That could mean higher electric rates for some Pennsylvanians if
companies increase their annual contributions to catch up.
If the companies don't close the shortfall, the study warns, taxpayers may face billions in cleanup costs when the plants' useful lives are ended, most likely decades from now...
The total decommissioning bill for all existing plants is estimated to be $33 billion.
The lifetime of a nuclear power plant is estimated to be 40-60 years.
At that age, industry experts say, facility wear and fatigue can make continued operation unsafe. The plants are licensed by the federal Nuclear Regulatory Commission for 40 years, with the opportunity to apply for extensions.

Continued on the following page...

Under federal law, decommissioned plants must be dismantled and the land returned to pristine condition.
Pennsylvania plants that are under-funded, according to the GAO report, are Limerick 1 and 2 in Montgomery County; Peach Bottom 1 in

York County; Three Mile Island 2, and Susquehanna 1 and 2 near Berwick. Both the GAO and the NRC projections could be wrong. No one knows for sure how much it will cost to decommission a nuclear plant, because it has not been done. "Estimates are based on the volume of materials that would have to be shipped and stored," Exelon's Nesbitt said. "... Nobody really knows [what the cost will be.] You base it on the best data you have available."

Eric Epstein, president of Three Mile Island Alert, and founder of the EFMR Monitoring Fund, who has helped negotiate cost-recovery plans for nuclear plants before the Pennsylvania Utilities Commission, estimates that the industry is billions short of what will be needed.
Estimates are based on plans that assume that low-level nuclear waste from Pennsylvania plants will be shipped to a dump site that doesn't exist, Epstein said. The estimate also assumes there will be a place to store the spent fuel rods and other high-level wastes. The federal government has yet to build such a site.

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.)
Facility: SUSQUEHANNA
HQ OPS Officer: GERRY WAIG Notification Date: 01/15/2004
Notification Time: 13:03 [ET]
Event Date: 01/14/2004 Event Time: 19:50 [EST]
Last Update Date: 01/15/2004
Emergency Class: NON EMERGENCY
1 N Y 9 4 Power Operation 9 4 Power Operation
2 N Y 1 0 0 Power Operation 1 0 0 Power Operation
Event Text
OFFSITE NOTIFICATION OF ACCIDENT INVOLVING 2 TRUCKS CARRYING EMPTY NEW FUEL SHIPPING CONTAINERS
- The following information was provided by the licensee via facsimile: "On 1/14/2004 at 19:56 hours, the Shift Manager was notified by the Clinton County, PA Emergency Management Agency of vehicle accident involving trucks that were carrying a shipment from Susquehanna. The

trucks were carrying empty shipping boxes from a shipment of new fuel
that had previously been delivered to Susquehanna. These empty boxes
were being shipped in accordance with U.S. Department of Transportation
regulations [49CFR173.428 Empty Class 7 (Rad Mat)].
"On 1/15/2004 at 10:20 hours, additional information was provided to the control room indicating that this accident could cause increased
public interest due to the severity of the accident. The two tractor trailers
involved in the shipment were amongst the vehicles in the accident. One of
the truck drivers was seriously injured. The trucks were severely damaged.
Clinton County, PA, Emergency Management Agency was called to the
scene by initial responders as well as the Pennsylvania Department of
Environmental Protection. Both surveyed the boxes and found no
indication of radiation/contamination. The shipping boxes and vehicles are
being held by the towing company until the shipping company can provide
replacement vehicles."
The licensee has notified the NRC Resident Inspector.
(See March 6, 2004, for a similar accident.)

Jan. 18, 2004- Power Reactor Event Number: 40486
Facility: SUSQUEHANNA
Region: 1 State: PA
Unit: [ ] [2] [ ]
RX Type: [1] GE-4,[2] GE-4
NRC Notified By: GORDON ROBINSON
HQ OPS Officer: STEVE SANDIN Notification Date: 01/29/2004
Notification Time: 00:05 [ET]
Event Date: 01/28/2004
Event Time: 20:33 [EST]
Last Update Date: 01/29/2004
Emergency Class: NON EMERGENCY
10 CFR Section:
50.72(b)(2)(xi) - OFFSITE NOTIFICATION Person (Organization):
GLENN MEYER (R1)
U n i t SCRAM Code RX CRIT Initial PWR Initial RX Mode Current
PWR Current RX Mode
2 N Y 1 0 0 Power Operation 1 0 0 Power Operation
Event Text
OFFSITE NOTIFICATION TO LOCAL LAW ENFORCEMENT DUE TO FIRE
BRIGADE ACTIVATION
"At 2018 hrs, the Control Room was notified of smoke coming from the Unit 2
Vital UPS room. The Field Unit Supervisor (FUS) was dispatched to the room to
investigate. At 2026 hrs, the Fire Brigade was activated. When the FUS arrived at
the Vital UPS Panel he reported that there was smoke coming from the panel.
He opened the panel and observed smoke coming from the transformer in the
panel. He did not observe any flames at any time while dealing with the event.
At 2029 hrs, Security was notified and subsequently notified the State Police at

2033 hrs. At 2033 hrs, the transformer was deenergized and the smoke began to dissipate. Entry into the Emergency Plan was evaluated and it was determined that no entry conditions exists at this time.
"Due to the notification of the Local Law Enforcement Agency, this event constitutes an Offsite Notification and therefore reportable under 10CFR50.72(b)(2)(xi) requiring a 4 hr ENS notification."
When the transformer was deenergized, all loads were automatically transferred to the alternate power supply. The loss of this transformer did not affect any safety related equipment and does not require entry into any TS LCO Action Statements.
The licensee notified state/local agencies and the NRC Resident Inspector. No press release is planned.
Power Reactor Event Number: 40498
Facility: SUSQUEHANNA Region: 1 State: PA
Unit: [1] [2] [ ]
RX Type: [1] GE-4,[2] GE-4 NRC Notified By: GRANT FERNSLER
HQ OPS Officer: MIKE RIPLEY Notification Date: 02/02/2004
Notification Time: 17:33 [ET]

February 2, 2004
Event Time: 09:01 [EST]
Emergency Class: NON EMERGENCY
Unit SCRAM Code RX CRIT Initial PWR Initial RX Mode
Current PWR Current RX Mode
1 N Y 87 Power Operation 87 Power Operation
2 N Y 1 0 0 Power Operation 1 0 0 Power Operation
Event Text
FITNESS FOR DUTY
A contractor foreman/supervisor was determined to be under the influence of alcohol during a pre-access FFD test as part of processing for unescorted access. The supervisor was denied unescorted access to the protected area. Contact the HOO for additional details The licensee notified the NRC Resident Inspector.
NRC: NRC Special Inspection Starts at Susquehanna Nuclear Plant
News Release - Region I - 2004-00
U.S. NUCLEAR REGULATORY COMMISSION
Office of Public Affairs, Region I No. I-04-003

February 9, 2004
CONTACT: Diane Screnci (610) 337-5330
Neil A. Sheehan (610) 337-5331
Several events involving loose bolts on emergency diesel generators.
The twin-reactor plant is located in Berwick, Pa., and operated by PPL Susquehanna, LLC.
The purpose of the inspection, which got under way today, is to determine the facts surrounding the discovery that several bolts on

emergency diesel generators at the plant were found to be not fully
tightened during the period from March 2003 through January. Among
other things, the team will independently evaluate the adequacy and
quality of PPLs response and the risk
significance of the problem.
Nuclear power plants generate and transmit electricity to the grid,
but they also receive power back for operational purposes. If the flow of
that off-site power is interrupted, emergency diesel generators are relied upon to power
key safety systems and safely shut down the plant. As such,
their proper functioning is of vital importance to plant safety. The
Susquehanna plant has five emergency diesel generators.
In March 2003, a bolt on a linkage that controls the diesel fuel supply
for one of the plants emergency diesel generators fell off during routine
testing, forcing the engines shutdown. On January 25 -- again during
routine testing -- PPL found the mounting bolts for the governor, or
control, on another emergency diesel generator were not fully tightened. In
addition, workers observed oil leaking from under the control. That engine
also had to be shut down during testing due to the problems. Subsequently,
PPL on January 30 identified several bolts that were not fully tightened on
a lube oil cooler, or heat exchanger, for a third emergency diesel generator.
The three-member NRC team will document its findings in an
inspection report that will be issued no more than 45 days after the exit
meeting for the review.
Last revised Tuesday, February 10, 2004

February 28, 2004 - SSES shut down for refueling and maintenance.
Power Reactor Event Number: 40571 Facility: SUSQUEHANNA
Region: 1 State: PA Unit: [1] [2] [ ]
NRC Notified By: GRANT FERNSIER
HQ OPS Officer: RICH LAURA Notification Date: 03/06/2004
Notification Time: 08:20 [ET] Event Date: 03/06/2004
Event Time: 05:28 [EST] Last Update Date: 03/06/2004
Emergency Class: NON EMERGENCY
OFFSITE NOTIFICATION Person (Organization):
MOHAMED SHANBAKY (R1)
U n i t SCRAM Code RX CRIT Initial PWR Initial RX Mode Current
PWR Current RX Mode
1 N N 0 Re fue l ing 0 Re fue l ing
2 N Y 1 0 0 Power Operation 1 0 0 Power Operation

AREVA Awarded Contract to Supply Fuel for PPL Susquehanna
3/8/2004 Bethesda, Md. -- AREVA`s joint subsidiary with Siemens,
Framatome ANP, has been awarded a contract to supply six batches of nuclear
fuel for PPL`s Susquehanna nuclear power plant. Delivery of the first reload
under this contract will be in early 2005.AREVA will supply its ATRIUM™ 10
boiling water reactor (BWR) fuel

assemblies for Susquehanna units 1 and 2. The fuel will be manufactured at AREVA`s nuclear fuel manufacturing facility in Richland, Washington. Since 1992, more than 2,900 ATRIUM™ 10 fuel assemblies have been installed in 17 reactors worldwide.
"We have enjoyed a longstanding relationship with PPL Susquehanna," said John Matheson, AREVA senior vice president, nuclear fuels. "We are pleased to have this opportunity to further support PPL`s generation goals by providing high-quality fuel that is capable of meeting the highest demands for performance and reliability." (Press Release).

Event Text
OFFSITE NOTIFICATION AT SUSQUEHANNA INVOLVING
A TRAFFIC ACCIDENT
"On 3/06/04 at 0528 Plant Security was notified of an accident at the entrance to the site involving an employee leaving work and a south bound vehicle on PA Route 11. There were no reported injuries. Local law enforcement was contacted and investigated the incident. Because of the involvement of a LLEA and potential media or general public interest in the event, the Pennsylvania Emergency Management Agency (PEMA) was notified of the incident at 0812 hours. Based on the notification to a government agency and possible public interest, this event was determined to be reportable under 10CFR50.72(b)(2)(xi)."
The NRC Resident Inspector was notified.
(See January 14, 2004, for a similar accident.)Power Reactor Event Number: 40602
Facility: SUSQUEHANNA
Region: 1 State: PA Unit: [1] [2] [ ] RX Type: [1] GE-4,[2] GE-4
NRC Notified By: RONALD FRY HQ OPS Officer: CHAUNCEY GOULD
Notification Date: 03/21/2004 Notification Time: 16:03 [ET] Event Date:

March 24, 2004
Event Time: 12:32 [EST] Last Update Date: 03/21/2004
Emergency Class: NON EMERGENCY 10 CFR Section:
INFORMATION ONLY Person (Organization): HAROLD GRAY (R1)
U n i t SCRAM Code RX CRIT Initial PWR Initial RX Mode Current
PWR Current RX Mode
1 N N 0 Re fue l ing 0 Re fue l ing
2 N Y 1 0 0 Power Operation 1 0 0 Power Operation
Event Text
THREE INJURED NONCONTAMINATED CONTRACTORS WERE TRANSPORTED
TO THE HOSPITAL.
"On 3/21/04 at 12:32 hrs a bucket truck working at the Unit 1 Cooling Tower came in contact with a 230KV transmission line causing the loss of one off site power supply to the plant. The 500 KV offsite circuit remained energized during the event. A contract employee at the base of the truck was thrown due to the electrical short. A contract employee in the

bucket of the truck was able to lower the bucket to the ground. A first aid crew was dispatched to the location and an Ambulance was requested. The Ambulance entered the site at 12:50 and at 13:02 the individuals were transported to the local hospital. Due to the electrical transient in the plant, a contract employee performing grinding activities lost control of the grinder and injured his middle finger. This individual received first aid and was transported to the local hospital by his supervisor. The individual injured in the plant was surveyed by Health Physics prior to leaving the site and no contamination was found. The Local Law Enforcement Agency was notified of the Emergency vehicle being dispatched to the site. The State Emergency Operations Center will be notified of the Emergency vehicle entering the site."

The NRC Resident Inspector and local agencies were notified and the state will be notified.

Power Reactor Event Number: 40605 Facility: SUSQUEHANNA
Region: 1 State: PA Unit: [1] [ ] [ ] RX Type: [1] GE-4,[2] GE-4
NRC Notified By: GRANT FERNSLER
HQ OPS Officer: STEVE SANDIN Notification Date: 03/23/2004
Notification Time: 11:00 [ET] Event Date: 03/23/2004
Event Time: 07:46 [EST] Last Update Date: 03/23/2004
Emergency Class: NON EMERGENCY 10 CFR Section:
50.72(b)(3)(ii)(A) - DEGRADED CONDITION Person (Organization):
FRANK COSTELLO (R1)
Unit SCRAM Code RX CRIT Initial PWR Initial RX Mode
Current PWR Current RX Mode
1 N N 0 Refueling 0 Refueling
Event Text
INDICATION OF CRACK FAILURE ON RCS PRESSURE BOUNDARY PENETRATION

"Unit 1 is currently in a refueling outage in Mode 5. During a routine inservice inspection of the reactor vessel, an indication was discovered on the N1B penetration. This is associated with the suction for B Loop of Reactor Recirculation. At 0746 on 3/23/2004, the Control Room was notified that the evaluation was completed and the indication was determined to be unacceptable under the ASME Section XI Code. Based on guidance provided in NUREG-1022, Rev. 2, this material defect in the primary coolant boundary constitutes a seriously degraded condition and is reportable under 10CFR50.72(b)(3)(ii)(A). A final evaluation of the flaw and a repair plan is being developed."

The licensee informed the NRC Resident Inspector.

PRN: PPL's Susquehanna Nuclear Power Plant Returns to Normal Operation

Small Flaw Found in Pipe at PPL Nuclear Site in Luzerne County, Pa.

Publication: Knight Ridder/Tribune Business News

March 25 2004- By Sam Kennedy, The Morning Call, Allentown, Pa. Knight

Ridder/Tribune Business News
Mar. 25--A crack was discovered in a pipe during a routine inspection of the Susquehanna nuclear power plant, PPL Corp. announced Wednesday. The defect posed no immediate threat to the public, according to PPL, which operates the plant. Risk of rupture within the Unit 1 reactor was not significant because the crack was so small, a company spokesman said. "This was nowhere near a break," Herb Woodeshick said. He likened the crack, found Tuesday, to a "blemish.

April 28, 2004 - BERWICK, Pa., /PRNewswire-FirstCall/ -- PPL's Susquehanna nuclear power plant in Luzerne County declared an end to an "unusual event" at 3:52 p.m. EDT on Wednesday (4/28), and plant operators have begun to return the Unit 2 reactor to full power.

The plant entered the lowest of the four emergency classifications for nuclear power plants at 1:25 p.m. EDT Wednesday because of an electrical failure in a power distribution panel located in the Unit 2 reactor building. As a result, the unit's power was reduced to about 80 percent.

"Plant equipment and personnel reacted as expected for this type of situation," said Herbert D. Woodeshick, special assistant to the president for PPL Susquehanna. "Workers isolated the electrical failure and restored power to the affected systems through an alternate electrical supply." The damaged distribution panel supplied power to the cooling system for the main generator and to the system that removes certain gases from the turbine's main condenser, without which the unit cannot operate at full power.

"The plant was in a stable condition throughout the event, and Unit 1 remains at full power," Woodeshick said.

Unit 2 now has been operating for 374 consecutive days.

PPL notified Luzerne and Columbia county emergency management agencies, the Pennsylvania Emergency Management Agency and the Nuclear Regulatory Commission.

The Susquehanna plant, located about seven miles north of Berwick, is owned jointly by PPL Susquehanna LLC and Allegheny Electric Cooperative Inc. and is operated by PPL Susquehanna.

PPL Susquehanna LLC is a member of the PPL Corporation family of companies. Headquartered in Allentown, Pa., PPL Corporation.

(Please reference the following dates for a list of chronic electrical problems at the SSES: “1986”; September, 1988; February 6, 1990; July 23, 1997; June 8-16, 1999; and, April 12, 2005.)

Power Reactor Event Number: 40777
Facility: SUSQUEHANNA
Notification Date: 05/26/2004
Event Text OFF SITE NOTIFICATION
"This event is being reported under 10CFR50.72(b)(2)(xi) as an item of public interest and an event for which other government agencies have been

notified. "At 1600 on 5/26/2004, the operations Shift Manager was notified by the Security Shift Supervisor that an individual [truck driver] had been arraigned by a LLEA [Local Law Enforcement Agency] judge for prohibited items (drug paraphernalia) which were discovered during a routine entrance search of personnel and vehicles. The items were discovered outside the protected area [and] were determined to not pose a threat or attempted threat. The LLEA was called and responded to the site access area and removed the individual to the local barracks, where he was subsequently arraigned on a misdemeanor. The individual's name has been removed from the Susquehanna LLC visitors list. "The Manager of Nuclear Security briefed NRC Region #1 Inspector, Dana Caran, concerning the incident."

The licensee notified the NRC Resident Inspector.

Citizens Voice: 5 detained near Salem nuclear plant
Wednesday 30 June, 2004
by Heidi E. Ruckno Citizens' Voice Staff Writer

Federal and state authorities reported Tuesday that several men of Middle Eastern descent were driving around the Berwick and Shickshinny areas Tuesday looking for the nuclear power plant in Salem Township.

The five men, four from Bangladesh and another of Pakistani descent, were reportedly seen at the Delaware Water Gap rest area along Interstate 80 around 8:20 a.m. They were also spotted in Bloomsburg, Columbia County. State police said they were asking directions to the river near the plant because they wanted to go fishing. Their minivan was pulled over by state police in Shickshinny around 11 a.m. on U.S. Route 11 in Salem Township, four miles south of the Susquehanna Steam and Electric Power Plant.

According to federal and state authorities, the Federal Bureau of Investigation was notified. Because of visa issues, two of the five men were detained by immigration authorities.

"We did stop and detain five individuals, who were believed to be of Middle Eastern descent, because of suspicious activity," FBI special agent Jerri Williams said. Their van was searched Tuesday and authorities did not find anything illegal.

All five men were released Tuesday evening. Williams said Tuesday that there was no cause for alarm, as authorities did not find any links to terrorist activity.

Both the Luzerne County Emergency Management Agency and power plant security were notified about the incident. When asked if the power plant had taken any special precautions, EMA operations and training officer Steve Bekanich said he couldn't speak for the plant.

Power plant spokesperson Joseph Scopelliti said he knew of no procedural changes resulting from the incident. "I know of nothing different," Scopelliti said. "I've seen state police vehicles up and down the highway, but that's every day. We were made aware by state police that there was a concern."

According to Scopelliti, security at the plant is normally very tight. He said that every employee must have proper identification or they will not be allowed on the grounds, and that all unknown people and vehicles and are searched and X-rayed.

"We're ready 24-7," Scopelliti said. "We're not sitting back waiting for something. Everyone that comes up here must have a business reason to come up."

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.

Power Reactor Event Number: 40196 Facility: SUSQUEHANNA

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 21, 2004- Citing rate hikes that take effect Jan. 1 in Pennsylvania, PPL Corp. expects to boost its 2005 earnings from current operations by about 8 percent, the company said Monday.The Allentown-based utility is forecasting earnings of $3.80 to $4.20 per share, up from a projected $3.65 to $3.85 per share this year.The rate hikes, approved by the state Public Utility Commission, affect 1.3 million electricity customers in central and eastern Pennsylvania.

Jan. 20, 2005- Susquehanna set plant record by generating 18-million MWH

Susquehanna's two units generated a record combined output of 18.03-million megawatt-hours (MWH) last year, besting 2003's output of 18-million MWH, PPL Corp. said this week.
Susquehanna-2 also set a site generation record, producing 10.03-million MWH, said PPL spokeswoman Constance Walker. The old record for unit 2 was 9.347-million MWH in 2000, Walker said.
Unit 1 generated 8-million MWH, short of its 2001 record of 9.413-million MWH, Walker said.
PPL said one factor in the record station generation was the installation of new turbines on unit 1 during its spring refueling outage ast year. Unit 2 received a similar upgrade in 2003.
Both units are operated by PPL subsidiary PPL Susquehanna. Unit 1 is a 1,142-MW BWR; unit 2 is a 1,147-MW BWR.
—Report by Daniel Horner

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 4, 2005- 'Unusual event' declared. No fire found and no one is hurt

Smoke at PPL Corp.'s Susquehanna nuclear power plant led to a
low-level emergency declaration on Friday afternoon.
Crews detected smoke in a construction area at one of the Luzerne
County facility's two nuclear units. The unit was out of service
for refueling.
As a result, an "unusual event" was declared for about 55
minutes.
An unusual event is the lowest of the four emergency classifications established by the U.S. Nuclear Regulatory Commission for nuclear power plants. "Our plant fire brigade responded and no fire was found. The

smoke has stopped," said Joe Scopelliti, spokesman for the Susquehanna plant. "There were no injuries. We are investigating the cause. No action by the general public was required."
Unit 2 had been shut down since Feb. 26 for a refueling and inspection outage.
The smoke was detected at 2:57 p.m. in a construction area near a
moisture separator, which is used to "dry" the steam heading for the turbines.
-By Sam Kennedy of The Morning Call

March 6, 2005 -Post-accident monitoring instrument inoperable

Susquehanna Steam Electric Station informed the NRC today by fax that: "At (3 p.m.) on March 6, 2005, the Control Room declared both required divisions for three functions (Primary Containment Pressure, Primary Containment Hydrogen and Oxygen Analyzer, and Drywell Atmosphere Temperature) of Post Accident Monitoring Instrumentation (a Safety System) inoperable. The control room was notified of 'Non Quality' (non-Q) parts installed in both required divisions of a Post Accident Monitoring Instrumentation Recorder. The appropriate LCO Conditions were entered for one or more functions with two required channels inoperable. This equipment has passed all surveillance requirements and has been functional since installation," the statement said.
"Plans are being developed to replace the non-qualified parts.
"This is being reported as an event or condition that could have prevented fulfillment of a safety function required to mitigate the consequences of an accident in accordance with 10CFR50.72(b)(3)(v)(D)."
The NRC Resident Inspector was notified.

April 12, 2005 - Berwick plant shut down

"PPL Corp. officials shut down the Unit 2 reactor at Susquehanna
nuclear power plant in Luzerne County Sunday to repair a battery
charger that is part of the site's electrical system. The plant's Unit 1 reactor continued to operate at 100 percent power."
"Allegheny Electric Cooperative and PPL Susquehanna jointly own the two-unit nuclear power plant, which has a 2,352-megawatt generating capacity.
-Report by the York Daily Record

April 14, 2005- Nuclear reactor restarted

"Operators safely restarted the Unit 2 reactor at the Susquehanna
nuclear power plant in Berwick Wednesday after completing electrical repairs to the unit's battery chargers. The battery chargers are part of the plant's electrical system and are located in a non-nuclear area of the plant."
"On Sunday, plant workers had discovered one of the unit's four chargers was not working properly. Because crews could not repair the electrical problem and conduct a thorough investigation of the Unit 2 direct current electrical system within a specified time period, they manually shut down the unit

as called for in plant procedures."
Susquehanna-2 was out of service this week as plant personnel repaired a battery charger and checked similar components in the 1,147-MW BWR, operator PPL Susquehanna said.
An "expert team" determined that two embrittled wires near a resistor came into contact with each other, creating a short circuit that caused three fuses in the charger to fail April 10, PPL spokesman Lou Ramos said. The charger provides a back-up power source for pump breakers, isolation valves, and other components, he said.

PPL found three similar chargers elsewhere in the reactor and now has configured them to make sure they won't have the same problem, he said. When PPL has collected and analyzed information from the repair and inspection, the company "probably will put something out to industry," as other plants probably have similar battery chargers, he said.
- Report from Nucleonics Week / Volume 7/ Issue 15 / April 14, 2005 and *t*he York Daily Record

April 29, 2005 -Troubled Reactor Shutdown Again Due to Electric Problems*

On Thursday, April 28 at 7:19 a.m. , PPL shut down the Unit 2 nuclear reactor for the second time in a month due a malfunction with a plant electrical transformer.
The main transformer is a non-nuclear component of the plant that increases the voltage of the electricity for distribution on the electrical transmission network. The malfunction appears to be related to the cooling system for the transformer.
Unit-2 was still shut down on April 29.

April 30, 2005 - PPL Susquehanna Restarts Unit 2 Reactor

Operators reported safely restarting the Unit 2 reactor at the Susquehanna nuclear power plant and reconnecting to the electrical transmission network Saturday, April 30 after repairing the cooling system on the unit's main transformer.
A worn motor for one of the transformer's cooling system fans caused the unit to be shut down Thursday morning, plant officials reported.
-Report by Marlene Lang

June 6, 2005 - Third forced closure since April 14, 2005

Unit 2 of PPL's Susquehanna nuclear power plant shut down automatically at 12:33 p.m. Monday, June 6 because of a problem with the electric transmission network.
-PRNewswire report

June 11, 2005 - Unit 2 at the Susquehanna nuclear power plant resumed generating electricity Saturday June 11.
The unit shut down automatically five days earlier after an electrical generator component - a voltage regulator - failed. Plant crews have replaced the regulator and have completed thorough inspections to ensure that the unit's electrical systems are operating properly.
-PRNewswire report

July 25, 2005- PPL Pa. Susquehanna 1 nuke dips to 73 pct power

PPL Corp.'s 1,140-megawatt Unit 1 reactor at the Susquehanna nuclear power station in Pennsylvania dipped to 73 percent of capacity by early Monday, the U.S. Nuclear Regulatory Commission said in a report.
On Friday, the unit was operating at full power.
Power was reduced throughout the weekend to replace feed water valves. PPL began a return to full power on
-Report by Rueters

Sept. 27, 2005- GE receives contract to increase output of PPL nuclear units

A General Electric Co. subsidiary said Sept. 22 that it won a $10 million contract to increase the electric gen-erating capacity of PPL Corp.'s two-unit Susquehanna nuclear plant by about 200 MW combined. This is part of an extended power uprate for the boiling water reactor units at the nuclear plant, near Berwick,
Pa. PPL Corp. currently lists a generating capacity of 2,360
MW for the facility plant. PPL Corp.'s PPL Susquehanna unit is 90% owner of
the nuclear plant. Allegheny Electric Coop. Inc. is a 10% owner. Unit 1 began commercial operation in 1983 and unit 2 in 1985. PPL Corp. will likely file for a 20-year oper-ating license renewal for both units next year.
GE Energy, the plant's original equipment manufac-turer, will work with PPL Corp. to prepare for the uprate, which will be implemented in phases during several refu-eling outages.
GE Energy will perform the engineering analysis and provide documentation support for the uprate as well as the generator scope of work. A combination of GE, PPL Susquehanna and other subcontractors hired by PPL Corp. will perform the balance of the plant work.

-Report by Wayne Barber

Oct. 29, 2005 - Friction in fuel assemblies, control rods shuts down plant

One of the reactors at the Susquehanna nuclear power plant near
Berwick will shut down late Friday for maintenance and should be generating power again within three weeks, PPL Corp. said Wednesday.
Routine testing showed that some of the control rods and fuel assemblies on the Unit 1 reactor are experiencing increased friction, slowing their response time, the company said. The Unit 2 reactor is expected to continue operating normally.
-Report by York Daily Record/Sunday News

March 14, 2006 - Proposed Spent Fuel Exemption for the Susquehanna Nuclear Generating Station Challenged

Eric. J. Epstein, chairman of Three Mile Island Alert, told the NRC why he was concerned about PPL's request to exempt fuel casks, allowing storage of spent fuel. Here is his statement to the Nuclear Regulatory Commission:

Thanks for the opportunity to offer input and share my concerns on PPL's spent fuel cask exemption request.
On April 16, 2003 at the Nuclear Regulatory Commission's (NRC) annual RIC workshop in Rockville, Bryce Shriver from PPL gave a presentation on Safety Management: An Integrated Approach. Among the key areas he touched upon were "Work Management," "Operational Decision Making," "Design and Licensing Basis Control," and "Business Planning and Budgeting". He emphasized that PPL's processes together with their "Independent Oversight" and "Culture" would produce "Safety Performance."
This approach seemed to make sense as PPL prepared for relicensing and power uprates:
- *The Company has contracted with GE Energy to prepare for additional uprates, i.e., Susquehanna 2 (1994) and Susquehanna 1 (1995) had 4.5% bumps. The 200 MWe uprates are scheduled to be implemented in phases during several refueling outages.*
- *Susquehanna Steam Electric Station, Units 1 and 2 are currently preparing for a license extension applications estimated to be somewhere from July- September 2006.*

What went wrong?
It appears PPL has poorly managed human and technical resources to complete projects.
***Background**: PPL submitted a request for an exemption that would enable the plant to begin loading Framatome 9x9-2 spent fuel into the Nuhoms 61BT storage system.*
The Company is not presently authorized to store the fuel.
***Statement of concern**: This "precedent" (1) would bypasses normal review and approval processes for cask loading and penalize plants like Peach Bottom that have followed the NRC's procedures and protocol.*
In my opinion, granting the exemption would weaken the NRC's regulatory protocol of firm, fair and consistent oversight.
***Background**: Normally, the NRC reviews exemption requests for changes the staff has already reviewed as part of an amendment to a cask certificate of compliance (COC). Such exemptions allow the utility to begin cask-loading before NRC completes its rulemaking process to formalize the amendment is complete.*

Statement of concern: *However, Transnuclear* ***has not yet submitted*** *the amendment request to make the change PPL needs. Any exemption would force the NRC to prematurely approve the cask to relieve a self-imposed economic hardship. There is a reason the Agency prides itself on a rigorous oversight process.*

PENNSYLVANIA PUBLIC UTILITY COMMISSION, A-00110550F014, OPINION AND ORDER, "Thus, PPL states that the Recommended Decision failed to address the distinction between the use of the settlement as "binding precedent" and itsadmissibility as evidence in future proceedings..."

Background*: PPL claims the exemption is necessary because the plant will lose full-core offload capability in December, 2006 when it receives and begins to stage new fuel for Unit 2's 2007 refueling outage. Susquehanna had originally scheduled cask-loading to begin in October, 2006.*
However, because of recent fuel channel performance problems at Unit 1, PPL expects Unit 2 will have to undergo a mid-cycle maintenance outage to inspect and replace any bowed fuel channels. That would limit space available in the pool, requiring the plant to accelerate its loading plans.
Statement of concern*: An exemption would reward poor planning (2) of a utility that owns and operates one plant vs. AmerGen and Exelon that own and operate three plants in the state. (3)*

Reactor	***Core Size***	***Lose Full Core Off load Capability***
Limerick 1	**764**	**2006**
Limerick 2	**764**	**2006**
Oyster Creek	**560**	**LOST**
Peach Bottom 2	**764**	**2000**
Peach Bottom 3	**764**	**2001**
Salem 1	**183**	**2012**
Salem 2	**193**	**2018**
Three Mile Island	**177**	**NA**

Station	***Dry Cask Technology***	***Deployment Date***	***Contractor***
Limerick	**BD**	**Summer 2010**	**TBD**
Oyster Creek	**NUHOMS 52B** (4)	**July, 2010**	**None**
Peach Bottom	**Trans-Nuclear TN-68**	**June, 2000**	**Raytheon**

I am asking the NRC deny the exemption and preserve a fair and level regulatory playing field.

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1 Please note that PPL opposed the merger of Come Ed and PECO based on one principal: "precedent."

2 Poor resource planning by a Company headed by a systems manager, i.e., William F. Hecht, warrants an independent NRC evaluation, e.g., Augmented

Inspection Team.
3 PENNSYLVANIA PUBLIC UTILITY COMMISSION, PECO's Response to Eric Epstein's Informal I-8.
4 Holtec has been chosen by AmerGen to provide dry cask services at Oyster Creek.

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 3, 2006 - Alert Declared at nuclear power plant in Luzerne County

Pennsylvania Emergency Management Director James R. Joseph announced that an ALERT was declared Wednesday night at the Susquehanna Steam Electric Station in Salem Township, Luzerne County. This action was necessary due to the activation of the fire suppression system in the Security Control Center. Plant operations have not been impacted and the plant fire brigade is investigating.
“No one has been injured and there was no non-routine release of radioactive material,” said Joseph. “The plant continues at normal operation, but the ALERT could last several hours overnight.”
“An Alert is the second-lowest of four emergency classifications for nuclear power plants. It is declared when an event has occurred that could reduce the plant's level of safety, but backup plant systems still work,” said Joseph.
Preparedness for commercial nuclear power plants includes a system for notifying the public if a problem occurs at a plant. The emergency classification level of the problem is defined by four categories: Unusual Event, Alert, Site Area Emergency and General Emergency. Listed in order of increasing severity.
Pennsylvania Power Light, which operates the Susquehanna Steam Electric Station declared the ALERT at 9:27 p.m.
The State’s Emergency Operations Center (EOC) in Harrisburg was partially activated to monitor the situation. Representatives from the state Departments of Agriculture, Corrections, Education, Environmental Protection, General Services, Health, Public Welfare and Transportation, the Office of Administration, the Department of Military and Veterans Affairs, the Pennsylvania Turnpike Commission, the Pennsylvania State Police, the Fish and Boat Commission, the Public Utility Commission and the American Red Cross joined staff from PEMA in the EOC. At no time during the incident was there a need to issue protective action recommendations to the public.
-Report by the Daily Item, Sunbury, Pa.

April 11, 2006 - NRC grants Susquehanna exemption for spent fuel storage

NRC's Spent Fuel Project Office (SFPO) granted an exemption April 11 to PPL Susquehanna, allowing the utility to load a previously unapproved fuel assembly design into Transnuclear Inc.'s Nuhoms-61BT spent fuel storage system. NRC has exempted the plant from Part 72 requirements that a licensee use systems that NRC approved for use under a general license.
The exemption will allow Susquehanna to start loading Framatome ANP 9x9-2 spent fuel containing 79 full fuel rods and no partial fuel rods. The certificate of compliance (COC) for the Nuhoms-61BT system currently allows the loading of GE 9x9-2 rods or their equivalent with 66 full rods and eight partial rods. Susquehanna has committed to loading fuel with maximum decay heat below 210 watts per assembly, lower than the COC's 300-watt limit. The fuel parameters are generally bounded by the existing COC.

PPL spokesman Joe Scopelliti said the plant will begin moving the spent fuel into dry storage next month. Susquehanna will lose full-core offload capability in December when it begins to stage fuel for Unit 2's refueling outage next spring. The start date for the loading campaign had to be pushed forward from October 2006 because of a possible outage this summer to inspect fuel channels and replace any that show signs of bowing. The spent fuel pool will be needed to store any bowed channels that are removed and must be cleaned out before that activity begins.

But NRC staff rejected PPL's suggestion that the exemption remain in effect until either the completion of its planned 2008 loading campaign or 60 days after NRC grants amendment 9 to the Nuhoms-61BT system, which would add the Framatome fuel to the system's approved contents.

Instead, NRC limited the exemption to the loading of the five casks that PPL said were needed to preserve full-core offload capability through summer 2007. "The staff believes that the use of exemptions in regulatory activities should be minimized," SFPO Deputy Director William Ruland said in an April 11 letter granting the exemption. He added that normal processes for amending COCs should be followed "whenever possible." The NRC believes TN could submit a focused amendment in the near term to allow the Framatome fuel to be added to the approved contents, Ruland said. The cask vendor is scheduled to submit amendment 9 to NRC this month.

In a separate letter April 12, Ruland notified TMI-Alert Chairman Eric Epstein that NRC did not agree with his request to deny the exemption. Epstein asserted in a March 14 teleconference that granting the exemption "would reward poor planning," something that he said "warrants an independent NRC evaluation."

Ruland emphasized that NRC regulations permit licensees to seek exemptions in special circumstances, so long as the exemption "is authorized by law and would not endanger life or property or the common defense and security and is otherwise in the public interest." He said the limitation on the number of casks loaded under the exemption should "enable PPL to avoid the need for further exemptions" for dry storage.

May 1, 2006 - Plant shuts due to leak

PPL Corp. shut the 1,140-megawatt Unit 2 at the Susquehanna nuclear power station in Pennsylvania on April 29 to repair a water leak, the company said in a release.

"The leak is minor – significantly less than the amount that would require us to shut down for repairs according to the plant's operating procedures -- and it does not affect our ability to operate safely," Robert Saccone, vice president of
Nuclear Operations for PPL Susquehanna, said in the release.

"We made the proactive decision to find and fix the leak now, so that we don't run the risk of having to shut down the unit during the summer if the leak gets worse. In the summer months, the regional power grid, consumers and PPL count on Susquehanna to provide reliable power as electricity use increases," Mr. Saccone added.

PPL said it planned additional maintenance in other areas of the plant during this short outage that will help maintain the reliability of the unit, which was in service for 322 consecutive days before this shutdown.

The unit was operating at full power early Friday.

The 2,245 MW Susquehanna station is located in Berwick in Columbia County, about 125 miles northwest of Philadelphia.
There are two units at the station, the 1,135 MW unit 1 and the 1,140 MW unit 2.
-Report from NuclearFuel Volume 31 / Number 9 / April 24, 2006

June 15, 2006 - Monitoring system trips shutdown at Unit 1

At 3 a.m. on June 15, the Susquehanna Unit 1 reactor automatically "scrammed due to an apparent neutron monitoring trip while transferring Reactor Protection System power supplies," company documents stated.
A "scram" means a shutdown in nuclear industry lingo.
"All rods [fully] inserted, and both reactor recirculation pumps tripped," according to the report, which explained, reactor water level lowered to -38" causing level 3 (+13") and level 2 (-38")isolations, and was restored to normal level (+35") ... and subsequently the feedwater system. All isolations at this level occurred as expected. No steam relief valves opened. Pressure was controlled via turbine bypass valve operation. All safety systems operated as expected."
A reactor recirculation pump was restarted to re-establish forced core circulation. The reactor is currently stable in condition 3. An investigation into the cause of the shutdown is underway. Unit 2 continued power operation, according to the report.
The NRC resident inspectors were notified, the company stated.
-Report by Marlene Lang

Sept. 6, 2006- Shipment to plant had radiation reading at 4 times allowed level

A container shipped from Vermont Yankee on Aug. 31 ended up at its destination later that night with radiation readings four times higher than those allowable under federal law, according to a report filed Sept. 1 with the Nuclear Regulatory Commission (NRC).
The shipment, a box measuring 6x7x8 feet containing a machine used to configure fuel rods in the power plant's spent fuel pool, registered no more than 60 millirem per hour before it left Vermont, according to Vermont Yankee (VY) records. That level is well below the federal Department of Transportation's (DOT) 200 millirem hourly contact exposure limit.
However, when it arrived at the Susquehanna reactor in Berwick, Pa., the bottom of the container registered 820 millirem per hour, more than four times the DOT limit.
The container was shipped on a flatbed truck by a private contractor Hittman Transport Services of Barnwell, SC. As of Tuesday the container remained closed in a controlled area at the Susquehanna plant, while inspectors made special preparations before opening it, according to NRC spokesman Neil Sheehan.
He said they planned to open the container Wednesday.
En route to its destination, the truck stopped at rest stops on the westbound side of the Massachusetts Turnpike and on southbound Interstate 87 after existing Interstate 90,

according to an incident report filed by Susquehanna officials, who were required to make a report to the NRC because of the high radiation recording.
No one to the knowledge of the driver came in contact with the shipment, the report states. The truck arrived at Susquehanna at 8:45 p.m. and the driver, who was wearing a radiation detection monitor, slept in the vehicle. Sheehan said the driver's dosimeter showed readings well within acceptable levels.
A spokeswoman for the trucking company said she had no knowledge of the incident.
According to the NRC report, the shipment was formally received at the Susquehanna facility at 8:05 a.m. the next morning. The high reading was recorded at 11:15 a.m., and Susquehanna officials notified the NRC at 12:15 p.m.
According to the report, the shipment showed no signs of surface contamination, and it exceeded the dose rate limit only on the bottom of the container once it was lifted off the truck. "Doses under the trailer prior to lifting the shipment did not exceed the limit," the report states.
Unless someone got right up under it, the probability that someone would have received any kind of exposure from that configuration is low, said NRC Region I deputy administrator Mark Depas.
VY spokesman Rob Williams also emphasized that point: Despite the unexplained high radiation levels, the shipment represented no threat to public health and safety in transit because the radioactive side was against the bed of the truck, which provided additional protection, he said.
At no time during the shipment was there any additional exposure to anyone because the flatbed truck provided adequate shielding, Williams said. "In fact, the radiation level in question was detected only at the bottom of the package, and only after it was lifted off the flatbed, so this had no impact on public health and safety."
Vermont Yankee is responsible for shipments while in transit, Williams noted. Two experts from VY's radiological shipping group had left for Pennsylvania to determine what may have caused the increase, he said Tuesday.
"We've reviewed our radiological survey and confirmed that the package left here in compliance," Williams noted.
Sheehan speculated the increase might have been due to the machine shifting during transit, resulting in a part with higher contamination levels closer to the bottom of the box. Or, he said, a piece of debris from the VY spent fuel pool could still have been attached to it.
The tool is what Sheehan called a cutter-shearer machine that crushes control rods in order to ship them more easily. Control rods are used to separate spent fuel rods in a fuel pool. They are inserted between the fuel rods in crucifix form, with a centerpiece and four blades inserted between the fuel bundles to stop the fusion process, Sheehan said.
He said reactor operators periodically install new control rods during cleanup of their spent fuel pools.
Anti-nuclear activist Ray Shadis, technical advisor to the Brattleboro-based New England Coalition, speculated that the discrepancy in radiation readings could have been due to inaccurate VY detection equipment.
What is serious is the possibility that VY radiation detection was off by a whopping factor of four and/or the probability that the contents of the package leaked and/or

became more exposed as shielding shifted or settled, Shadis said in an e-mail to the Vermont Guardian.

At 820 millirem/hour, a person exposed to the hottest part of the container could have, in one hour, received eight times the annual dose allowed by the NRC, or their annual allowable dose in less than eight minutes, Shadis noted.

Unlike the DOT, the NRC does not set a contact exposure ceiling, but the agency limits exposure for members of the public to 100 millirem annually.

"This is just a real sloppy performance," Shadis continued. "Let's hope it is an exception and not the standard.

-Report by Kathryn Casa of the Vermont Guardian

Sept. 6, 2006

High radiation reading receives "White" violation rating

A shipment from the Vermont Yankee nuclear plant that was giving off more than four times the allowable level of radioactivity posed a "low to moderate" safety risk to the public, federal regulators said Tuesday.

The Nuclear Regulatory Commission issued a preliminary "white" finding about the August shipment of a device designed to crush and cut reactor control rods from the plant site in Vernon to Salem Township, Pa.

The NRC uses a color-coded system to denote safety risks, with "green" indicating a very low risk, "white" low to moderate, "yellow" substantial and "red" high, said agency spokeswoman Diane Screnci.

In a letter dated Tuesday to Vermont Yankee, the NRC said its finding was preliminary and that it had not yet made a final determination of what enforcement action might be taken.

Screnci said she doubted the plant would be fined, but said it would get some stepped-up scrutiny.

- Associated Press report.

Nov. 8, 2006 - Nuclear regulators slapped Vermont Yankee with a safety violation Tuesday, after determining plant owners failed to take the highest level of precaution when they shipped radiation-exposed equipment.

Two months ago a piece of equipment was sent from Vermont Yankee in a shielded container on a flatbed truck to a nuclear power plant in Pennsylvania. When it arrived, the freight's radiation level measured at four times the allowable level.

Entergy Nuclear received a "white" inspection finding from the Nuclear Regulatory Commission, the second lowest of the four levels of findings. That means the radioactivity posed a "low to moderate" safety risk to the public, according to Neil Sheehan, spokesman for the NRC.

The equipment Entergy was sending to the Susquehanna nuclear power plant was a control rod crusher and shearer, owned by a separate vendor. In Pennsylvania, inspectors found a "sliver of metal" of high radioactivity and two small "hot particles" fell from the top of the crusher to the bottom, Sheehan said. That kind of disturbance in the equipment, when in transit, is not uncommon, he said.

A white inspection finding from the NRC triggers an increased oversight at Vermont Yankee. For the next four quarters, federal inspectors will have an enhanced role in reviewing how Entergy decontaminates and prepares freight before it leaves the Vernon campus.

But first Entergy has 10 days to file an appeal with the NRC, challenging the finding. For now, the NRC is still calling the white finding "preliminary," and has not said for sure what enforcement action will be taken.

Efforts to reach Entergy officials Tuesday were unsuccessful.

This is the first time in two years Vermont Yankee has received a white inspection finding. The plant hasn't gotten anything higher than a "green" inspection finding for the last two years, the lowest finding. In 2004, the NRC gave the plant a white finding for its distribution, or insufficient distribution, of tone alert radios.

The NRC uses a color-coded system to denote safety risks, with "green" indicating a very low risk, "white" low to moderate, "yellow" substantial and "red" high.

Reporty by Kristi Ceccarossi of the Reformer, New England Newspapers

Dec. 18, 2006 - Sirens mistakenly sound at nuclear power plant

Emergency sires near PPL’s Susquehanna nuclear power plant went off around 11 this morning, but company officials said it was part of a test and not an actual emergency.

“We conduct silent tests of the siren system every two weeks,” said Lou Ramos, spokesman for the plant. “During a scheduled test this morning, the sirens mistakenly received a signal to sound, rather than a signal for a silent test. We apologize for any anxiety that this may have caused among area residents.”

The sirens can be sounded by PPL Susquehanna or by emergency management agencies in Luzerne or Columbia counties.

“The sires that sounded today were part of the old siren system, which PPL Susquehanna is in the process of replacing,” Mr. Ramos said. “We will conduct a full-scale test of the newly installed siren system tomorrow.”

Emergency sirens around the plant are in place to notify the public to tune into emergency broadcast stations on television

or radio in the event of an emergency at the nuclear plant or in the community.
-Report by The Daily Item Publishing Company

Dec. 20, 2006- NRC Finalizes White Finding for Vermont Yankee Nuclear Plant over Shipment of Radioactively Contaminated Equipment

The Vermont Yankee nuclear power plant will receive additional oversight from the Nuclear Regulatory Commission based on a violation involving a shipment of radioactively contaminated equipment. The violation, which has now been finalized, stems from a shipment that went from Vermont Yankee to a Pennsylvania nuclear power plant last summer.
The NRC uses a color-coded system to categorize inspection findings. They range from green, for a very low safety issue, to red, for a highly significant safety issue. In this case, the Vermont Yankee violation has been determined to be white, which signifies the issue is of low to moderate safety significance.
The finding is based on an inspection the NRC carried out from Sept. 6 through Oct. 6, 2006.
On Aug. 31, 2006, Vermont Yankee, which is located in Vernon, Vt., and operated by Entergy, prepared and shipped a package containing a radioactively contaminated control rod crusher/shearer to the Susquehanna nuclear power plant, in Salem Township, Pa. U.S. Department of Transportation (DOT) requirements apply to such shipments. DOT requires that this type of shipment be prepared so the radiation level on any external surface of the package not exceed 200 millirems per hour.

However, upon arrival at the Susquehanna plant on Sept. 1, 2006, the radiation level at a location on the bottom exterior surface of the package was measured at about 820 millirems per hour. It was later determined that during transit, discrete highly radioactive particles shifted to the bottom of the package, resulting in the radiation levels in excess of the DOT limits.
It is important to note that no actual public radiation exposure occurred during the shipment from Vermont to Pennsylvania because the affected package surface was inaccessible to members of the public.
The actual condition did not involve an exposure or hazard to the public, but it had the potential to adversely affect personnel who would normally receive the package or respond to an incident involving the package since responders could have a reasonable expectation that the package conformed with DOT

radiation limits, NRC Region I Administrator Samuel J. Collins wrote to Entergy in a letter regarding the enforcement action.
In addition, it was fortuitous that the surface of the package was inaccessible to the public during transport.
The company did not request a regulatory conference on this matter but is required to respond to the violation within 30 days.
The NRC will conduct a supplemental inspection at a future date to evaluate the companys corrective actions.
-NRC report

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

2007

PPL to seek license for new nuclear generator at Berwick

PPL Corp. announced on Wednesday it notified the U.S. Nuclear Regulatory Commission that it plans to apply for a license to construct and operate a third nuclear generator at its Susquehanna River plant near Berwick.

The Allentown-based company also filed a request for an interconnection study with PJM Interconnection, an organization that coordinates the movement of electricity throughout much of the mid-Atlantic region.

PPL is awaiting a license renewal for its two Salem Township nuclear generators, which supply about 25 percent of PPL’s total output, and company spokesman Dan McCarthy said a rejection of those renewals could have serious repercussions for the new license.

“If we didn’t get them, I don’t know that we would go ahead with building the third one,” he said.

The company is also considering expansions of hydro and coal plants, he said.

The letter of intent to the NRC lets the company hold a place in the processing line and retain the potential for federal production tax credits and federal loan guarantees, which expire for any application submitted after 2008, according to Jim Miller, PPL chairman, president and chief executive officer. The study request gives the new generator consideration in future regional power planning studies.

Miller said the construction would only go forward as a joint venture with another energy company, which hasn’t been chosen, according to McCarthy.

The $70-million cost of the licensing application wouldn’t be accounted for until the plant goes online, meaning the company doesn’t expect the expense, which would mostly be spent by the end of 2008, to affect earnings forecasts for current operations.

McCarthy said no specific timelines for construction or power generation exist. Studies of safety and environmental impacts have not yet been done.
Though he didn't expect the 10-mile-radius emergency planning zone to increase with a third generator, McCarthy said there would be more nuclear material onsite.
Critics believe PPL needs to take care of its current site before moving on to new ventures.
"Rate payers are bailing PPL out for the initial boondoggle," said Eric Epstein, chairman of TMI Alert, among membership in other organizations. "There's just not enough water resources available to support another nuclear reactor."
The plant already uses millions of gallons of water a day from the river, much of which evaporates through its cooling towers, he said, raising concerns that a third generator would seriously affect the downstream flows.
McCarthy said the company maintains a reservoir in New York that could be diverted into the river on low-flow days to compensate.
PPL has 30 generating sites in Pennsylvania, Connecticut, Maine, Illinois, Montana and Long Island, N.Y., but the Susquehanna site is the company's only nuclear plant, McCarthy said. Coal plants produce about 55 percent of the company's output, with generation from hydro, oil and natural gas producing the remaining 20 percent.
-Report by Rory Sweeney of the Times Leader

Aug. 2, 2007 - PPl reports earning jump, raises forecast

PPL Corp. reported second-quarter earnings of $345 million, a jump of more than 90 percent compared to the same period of 2006. Earnings per diluted share rose about 87 percent, to 88 cents.
Allentown-based PPL distributes and generates electricity in the midstate.
The earnings increase was driven by gains on the sale of a business in El Salvador, according to PPL. Excluding that and other special items, operating earnings rose by almost 19 percent, to 63 cents per share, according to the company.
PPL beat the average analyst estimate of 51 cents per share, according to Yahoo Finance.
PPL raised its forecast for full-year earnings from ongoing operations to $2.40 to $2.50 per share, up from $2.30 to $2.40 per share. -
-Report by David Dagan

Sept. 12, 2007- PPL fires and sues its siren installer

PPL Corp. has fired and sued the Boston company it hired to replace the siren system around the Susquehanna nuclear power plant in Salem Township.
PPL claims the siren vendor, Acoustic Technology, failed to deliver on the contract because some of the 76 warning sirens it installed in a 25-mile radius around the plant failed to sound during tests earlier this year.
Attempts to reach Acoustic Technology were unsuccessful.

PPL's existing siren system, installed 25 years ago, continues to be fully functional and in use until the company selects a new vendor. The sirens are intended to alert the public to emergencies at the plant or in the community.
- Report by David Falchek of the Citizens Voice

Sept. 19, 2007- PPL pays to settle dispute over water use at plant
Two electric utilities, PPL Corp. and Exelon Corp., have paid large sums of money to settle disputes with the Susquehanna River Basin Commission over the amount of water they use to operate their nuclear power plants.
PPL last week agreed to pay $500,000 to the commission to settle a claim that it did not get permission six years ago to increase the water it takes from the river.
Last December, Exelon Nuclear paid $640,000 to settle a similar claim related to its Peach Bottom plant in York County.
The commission controls water withdrawals within the Susquehanna River basin in Pennsylvania, New York and Maryland to ensure that adequate supplies are available to all users. Under its rules, companies like PPL and Exelon must seek the commission's approval for any change in processes that requires them to increase water usage by 100,000 gallons a day, said Susan Obleski, commission spokeswoman.
The commission contended that PPL exceeded that threshold in 2001.
PPL disagreed with the commission's finding, but it agreed to settle the dispute so it could proceed with a request to increase its water use from 47 million gallons to 66 million gallons a day, said Luis Ramos, a spokesman for the utility. The increase was approved by the commission last week.
With the increase, the company uses about six-tenths of 1 percent of the river's water supply, Ramos said.
The monetary settlements, though large by the commission's standards, are inadequate, said Eric Epstein, chairman of Three Mile Island Alert, a watchdog group that has challenged PPL's requests. The settlements fail to underscore the commission's message that water is a finite resource, he said.
"The New England Patriots paid more for stealing football signals than PPL was fined for stealing water from the river," Epstein said.
PPL will need the water if the U.S. Nuclear Regulatory Commission approves its request to increase the amount of electricity its two Susquehanna reactors produce by about 100 megawatts, Ramos said. If approved, the increase would allow the company to produce electricity sufficient to power about 60,000 additional households.
The two reactors produce enough electricity to power about 1 million homes.
As the demand for electricity increases, the commission anticipates that the demand from utilities for water will grow. PPL already has announced that it is considering adding a third nuclear reactor at its plant north of Allentown.
"Right now the basin is a hotbed for future power production," Obleski said. "We see that as a growing sector."
-Report by Garry Lenton of the Patriot-News

Jan. 24, 2008 - Refueling shipment exceeded radiation limit

A shipment to the Susquehanna nuclear plant arrived on Friday emitting radioactivity beyond the limit allowed by the federal Department of Transportation, the U.S. Nuclear Regulatory Commission announced on Tuesday.

"This did not impact the public," NRC spokesman Neil Sheehan said. "Nevertheless, DOT sets these limits so the public is protected."

He said it is "premature" to discuss potential enforcement actions.

The plant is jointly owned by PPL Corp. and Allegheny Electric Cooperative Inc.

The shipment, containing equipment to be used during an upcoming refueling and maintenance outage, was surveyed for radioactivity and passed before leaving North Carolina. A similar survey upon arrival found the underside of a box containing equipment used on the refueling floor emitted 350 millirems per hour, above the 200-millirems-per-hour exposure limit.

"The spot was in a place that was inaccessible to anyone," PPL spokeswoman Nancy Bishop said. "When it left North Carolina, the measurements were below the limit. When it arrived here, the measurements were above the limit. What probably happened is that the components shifted in transit."

The box was put into an onsite facility "designed and licensed to hold radioactive material," she said, where it will stay until it's needed for refueling.

The equipment was being shipped by GE Hitachi Nuclear Energy, which PPL hired to execute the refueling. The equipment can become radioactive, Bishop said, because "it can come in contact with various radioactive components when it's on the refuel floor … during maintenance."

- Report by Rory Sweeney of the Times Leader

Oct. 27, 2008- NRC Monitoring alert issued at Susquehanna plant

The Nuclear Regulatory Commission (NRC) is monitoring an Alert declared this afternoon at the Susquehanna 2 nuclear power plant in Salem Township (Luzerne County), Pa. An Alert is the second-lowest of four levels of emergency classification used by the NRC.

At 4:15 a.m. today, maintenance work was initiated on a water line that is part of a reactor safety system for the plant. That work involved the use of a "freeze seal" – that is, placing a device containing nitrogen over a section of piping so that the water inside the line can be frozen. Once frozen, the line can be isolated to allow maintenance to be performed on it.

PPL, the plant's owner and operator, declared an Alert at 12:06 p.m.

-Report from Nuclear Regulatory Commission

Sept. 22, 2010 – Plant officials notify NRC of a non-emergency event. Plant says the Unit 2 high pressure coolant injection system was determined to be in operable due to a minor lube oil leak that could not be corrected immediately.

<u>Nov. 12, 2010</u>- The NRC issued its findings from an inspection of Units 1 and 2 for the third quarter ending Sept. 30, 2010. In its report, the NRC said it issued a preliminary white finding (the second lowest in severity) based on a July 16, 2010, flooding event in the Unit 1 condenser bay. The flooding event also yielded two non-cited violations. In addition, the NRC said two other non-cited violations were found during the quarterly review.

The preliminary white violation stems from inadequate procedures in the maintenance and operation of the main condenser water boxes and circulating water system, the NRC said. This resulted in an internal flooding event on July 16, 2010, that resulted in 1 million gallons of water 12 feet deep in the Unit 1 main condenser bay. The flooding caused a shutdown of the reactor for about 20 days.

The cause and severity of the flooding was the improper installation of a gasket and deficiencies that led to a delayed response in controlling the leak.

The NRC said, "It was determined that the leak initiated from the D main way cover gasket being partially extruded under normal system operating pressures," the NRC said. "This was caused by an inadequate procedure to install the main way gaskets upon completion of maintenance."

In addition, the NRC said that D water box was mislabeled as B. "This led to operators in the field misidentifying the water box that was leaking and the operators in the control room selecting the wrong water box to isolate," the NRC report said.

Finally, the NRC said, it was determined that plant procedures "did not have specific instructions on how to isolate a condenser water box leak. … No guidance was provided to assist the operator in identifying the location and isolating leaks associated with the water boxes."

The NRC noted that plant operator PPL "did not adequately: 1) evaluate previous circulating water system water box main way gasket leaks (April 2007 and March 2008) to ensure that future occurrences could be prevented; and 2) evaluate and correct a known issue in an off-normal procedure that complicated the operator's response to the event (November 2009.)"

The NRC said it issued a preliminary white finding of low to moderate safety significance, and said a final determination would be announced within 90 days of its Nov. 12, 2010, letter to the plant.

As offshoots from the July 16, 2010, incident, two non-cited violations were issued of low safety significance.

One of them involved an inadequate procedure to transfer water from the condenser area to a condensate storage tank berm. The NRC noted that the procedure failed to include a maximum level at the storage tank berm that was acceptable to limit interaction with other safety-related equipment.

The NRC said water was transferred to the berm to a level that caused water intrusion into cable conduit and junction boxes of other equipment.

"Failure to have an adequate procedure for transferring water from the condenser area to the berm to limit interactions with other safety-related equipment is a performance deficiency which was reasonably within PPL's ability to foresee and correct," the NRC report said. "The finding was not subject to traditional enforcement because there were no actual consequences, it was not willful, and did not impact the NRC's ability to regulate." The matter was entered into PPL's correction action program, and was treated as a non-cited violation by the NRC.

Another non-cited violation ascertained after the July 16, 2010, flooding event was the failure to accurately model the simulator for the reactor core isolation cooling (RCIC) operation at reduced flow rates. Following the July 16, 2010, incident, PPL identified that the RCIC system operation was unstable when attempting to operate in automatic flow control with the flow control set below

the designed flow rate. "Simulator training conditioned the operators to expect RCIC system operation to be stable at all selected flow rates when operated in automatic," the NRC said. "As a result, during an actual event, the operator could misdiagnose the cause or means to correct unstable RCIC operation and eliminate an injection system to the reactor pressure vessel unnecessarily."

The NRC said PPL entered the matter into its correction action program.

Two other self-revealing non-cited violations were found. One involved an Aug. 10, 2010, incident in which operators discovered a Freon leak from the Unit 1 chiller. Because of the leak, an alert was issued, the second lowest of four emergency classifications.

During the incident, PPL said it did not have installed or portable means to determine Freon concentrations, the NRC said. "Without the ability to remotely measure Freon concentrations or measure Freon concentrations using a portable meter, PPL could not evaluate the atmospheres during a known Freon leak and was forced to rely upon personnel showing exposure effects to declare this event," the NRC report said. "Furthermore, PPL did not have the Freon measurement capability to determine if respirators were required. Thus, PPL did not have two of three methods for determining (what was) available to them for a known hazard."

PPL entered this matter into its corrective action program.

Another self-revealing non-cited violation involved simulator modeling for its integrated control system. "Since the simulator model did not reflect actual plant performance, the Susquehanna simulator introduced negative operator training that affected the ability of the operator to take the appropriate and timely actions during an actual event to prevent a plant scram (emergency shutdown)," the NRC said. The NRC said this was of very low safety significance and was treated as a non-cited violation because it was entered into PPL's corrective action program.

The NRC report also listed three violations of low safety significance.

Nov. 19, 2010 – The NRC issued a report on an inspection of Units 1 and 2 conducted from Sept. 13 to Oct. 8, 2010. The inspection centered on selected risk components and operator actions in both safety-related and non-safety related systems. The review included components such as pumps, breakers, heat exchangers, transformers, and valves.

In the report, the NRC said it found one item of very low safety significance that was treated as a non-cited violation. The item involved the design, testing and operation of a 125-volt direct current battery charger circuit breaker.

According to the report, plant operator PPL "did not adequately evaluate the over-current trip setting test results" for a particular breaker "to ensure they were within the established acceptance limits, and subsequently placed the breaker in-service with an as-left trip setting outside of the approved acceptance band." The breaker was returned to service on Feb. 8, 2010, the NRC said.

It added that other breakers were returned to service prior to that Feb. 8, 2010, date with setting values outside of acceptance levels. "The team identified that six of the 12 breakers reviewed had recorded as-found trip setting values outside of the acceptance range," the report said. "PPL performed the six-year breaker preventive maintenance work only during plant outages, by replacing an installed breaker with one for which a preventive maintenance was recently completed, then placing the just-removed breaker into a spare status. Then, during the next outage, typically one to three years later, a

preventive maintenance is performed on the spare breaker and it is returned to service in a different load center location." The NRC added that it noted "there were several different trip setting values for the various direct current load center breakers."

The NRC noted in its finding that a test program much be established to ensure that all testing performs satisfactorily and that test results are documented to make sure that test requirements have been satisfied. However, the NRC noted that between Jan. 16, 2008, and Oct. 8, 2010, PPL "did not adequate evaluate direct current circuit breaker test results to ensure that the test requirements had been satisfied."

These issues were entered into PPL's corrective action program.

On Nov. 16, 2010, the NRC issued a brief report on its evaluation of an Oct. 5, 2010, emergency preparedness exercise at the plant. No findings were identified.

May 31, 2011 – The NRC issued a determination stemming from a request originally submitted in January 2008 for information on the Berwick plant's ability to manage gas accumulation at its facilities.

Based on the responses from PPL, the plant operator, the NRC said the licensee has "acceptably demonstrated" that gas accumulation "is maintained less than the amount that challenges operability of these systems, and that appropriate action is taken when conditions adverse to quality are identified."

July 20, 2011 – The NRC issued a letter on the completion of its triennial (every three years) fire inspection of Units 1 and 2 at the plant.

Based on the inspection, two findings of very low safety significance were identified. The NRC said it would treat the findings as non-cited violations because they were entered into the plant's corrective action program and they were of very low safety significance.

One of the violations involved the failure of plant operator PPL to adequately implement "a fire water supply system with two redundant 100 percent capacity fire water pumps and three sources of supply water."

"Design flow rates could not be achieved and maintained by a single fire water pump for all required sprinkler systems," the report said. "PPL performed an operability evaluation and determined the affected sprinkler systems were capable of performing their intended functions at lower flow rates and for a shorter duration than originally specified by plant design. In addition, the Unit 2 cooling tower basin was determined to be inoperable as a sole source of supply water for the fire water system."

"From initial plant conduction until present," the report added, "PPL failed to provide two redundant fire water pumps that could be supplied from any of three separate water sources." The NRC said the issue was entered into PPL's corrective action program.

The other finding involved the failure to implement all provisions of the approved fire protection program. “Specifically, PPL established acceptance criteria in the fire pump performance tests that were non-conservative compared to design basis requirements and the test acceptance criteria were insufficient to demonstrate that the fire pumps could provide sufficient pump pressure to satisfy required sprinkler system hydraulic needs.”

The report added, “PPL’s corrective actions program required fire protection deficiencies be identified and corrected. The team determined that PPL had not adequately implemented the required quality assurance criteria for fire pump testing, in that the combined tests did not demonstrate that pump performance conformed to design requirements or would perform satisfactorily in service.”

July 27, 2011 - The NRC staff issued a letter on its inspection of TMI for the quarter running from April through June 2011. The staff said no findings of significance were identified.

The report added that inspectors determined “that corrective actions to address configuration control performance deficiencies from the first half of 2010 and transient material control deficiencies from all of calendar year 2010 continued to be effective.” It added that the number of configuration control deficiencies identified in the first half of 2011 “were notably reduced from the first half of 2010.”

But the report noted that inspectors “identified several instances for which corrective action timelines was not commensurate with potential significance of degraded equipment conditions.” It added, “Station management acknowledged the issues, verified they were captured in the corrective action program, and initiated several significant station-wide actions to reemphasize worker performance fundamentals. The inspectors determined these correction actions were appropriate and observed improved worker fundamental performance through the end of June 2011.”

Aug.19, 2011 – The Unit 2 reactor of the nuclear power plant shut down automatically at 10:46 a.m. The unit was operating at full power at the time. The plant resumed generation of electricity on Aug. 23, 2011.

The shutdown occurred during scheduled equipment testing. A review by staff found a single-point wiring deficiency in the unit’s digital control system, the plant said.

Unit 1 was not affected by the events.

Sept. 1, 2011 –The NRC completed its mid-cycle performance of Susquehanna Units 1 and 2

The NRC determined that the performance of Unit 1 during the most recent quarter ending June 30, 2011, was within the "degraded cornerstone column" of its oversight process. This was due to one finding having low to moderate safety significance and one performance indicator having low to moderate safety significance.

The one finding related to an internal flooding event on July 16, 2010, that required a plant shutdown. The performance indicator involved unplanned shutdowns occurring in 2010 on April 22, May 14, and July 16, and on Jan. 25, 2011.

The NRC found that the performance of Unit 2 was within the licensee response column of the oversight process.

Nov. 8, 2011 – The NRC issued a severity level IV violation against the plant operator for failure to notify the NRC of the change in medical status of a licensed reactor operator. It was determined that the operator needed to wear eyeglasses as early as April 2009, but plant licensee PPL "did not inform the NRC or request an amended license" for the operator until August 2011.

"Therefore," the NRC said, "the reactor operator performed license duties without an NRC-approved, amended license from April 2009 through August 2011, until the NRC identified the issue."

The NRC noted that this is a "repetitive" issue. (See report dated Jan. 28, 2010, in which a senior reactor operator continued to conduct NRC-license activities after not meeting a specific medical prerequisite and there was no notification to NRC to ensure the person's license was conditioned to require corrective lenses.) In that Jan. 28, 2010, report, the NRC noted that a civil penalty would not be proposed, but "significant violations in the future could result in a civil penalty."

The latest NRC report does not mention any possible civil penalty for the level IV violation.

The violation was found during an examination for the third quarter from July through September 2011. In the report, the NRC also found a non-cited security level IV issue and two NRC-identified and one self-revealing finding, all of very low safety

significance. Additionally, the report said two PPL identified violations were determined to be of very low safety significance and were treated as non-cited violations.

The other level IV violation involved the recording of reactor coolant system leakage values under the performance indicators for Units 1 and 2.

“”PPL submitted inaccurate data for the affected performance indicators for Units 1 and 2 every quarter from April 2000 through its current submittal of June 2011,” the report said. “PPL’s failure to identify and correct the recurring errors over this period of time indicate the existence of a programmatic issue.”

Even though the data didn’t cross certain thresholds, “the inspectors concluded that PPL had reasonable opportunity to foresee and correct the inaccurate information prior to the information being submitted to the NRC,” the NRC report said. “The finding was not considered to be more significant since had this information been accurately reported, it would not have likely caused the NRC to reconsider a regulatory position or undertake a substantial further inquiry.”

The matter has been placed into PPL’s corrective action program.

Jan. 6, 2012 – The NRC issued a notice of violation to a senior reactor operator who failed to notify officials of the Susquehanna Steam Electric facility of a criminal violation filed against him by Indiana State Police prior to his return to work in July 2010.

The NRC said the senior operator had been issued a citation on July 10, 2010. The citation was for public indecency/indecent exposure, according to NRC records.

The senior operator did not report the legal action to his superior or any other PPL related official when he returned to work at the Berwick plant on July 18, 2010. He subsequently reported the legal action on July 21, 2010.

The senior reactor operator had unescorted access at the plant and was required by NRC regulations to promptly report legal actions issued to him by law enforcement agencies. The senior reactor operator was on vacation on July 10, 2010, and was scheduled to return to work on July 21, 2010. However, he reported back three days earlier to assist in a plant-flooding event, the NRC said.

The operator is no longer employed by PPL, the owner of the plant. He was issued a notice of violation, but no enforcement action is being taken against PPL, the NRC said.

May 2, 2012 – The NRC issued a report on the first quarter inspection of Units 1 and 2. The report listed three NRC-identified findings and one self-revealing finding of very low safety significances. Also listed were two licensee-identified violations determined to be of very low safety significance. All findings were treated as non-cited violations.

One NRC-identified issue involved plant licensee PPL's safety-related motor operated valve program. The NRC noted that the program "lacked a procedure, qualification and prescribed acceptant criteria for actuator grease analysis and PPL improperly implemented maintenance instruction for lubricating valve stems."

In the report, the NRC noted that "PPL did not have a procedure for qualitative motor operated valve grease analysis … there was a general lack of documentation of grease analyses associated with the grease sample work orders…(and) the current motor operated valve engineer and predecessor did not possess a qualification for grease analysis."

The report added, "The lack of a procedure, repeatable acceptance criteria, qualification, and multiple cycles without stem lubrication could result in untimely actuator overhauls and ultimately motor operated valve degraded performance."

The NRC also identified a problem in that "PPL did not have adequate instrumentation to assess and determine if an abnormal radiological effluent release was in progress such that the emergency action level classification process would declare an Alert accurately and in a timely manner." The report noted that PPL had previously received two non-cited violations for inadequate instrumentation since 2008.

A third NRC identified issue involved written procedures for radiation work permits. The issue materialized when some workers attempted to transfer an 1100 Curie Cesium 137 source from a shipping cask on Dec. 5, 2011. During this project, the contractor directed the effluents technician to use additional tooling to provide more manual pressure to withdraw a shield plug. According to the report, the plug was withdrawn about three inches more than prescribed and the electronic dosimeters worn by the contractor and the effluents technician immediately went off, indicating high dose rates. The exposure rate was approximately three seconds before corrective actions took effect.

However, higher levels of PPL management was not informed of the incident until the source load operation had been successfully completed, the NRC report said. "Consequently, the required actions were not completed prior to restarting work and measures to prevent reoccurrence were not fully implemented," the report said.

The self-revealing finding was identified "when a worker did not comply with a radiological barrier and protective measures for high radiation area entry." On March 22, 2012, an effluents department employee was working in the Unit 1 turbine building when he tried to get a better view of a doorway for a future high-efficiency particulate air filter move, the report said. The worker leaned into a posted high radiation area during this process. The worker exited the area and it was determined the total dose was 1.5 millirem.

The PPL-identified issues involved transient combustibles being stored in a restricted area in the Unit 1 reactor building on Nov. 30, 2011, and the lack of preventative maintenance or replacement of the overspeed test controller at the electronic governor module of Unit 2's high pressure coolant injection.

May 7, 2012 – The NRC issued a report dealing with a supplemental inspection at the Unit 1 reactor from Feb. 13 through March 2, 2012. The inspection stemmed from unplanned scrams (plant shutdowns) in 2010 and early 2011, and an internal flooding incident in the third quarter of 2010 that resulted in a white finding from the NRC of low to moderate safety significance.

In the report, the NRC said that plant licensee PPL "adequately addressed the unplanned scrams." However, the report said the plant had not made "sufficient progress on the procedure quality upgrade project for the internal flooding event for the NRC to evaluate its effectiveness."

The internal flooding event was previously discussed in NRC reports issued in Nov. 12, 2010, and Sept. 1, 2011. The incident occurred on July 16, 2010, resulting in 1 million gallons of water 12 feet deep in the Unit 1 main condenser bay The flooding caused a shutdown of the reactor for about 20 days. It was attributed to inadequate procedures in the maintenance and operation of the main condenser waterboxes and circulating water system.

The incident was part of the unplanned scrams affecting the plant. Others occurred on April 22 and May 14 of 2010, and Jan. 25, 2011.

The NRC report said PPL performed a comprehensive evaluation relating to the scrams. "Two of the four unplanned scrams were caused by inadequate performance of maintenance, and the remaining two scrams occurred during the testing of a new Integrated Control System," the report said.

In addition, the report said, PPL determined that the primary causes for the unplanned scrams were "less that adequate risk informed decision making; less than adequate problem identification and resolution, including use of the Corrective Action Process; operating experience and cause analysis; less than adequate procedure quality use and adherence; maintenance performance that was not adequate; and management oversight that provided less than adequate enforcement of standards and expectations."

Regarding the July 16, 2010, flooding event, the NRC report noted PPL completed three root cause evaluations. "The inspectors determined that PPL failed to adequately address extent of condition and extent of cause for the white finding," the NRC said. "The inspection team concluded that the corrective actions taken for extent of cause were narrow because torque checks of selected flanges of other plant equipment were not included … Consequently, the NRC was not able to effectively evaluate the robustness, adequacy and effectiveness of future actions to address extent of condition and extent of cause, including procedure quality improvements."

As a result, the NRC said the white finding will remain open to verify that "the concerns of extent of condition and extent of cause of inadequate procedures used to torque gasketed flanges are appropriately assessed and that adequate corrective actions are identified and implemented; and to verify the effectiveness of the station's procedure quality upgrade project."

As part of the report, the NRC noted that inspectors "determined that the safety conscious work environment (at the pant) is not currently degraded. Interview comments indicated that the plant staff members are not deterred from reporting safety concerns using the condition reporting system. Plant staff members interviewed consistently express an awareness of the necessity of reporting safety concerns and frequently expressed their commitment to assuring that any reported safety concerns were clearly understood."

June 19, 2012 – Operators at the Unit 1 reactor performed a planned shutdown to investigate the source of a minor water leak inside the containment structure.

A plant official said the leak does not affect the safety of the plant or the public. Unit 2 is continuing to operate at full power.

July 2, 2012 – Unit 1 at the Susquehanna power plant resumed generating electricity after repairs were made of a small water leak inside the containment structure surrounding the reactor.

Officials said a weld was repaired where the leak was found and they inspected similar equipment elsewhere to make sure there were no problems.

July 19, 2012 – The NRC completed a security inspection at Units 1 and 2 on June 15, 2012.

In a letter to the plant operators, the NRC said it identified two findings of very low security significance. "The deficiencies were promptly corrected or compensated for, and the plant was in compliance with applicable physical protection and security requirements within the scope of this inspection before the inspectors left the site," the letter said.

Details of the findings were not released. The letter said the findings involved violations of NRC requirements.

Nov. 7, 2012 – Unit 1 at the Susquehanna Steam Electric Station resumed service after completing a turbine blade inspection. PPL, the plant owner, said the inspection found signs of cracking on a small number of turbines. The blades were replaced.

PPL also said it will shut down Unit 2 for a similar inspection in the near future.

<u>Nov. 9, 2012</u> – Unit 2 at the Berwick area plant was shut down because a computer system controlling the reactor's water level was not functioning properly.

<u>Nov. 13, 2012</u> - The NRC issued a report on its third quarter inspection of Units 1 and 2 at the Susquehanna Steam Electric Station.

The report listed two NRC-identified findings and one self-revealing finding of very low safety significance.

The report also detailed a review conducted over the failure of an emergency diesel generator in December 2011., The NRC initiated an investigation at the start of 2012 to determine whether maintenance technicians and a quality control inspector "deliberately failed to property assemble delivery values on 15 fuel pumps." As a result of the investigation, the inspectors determined that the diesel generator failure was the result of "improper planning and implementation of work instructions" and not due to deliberate actions by the technicians and quality control inspector.

The NRC findings included a concern that PPL, the plant owner, "did not maintain adequate procedures to respond proactively to acts of nature." Specifically, the NRC report said, PPL's "adverse weather procedure did not ensure timely risk management activities for imminent adverse weather" despite advisories of a high wind watch and a tornado watch.

The National Weather Service had issued a high wind watch for Luzerne County from Sept. 17, 2012, through the evening of Sept. 18, 2012. A high wind advisory was issued on Sept. 19, 2102, and there also was a tornado watch for the county, the report said.

"The inspectors noted a number of items that could be potential missile hazards" such as "loose pieces of wood, loose wood blocks, wooden pallets, a wooden cable spool, stanchions, piping, piping flanges, a metal–frame door and pieces of sheet metal." Despite the wind and tornado advisories, "the inspectors observed that not all of the items the inspectors had observed were noted by PPL nor were they all removed during the PPL walkdown."

"The inspectors," the report added, "concluded that, procedurally, PPL would not take anticipatory actions until there is a confirmed tornado and that tornado has probable impact on the station. This approach was determined to be inadequate given that the touchdown of a tornado with probable impact on the station did not allot sufficient time to take preventive measures or mitigating actions and that a proactive approach to acts of nature was warranted."

The report said PPL entered this matter into its corrective action program.

The NRC's second finding indicated that PPL did not implement risk management actions during maintenance as required by station procedures. This stemmed from various activities.

"During the months of July and August 2012, there were multiple instances of inadequate implementation of risk management actions while maintenance was conducted," the report said. The NRC said the matter would be treated as a non-cited violation due to its low safety significance and because the finding was entered into PPL's corrective action program.

The self-revealing finding involved inadequate troubleshooting measures that caused repeated inoperability of secondary containment. This stemmed from an April 13, 2012, incident in which load centers were affected. The loss of the load centers "impacted secondary containment in that both reactor building heating, ventilation and air conditioning (HVAC) Zone I equipment compartment exhaust fans tripped due to the loss of power." This set off a cascade of events that rendered Unit 2 secondary containment inoperable and affected the Unit III supply fans.

After reviewing an evaluation of the problem, it was determined that "the troubleshooting plan was limited in scope due to the desire to limit interruption to refueling floor work and pose minimal risk to the operating unit's Zone III HVAC," the report said. "The troubleshooting did not identify all of the faulted heaters and PPL did not account for this by ensuring that system configuration at the time of the equipment's restoration would not result in the subsequent loss of secondary containment or protected equipment."

In a licensee-identified violation in the report, the NRC noted that PPL said a 10-meter wind direction instrument on its primary meteorological tower was inoperable on Sept. 27, 2011. However, the Nuclear Emergency Response Organzation was not notified of this problem. "From Sept. 27 through Sept. 30, 2011, PPL did not maintain an adequate method for accurately calculating dose projections and issuing publicly available records to offsite agencies,

The NRC said this matter was a green finding of low safety significance "since the capability for immediate dose projection existed via alternative meteorological towers." The matter was entered into PPL's corrective action program.

Nov. 19, 2012 – Unit 2 at the power plant resumed generating electricity after completing a turbine blade inspection and repairing a computer system that malfunctioned on Nov. 9. A previously announced turbine inspection revealed signs of cracking on a small number of blades. Those blades were replaced.

The computer system malfunction was caused by a failure of a processing unit that was replaced during the outage, PPL, the plant owner, said.

Nov. 20, 2012 - Unit 2 at the plant was shut down shortly after returning to service because of a hydraulic oil leak on a system that controls the flow of steam into the turbine, PPL said.

Nov. 29, 2012 - Unit 2 returned to service after repairs of the hydraulic system associated with the unit's main turbine. PPL, the plant owner, said officials detected leaks in the system as part of a routine inspection during startup procedures while at very low power levels.

Dec. 14, 2012. The inspection focused on an evaluation of changes, tests or experiments, and permanent plant modifications.

No findings were identified in the inspection, the NRC said.

Dec. 14, 2012 – The NRC approved an exemption allowing the owner of the plant to postpone its biennial emergency preparedness exercise from Oct. 23, 2012, to Feb. 26, 2013.

Plant owner PPL requested the exemption due to an unplanned Unit 1 outage due to cracking experienced on some turbine blades (discussed in previous NRC reports).

Dec. 16, 2012 – Unit 2 at the nuclear power plant shut down automatically during routine testing of a valve on the unit's main turbine system. Operators were investigating why the testing caused a shutdown.

Dec. 28, 2012 – Unit 2 at the nuclear power plant resumed generating electricity after its Dec. 16, 2012, shutdown.

Operators said an electrical connection problem caused the shutdown during a routine valve test. "An unrelated issue with the positioning of a valve on one of the unit's main water pumps during start-up activities extended the out-of-service time," plant owner PPL said.

Jan. 25, 2013 – The NRC issued a follow-up supplemental inspection report relating to a July 16, 2010, internal flooding incident at Unit 1 of the Susquehanna Steam Electric Station.

The NRC had issued two previous reports on the incident, one in late 2010 and another on May 7, 2012. The NRC had issued a white finding of low to moderate importance to safety.

The flooding incident, totaling 1 million gallons of water 12 feet deep in the main condenser bay, was one of four unplanned scrams (plant shutdowns) in 2010 and early 2011. In its May 2012 report, the NRC noted that plant owner PPL had not made sufficient progress stemming from the flooding incident.

NRC inspectors returned to the site in late November 2012 and "determined that PPL's extent of condition reviews and progress on the procedure upgrade project were sufficient and appropriate to address the identified significant weakness as documented during the initial supplemental inspection report." Because of this, the NRC determined the inspection objectives were satisfied and the white finding was closed.

"The inspectors determined there was adequate and reasonable progress accomplished on the procedure upgrade project since April 2012, especially when considering the number of potential distractions posed by planned and unplanned plant shutdowns," the NRC report said. "Based on review of condition reports and personnel interviews, the inspectors determined PPL personnel have checked and adjusted the upgraded procedure progress based on initial implementation learnings and station personnel feedback," the report added. "The inspectors concluded completed upgraded procedures are of good quality with positive station response."

The flooding incident occurred when a manway gasket rolled out of position, the result of inadequate maintenance procedures. While PPL addressed the direct cause of the flooding incident, the NRC previously noted that PPL's assessment was narrowly focused because the company "did not include a sampling of other gaskets that could have been similarly affected by inadequate maintenance procedures." Those issues were satisfactorily addressed in the latest NRC report.

Feb. 13, 2013 – The NRC issued its report of a quarterly inspection for the last three months of 2012. In the report, the NRC observed three findings of very low safety significance and two Severity Level IV violations that were also viewed of very low safety significance and treated as non-cited violations.

In addition, the report detailed problems with timely notification and management oversight regarding medical conditions of licensee employees.

The non-Level IV violations involved a failure to timely notify some emergency agencies during a emergency preparedness drill; improper valuation of a stress fabrication factor that resulted in a weld failure in June 2012; and not properly classifying a functional failure of the Unit 2 125 Volt Direct Current system on Nov. 23, 2011.

The emergency drill occurred on Nov. 13, 2012. An unusual event was declared in the drill at 8:28 a.m. Attempts to contact the offsite response organizations - Pennsylvania Emergency Management Agency, Luzerne County Emergency Management Agency and Columbia County Emergency Management Agency - were initially unsuccessful because the "phone had no dial tone," the NRC report said. Some connectivity was subsequently restored, but two of the three emergency response organizations were not notified within the 15 minutes as required after declaration of an unusual event. Moreover, the NRC observed the post-drill condition report made no mention that the two agencies were not notified within 15 minutes of the declared emergency or that "equipment performance or controller intervention potentially interfered with adequate observation of emergency response organization performance."

The report added, plant licensee PPL "did not identify that timely notification was not made with two of the off-site response organizations as required by regulatory requirement and the (plant's emergency plan). Additionally, PPL evaluated a performance indicator opportunity as a success despite drill controller action precluding satisfactory observation of emergency response organization performance."

The NRC noted that PPL entered the drill critique deficiency into its corrective action program, and the matter was treated as a non-cited violation.

The weld failure involved a unexpected increase in the drywell leak rate and a shutdown of Unit 1 on June 19, 2012. The problem stemmed from improper stress calculations dating to 2004.

"From 2004 until June 19, 2012," the NRC report said, "PPL failed to accurately translate design basis requirements to ensure Unit 1 reactor coolant system piping systems met American Society of Mechanical Engineers core requirement to pipe stress analysis calculations … due to using an incorrect stress intensification factor," the report said. "The weld in question subsequently failed, resulting in pressure boundary leakage in excess of technical specification limits from June 16 to June 18, 2012.

The report said PPL acted to make repairs to the piping. The matter was treated as a non-cited violation because of its very low safety significance and because the finding was entered into PPL's corrective action program.

The other low-level violation involved the failure of PPL staff to demonstrate that performance of the Unit 2 125 Volt Direct Current was being effectively controlled through appropriate preventive maintenance. "Specifically," the report said, "PPL staff did not property classify a functional failure of the … system on Nov. 23, 2011, as maintenance preventable until prompted by questions from inspectors." The issue also was treated as a non-cited violation.

Among the Security Level IV issues, NRC inspectors identified a failure of PPL to submit an event report dealing with electrical power monitoring associated with several Unit 1 reactor protection system breakers on May 8, 2012. The report is to be submitted within 60 days. The report said "PPL personnel had determined that the event was not reportable because it did not result in a loss of safety function or condition prohibited by plant technical specifications."

But the NRC noted that plant licensees must submit an event report for "any event where a single cause or condition caused two independent training of channels to become inoperable in a single system designed to shut down the reactor within 60 days of discovering the event." Despite this, PPL did not submit a report within the allotted time period. The NRC said it was treating the mater as a non-cited violation, and it was entered into PPL's corrective action program.

The other Level IV violation involved a failure of PPL to notify authorities within eight hours of a valid actuation of the Unit 2 reactor protection system on Nov. 9, 2012. On that date, Unit 2 at the facility was manually scrammed (shut down) following a failure in the integrated control system and a subsequent lowering of reactor water level.

A few hours after this action, an automatic scram was generated. The NRC said PPL submitted a report within the required four hours of the original scram, but questioned whether PPL operators made a report within the required eight hours after the second scram.

The NRC said the issue was of very low safety significance, was not repetitive or willful, and was entered into PPL's correction action program. It was treated as a non-cited violation.

The report also addressed other issues involving notification deficiencies at the plant. The report said PPL staff became an investigation in February 2012 "in response to a series of NRC findings from 2007 to present involving required NRC notifications not being made that affect license conditions of licensed operators." As a result of the review, PPL submitted on July 20, 2012, 10 medical updates to the NRC, four of them permanent changes in medical conditions that were "not submitted in a timely manner as required."

"Over a period of four years, a number of licensed operators developed potentially disqualifying medical conditions that were not property evaluated by PPL" in accordance with requirements, the report said. "In addition, during this same time frame, there were a number of cases (i.e., both historical and current) where PPL potentially failed to notify the NRC of a change in medical condition within 30 days" as required.

Based on the PPL review, the problems "appear to be associated with PPL's failure to properly train and provide oversight for their medical review officer and the Berwick examining physician regarding compliance with the requirements," the NRC report said. "The medical issues identified during this time frame appear to be related to a lack of knowledge and inadequate oversight."

The report added, "The inspectors concluded that PPL's failure to properly identify potentially disqualifying medical conditions resulted in failure to notify the NRC of these changes in medical conditions within 30 days, and in some cases may have affected the operator's ability to comply with operator license conditions that should have been in effect while standing watch. This was a performance deficiency within PPL's ability to foresee and correct and should have been prevented. The NRC has issued conditioned individual operator licensees which address the potentially disqualifying conditions for the operators."

The NRC said this was an unresolved issue.

<u>Feb. 25, 2013</u> – The NRC issued a report covering a two-week inspection completed on

March 4, 2013 - In an annual assessment letter for 2012, the NRC said it found that Unit 1 was within the regulatory response column of the NRC's Reactor Oversight Process because of one finding having low to moderate safety significance that was related to an internal flooding event on July 16, 2010. Unit 1 began the assessment period in the Degraded Cornerstone Column due to this finding and due to unplanned shutdowns per 7,000 critical hours. On May 7, 2012, the NRC issued an interim response that closed the finding related to the unplanned scrams, or shutdowns. The other finding was closed in early 2013, moving Unit 1 to the licensee response column.

For Unit 2, the NRC determined during the most recent quarter that the plant was within the licensee response column because all inspection findings had very low safety significance.

The NRC also issued a concern over cross-cutting issues, and said this matter will remain open until PPL (the plant licensee) "has demonstrated sustainable performance improvement as evidenced by effective implementation of an appropriate corrective action plan that results in no safety significant findings and a notable reduction in the overall number of inspection findings with the same cross-cutting aspect."

The NRC said this was the fourth consecutive assessment letter documenting "substantive" cross-cutting issues.

May 9, 2013 – Operators at the power plant disconnected Unit 1 from the regional power grid as part of a scheduled outage to install turbine modifications.

May 14, 2013 – The NRC issued a quarterly report for the first three months of 2013. In the report, the inspectors identified four findings of very low safety significance, and two severity level IV non-cited violations, one of them associated with one of the four findings.

The findings include plant licensee PPL's incorrect implementation of the clearance process while returning the common off-gas recombiner to service after maintenance; PPL's failure to accurately report unplanned scrams (plant shutdowns) with complications for the period of October 2012 through December 2012; storage of transient combustibles in restricted areas without evaluations by the site fire protection group; and failure of PPL to ensure that alarm response procedures for control room cooling fan train failures were adequate.

The first finding involved a Dec. 12, 2012, incident when operators incorrectly left a manual isolation valve in the closed position for the common recombiner. Discovery of this problem was made on Feb. 4-5, 2013, when plant staff observed a steam leak on the Unit 2 off-gas recombiner. Operators reduced power at Unit 2 to 64 percent due to this problem.

The second finding stemmed from a Dec. 16, 2012, reactor scram at Unit 2 during turbine control valve testing. Inspectors reviewed PPL's reporting of the scram and determined that staff did not view the matter as “complicated” based on Nuclear Energy Institute standards. “This scram, when combined with a second complicated scram, which was accurately reported in the same quarter, caused the performance indicator to cross the green-white threshold,” the NRC said. (Green findings are the lowest, and white findings are the next lowest.) The finding also was determined to be a severity level IV violation that was treated as a non-cited violation because it was of very low safety significance, was not repetitive or willful, and was entered into PPL’s corrective action program.

The third finding involved storage of transient combustibles in restricted areas without an evaluation by site fire protection personnel. NCR inspectors found materials on Jan. 4, 2013, during a walkdown in the Unit 2 reactor building. During the walkdown, inspectors said an overhead crane and two trash cans were being stored in a restricted area. The crane and trash cans were relocated after PPL was notified. Other walkdowns uncovered improper storage of combustibles on Jan. 22, 2013 and March 14, 2013.

“PPL staff completed an apparent cause evaluation that determined there was not awareness of fire protection requirements and locations of restricted areas and that those requirements were not adequately or repeatedly stressed to plant personnel,” the NRC report said. “Based on this, inspectors determined that management and supervisory oversight was the most significant contributor to the performance issue.”

In the fourth finding, the NRC report said that “adequate instruction did not exist to align equipment in response to a tripped fan train condition and this, subsequently, resulted in the unexpected loss of both control room cooling trains during the implementation of the clearance order process,”

The other severity level IV non-cited violation involved PPL personnel making changes affecting Units 1 and 2 without obtaining a license amendment. The report said PPL approved changes to support raising the American Petroleum Institute gravity of ultra low sulfur diesel fuel oil deliveries. The NRC said such a change required a license amendment prior to implementation. “The inspectors noted the change to accept ultra low sulfur diesel with a higher specific gravity fuel oil had not yet been physically implemented because it had not been accepted for delivery prior to the inspector’s questions,’ the report said. The report added that PPL entered the matter into its corrective action program, and the issue was treated as a non-cited violation.

The report also listed two items of very low safety significance identified by PPL, the plant licensee.

<u>June, 5, 2013</u> – Unit 2 at the Susquehanna nuclear power plant resumed operations following a refueling and maintenance outage.

Workers replaced about 40 percent of the Unit 2 reactor fuel during the outage, and inspected and replaced several pieces of the unit's turbine assembly. In addition, crews replaced a 24-ton motor and pump that helps circulate coolant water through the reactor.

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 1 and 2 at the Berwick facility are affected by the directive.

June 14, 2013 – Unit 1 was returned to service after improvements were made addressing turbine issues

June 17, 2013 – The NRC issued a report on its inspection of issues relating to the proper licensing and notification procedures of some workers with medical conditions.

According to the report, the NRC said there were two apparent violations of NRC requirements. In addition, the NRC issued a green finding of very low safety significance due to failure to implement effective corrective actions.

The NRC report said one apparent violation found that four licensed operators developed disqualifying medical conditions that were not property evaluated by staff of PPL. Additionally, the report said PPL did not restrict the operators from performing licensed duties or obtain NRC approval to continue these duties by requesting conditional licenses. NRC inspectors also identified eight instances in which PPL failed to notify the NRC within 30 days after learning of changes in licensed operator medical conditions that involved performance disabilities or illness.

The second apparent violation stemmed from PPL's "failure to provide information to the NRC regarding medical examinations of licensed operators that was complete and accurate in all material respects," the NRC report said. "Specifically, PPL submitted three NRC licensed operator renewal applications and one initial license application, each of which certified the medical fitness of the applicants and that no restricting license conditions were necessary. However, the applicants, in fact, each had medical conditions that did not meet the minimum standards."

The NRC report notes that since 2008, plant licensee PPL had been issued three severity level IV violations and one severity level III violation related to the medical qualifications of its licensed operators. Because of these prior violations, the NRC said PPL staff reviewed the medical records of all of its licensed operators and submitted 10 medical updates on July 20, 2012. "Four of the 10 updates involved permanent changes in medical conditions that had not been previously submitted within the required 30 days," the NRC report said. "The other six submittals involved conditions that PPL initially stated were being provided to the NRC 'for information only.' However, the NRC independently identified … that three of these six 'information only' submittals actually involved operators with permanent changes in medical conditions. These medical conditions did not meet the minimum standards to conduct licensed activities and, therefore, the affected operators should have been removed from licensed activities, or conditions added to their licenses before being permitted to continue watch standing."

In evaluating this problem, NRC determined that PPL had not provided adequate training for the medical review officer and examining physician at Berwick Hospital., "nor did the root cause assign corrective actions to address these issues." The report noted that PPL staff assigned corrective actions to include training of the medical review officer and nurse. The training was completed in November 2012 for the medical review officer, and in December 2012 for the nurse.

July 21, 2013 – Operators disconnected Unit 1 at the power plant facility to repair one of four valves controlling the amount of steam going into the turbine. The unit was returned to service later the same day.

Aug. 14, 2013 – The NRC completed a quarterly inspection of Units 1 and 2 for the period ending June 30. In the report, the NRC identified three findings of very low safety significance. "Separately," the report added, "a violation involving a failure to set secondary containment during operations with the potential to drain the reactor vessel was identified during the Unit 2 refueling outage from April 17 to May 7, 2013, and from May 10 to May 17, 2013.

One finding involved an inadequate operability determination for a synchroscope switch failure that rendered offsite power and four emergency diesel generators inoperable. This occurred early on May 7, 2013, resulting in all four emergency diesel generators and offsite power being inoperable from May 7 through May 10, 2013. The problem was placed in the plant's corrective action program.

The second finding involved an issue with PPL, the plant owner, not adequately incorporating acceptance criteria for heatup rates during a plant startup of Unit 2 on May 28, 2013. "Heatup rate was assessed as high as 105-degrees Fahrenheit for two different periods during the plant startup," the report said. "Approximately 15 hours later, following review of the data and technical specifications (TS) basis, PPL engineering concluded that the TS limit was exceeded."

The NRC noted that during a plant startup in June 2012, inspectors questioned whether PPL was adequately incorporating the heatup rate limits as prescribed. PPL has placed the matter into its correction action program.

A third finding involved PPL staff allowing unacceptable preconditioning by performing corrective maintenance work on April 25, 2013, before recording time responses of the reactor protection system and other functions for the turbine control valve. "The failure to collect as-found data could result in the inability to verify the operability of (structures, systems and components)," the report said. "In this case, the test of the subject pressure switch had exhibited decreasing margin and inconsistent performance during its previous surveillance test." The NRC report noted that procedures state that the "performance of maintenance activities prior to a surveillance test with the intent of ensuring favorable test results is unacceptable preconditioning."

The other matter stemmed from actions from April 17 to May 17, 2013, when PPL performed operations with a potential for draining the reactor vessel without establishing a secondary containment. The NRC said it would issue no enforcement action for the violation.

Aug. 28, 2013 – The NRC decided not to impose a $70,000 fine against PPL Corp., owner of the Susquehanna nuclear power plants, despite identified violations regarding medical examinations and fitness of some workers. (See NRC report dated June 17, 2013.)

The NRC decided not to impose a fine because of corrective actions taken by PPL and because PPL had not been the subject of escalated enforcement action within the last two years.

Sept. 24, 2013 – Operators reconnected Unit 2 to the regional power grid after completing an inspection of turbines. Workers replaced a small number of turbine blades and performed other minor repairs.

Nov. 5, 2013 – The NRC updated its assessment of Unit 2 after completion of a quarterly review. The assessment related to unplanned scrams (shutdowns) at the facility.

The NRC said the third quarter review of Unit 2 "determined that the 'unplanned scrams with complications' performance indicator remained White" and that the unplanned scrams were greater that three per 7,000 critical hours over a four-quarter period.

The NRC noted that Unit 2 had unplanned scrams on Nov. 9, Dec. 16, and Dec. 19 in 2012, and Sept. 14, 2013.

Feb. 14, 2014 - The NRC issued a report of its quarterly inspection of Units 1 and 2 for the period October through December 2013. In the report, the NRC found three findings of very low safety significance treated as non-cited violations. There also was a licensee-identified violation determined to be of very low safety significance.

One finding involved procedures that could complicate an internal flooding event. Specifically, the NRC said procedures from PPL, the plant operator, "directed operators to enter a flooded room to assess the extent and source of the flooding," an action that could flood adjacent rooms. PPL entered the matter into its corrective action program.

The second finding was PPL's failure to ensure that all testing needed to demonstrate the performance of various systems was "identified and performed in accordance with written test procedures." Specifically, the NRC noted, PPL "did not ensure that secondary containment integrity was tested in all required configurations."

The third finding involved PPL's failure to have "temperature indication installed in some areas of the reactor building that are required to support assessment and determination of entry conditions into the fission product barrier emergency action levels."

The report added, "During the course of questioning, it was determined that nine of the 21 areas listed do not have installed temperature indication. Therefore, there would be no installed instrumentation to declare the appropriate emergency action level for a break that was not isolated in those rooms." PPL entered this matter into its corrective action program.

The PPL identified violation stemmed from improper authorization of hours for some senior reactor operators and reactor operators. Such personnel must perform a minimum of seven eight-hour shifts or five 12-hour shifts per calendar quarter to retain credentials. However, the NRC report said, PPL did not ensure that eight licensed senior reactor operators and two licensed reactor operators met those standards from April 1, 2010, to Dec. 31, 2013. "Specifically," the NRC report said, "the operators stood watch as members of a reactivity management team, which is not a credited shift crew position. These watches were incorrectly credited toward meeting their minimum required quarterly proficiency requirements."

The operators have been re-certified, and the plant revised its procedures "to identify the shift positions that are creditable for proficiency," the NRC report said.

The NRC said the issue matches a severity level III violation in its performance policy. "However," the report concluded, "after review of the responsibilities of the reactivity management team positions and that none of the operators were responsible for operational errors as a result of not standing the required number of proficiency watches and there were no other factors impacting their ability to hold a shift position, NRC management has determined this issue to be more appropriately evaluated as a severity level IV."

Feb. 12, 2014 – A secondary containment boundary door was found propped ajar at Unit 1 at 7:11 a.m. The last record of access to the area in question was about 45 minutes after midnight, so the potential duration of the door ajar was around 6.5 hours.

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The NRC said the issue matches a severity level III violation in its performance policy. "However," the report concluded, "after review of the responsibilities of the reactivity management team positions and that none of the operators were responsible for operational errors as a result of not standing the required number of proficiency watches and there were no other factors impacting their ability to hold a shift position, NRC management has determined this issue to be more appropriately evaluated as a severity level IV."

March 4, 2014 – The NRC issued its annual assessment of Units 1 and 2. It determined that Unit 1 "operated in a manner that preserved public health and safety and met all cornerstone objectives." It also determined that Unit 1 was within the "Licensee Response Column" of its oversight process.

As for Unit 2, the NRC determined that performance during the most recent quarter was within the "Degraded Cornerstone Column" of its oversight process. That's because there were two white performance indicators existing from events of unplanned scrams (shutdowns) in the fourth quarter of 2012 that moved Unit 2 from green (least severe) to white (more severe) category in terms of safety significance. While the plant licensee was showing progress in correcting the issue, Unit 2 "had an unplanned scram on Sept. 14, 2013, that resulted in crossing the green to white threshold…This performance indicator result, in conjunction with the earlier white performance indicator, moved Susquehanna Unit 2 to the degraded cornerstone column from the regulatory response column."

The NRC also said it planned to conduct a public meeting with the plant operator "in which we will review station performance."

The NRC added that it issued three severity level IV traditional enforcement violations associated with willfulness in 2013. The NRC said it would conduct inspection procedures to follow up on these violations.

June 25, 2014 – Operators began shutting down Unit 2 at the Susquehanna nuclear power plant to inspect the unit's turbine blades.

Officials said data from the extensive vibration monitoring equipment installed on the turbine indicate that a few blades may have developed small cracks.

Newly designed blades were recently installed at Unit 1 of the nuclear power facility. If an evaluation determines that those blades work efficiently, then similar blades will be installed on the Unit 2 turbine during its next scheduled refueling outage in the spring of 2015, the company said.

July 5, 2014 – Operators reconnected the Unit 2 reactor to the electrical grid after a shutdown to inspect some turbine blades.

The company said plant personnel replaced a number of blades and performed other maintenance activities while the plant was in shutdown mode

Aug. 1, 2014 – The NRC issued a report after completing an inspection at Units 1 and 2. In the report, the NRC noted "there were several continuing weaknesses associated with the implementation of certain aspects of (plant operator) PPL's corrective action program. Specifically, the inspectors determined that PPL did not consistently prioritize and evaluate issues commensurate with the safety significance of the identified problem:"

The report issued one notice of violation for a matter of very low safety significance, and it also reported three other findings of very low safety significance that were treated as non-cited violations.

The issue under citation found that "PPL did not follow and maintain a standard emergency classification and action level scheme. Specifically, PPL did not take timely corrective actions to provide an adequate means to measure temperature in nine out of 21 areas where reactor building temperatures are considered for the fission product barrier degradation emergency action levels." The NRC said this failure dated back to October 2003.

"The lack of installed temperatures indication had the potential to impact declaration of all four emergency classifications; however, due to the redundancy within the fission product barrier matrix, the inspectors determined that it was reasonable that a general emergency would be declared in a timely manner. The inspectors determined that the lack of installed instrumentation could result in untimely declarations of a site area emergency, alert, or unusual event."

NRC said it is citing this violation because PPL "has failed to restore compliance or demonstrate objective evidence of plans to restore compliance at the first opportunity and in a reasonable period of time following discussion in a formal exit meeting on Jan 24, 2014, and documented" in a NRC inspection report of Feb. 14, 2014.

The three non-cited violation are as follows:

- PPL's "failure to take adequate corrective action for a condition adverse to quality involving the emergency service water and residual heat removal service water systems." The NRC said PPL failed to take timely corrective action to address carbon steel pipe wall thinning. "PPL did not take timely and appropriate corrective actions to assess the corrosion, address wetting conditions, and perform an appropriate operability determination that included assessing the piping degradation rate and calculate the minimum wall thickness to ensure that structural integrity requirements were maintained, " the NRC report said. The agency noted that PPL left the matter uncorrected from November 2010 to June 2014.

- PPL's "failure to complete and document initial operability determination in a timely manner in accordance with station procedures." From May 24, 2013, to June 6, 2014, the NRC said, "PPL failed to accomplish activities affecting quality in accordance with prescribed procedures." These procedures, it said, require the completion of initial operability screening within eight hours or the end of work shift, whichever comes first.

- PPL's failure to promptly correct an issue involved with the emergency service water supply lines. "Since April 30, 2009, the NRC said, "PPL had not established measures to assure a condition adverse to quality had been corrected. Specifically, PPL had not taken measures to eliminate pipe vibration and water hammer that are causing fatigue stress in the emergency service water supply lines" to various pump motor oil coolers

Aug. 13, 2014 – The NRC issued a report of its inspection for the three-month period ending June 30, 2014. In the report, the NRC identified one non-cited violation, and noted that plant operator PPL found a violation of very low safety significance.

The NRC finding involved PPL's failure to implement timely actions "to address the extent of a previously identified inoperable condition." .

The PPL finding involved a failure to control the concentration of airborne radioactive materials during weld preparation on reactor water cleanup piping on April 27, 2014. "A radiation protection technician monitoring a continuous air monitor noticed increasing airborne radioactivity and subsequently stopped the work," the NRC report said. "This failure to use, to the extent practicable, process or engineering controls led to a worker receiving an unplanned, unintended uptake of approximately 11 millirem." The violation was entered into PPL's corrective action plan.

Sept. 6, 2014 – Operators at the plant disconnected Unit 2 from the power grid to inspect its turbine blades. Data showed that a few of the blades may have developed small cracks.

Sept. 15 – The Unit 2 reactor was reconnected to the electrical grid. During the shutdown (see Sept. 6, 2014), workers replaced one row of blades, although only a small number were found to have indications of cracking. PPL has already installed newly designed blades at Unit 1, and similar blades are to be installed at Unit 2 during the next scheduled refueling in the spring of 2015.

June 22, 2015- NRC Finalizes 'White' Inspection Finding for Susquehanna Nuclear Plant, Resulting in Additional Oversight

The Nuclear Regulatory Commission will increase its level of oversight at the Susquehanna nuclear power plant, in Salem Township (Luzerne County), Pa., as a result of the finalization of a "white" (low to moderate safety significance) inspection finding and related violation in the area of emergency preparedness. NRC inspectors, during an in-depth review of plant drill scenarios, identified a concern with how plant personnel would determine the start of a 15-minute clock for emergency assessment and declaration for a scenario involving the potential loss of primary containment. (Both of the plant's units have primary and secondary containments to prevent the release of radioactivity to the environment following an accident.) The inspectors found that Susquehanna's interpretation of the 15-minute assessment and classification period degraded plant personnel's ability to make a timely "Site Area Emergency" declaration in certain cases. (A Site Area Emergency is the third tier of the four levels of emergency classification used by the NRC.)

Specifically, the plant's owner, Susquehanna Nuclear LLC, interpreted the requirements as having the 15-minute clock begin when operator actions were, or were expected to be, unsuccessful in halting reactor coolant system leakage rather than when indications of a

leak's onset are available to plant operators, signaling that an emergency action level has been exceeded.
"It's important during an emergency situation that state, county and local officials are provided with information in a timely manner to assess the situation and implement protective actions, if warranted," NRC Region I Administrator Dan Dorman said. "While the probability of an event of this magnitude is extremely low, this finding points to a weakness in that area that the company will need to address." Prior to making a final enforcement decision, the NRC offered the company the opportunity to accept the finding without any formal response or provide additional information in a Regulatory Conference or in writing. The company submitted a written response dated May 15 in which it acknowledged the finding but stated that training and programs already in place prior to the finding would have ensured the impact of the issue would have been relatively minor.

The NRC considered the information but determined the finding was appropriately characterized as "white." The finding also involved a violation of NRC requirements regarding maintaining an emergency plan that meets federal standards. The NRC, in response to the "white" finding, will perform a supplemental inspection at the plant to ensure the company has completed a thorough root-cause evaluation of the issue and put in place effective corrective actions. Subsequent to the issuance of the preliminary "white" finding, the Susquehanna emergency action level basis was revised to correct the declaration timeliness issue

May 1, 2018 - Letter dated May 1, 2018, the Nuclear Regulatory Commission issued a letter to Senior Vice President, Bryan Hanson of Exelon Generation Company with the subject of: Susquehanna Steam Electric Station – Integrated inspection report 05000387/2018001 and 5000388/2018001

On March 31, 2018, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Susquehanna Steam Electric Station (SSES), Units 1 and 2. On April 13, 2018, the NRC inspectors discussed the results of this inspection with Derek Jones, Plant Manager, and other members of your staff. The results of this inspection are documented in the enclosed report.

No NRC-identified or self-revealing findings were identified during this inspection. NRC inspectors documented a licensee-identified violation which was determined to be of very low safety significance in this report. The NRC is 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 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 at Susquehanna.

Inspection Report – inspection dates January 1, 2018 to March 31, 2018

1. Condition Prohibited by Technical Specifications Due to a Loose Terminal Block Associated with Primary Containment Isolation Valves
2. Loss of Secondary Containment Zone 3 Due to Fan Trip

Licensee Identified Non-Cited Violation

- Violation: Susquehanna Unit 1 TS section 5.4.1 requires that "written procedures shall be implemented covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978." Susquehanna's implementing instruction NDAP-QA- 0503, General Housekeeping, Transient Material and Internal Cleanliness, Revision 45 implements aspects of the Regulatory Guide administrative procedures requirements. NDAP- QA-0503 section 6.1.5.h requires, in part, that "transient equipment shall be located such that it will not impact safety related equipment during a seismic event. Locate all items at a distance greater than the height of the item from safety related equipment." Additionally, TS 3.5.1 Action Statement I directs immediate entry into Limiting Condition for Operation (LCO) 3.0.3 if one core spray subsystem is inoperable with one low pressure coolant injection (LPCI) subsystem inoperable. LCO 3.0.3 requires action to be taken within 1 hour to place the unit in MODE 2 within 7 hours and MODE 3 within 13 hours.

    a. Contrary to the above, from December 1, 2017 to December 3, 2017, Susquehanna staged a 540 pound, ten foot long replacement pipe on 34 inch high stands within 34 inches of the safety related Unit 1, "B" Core Spray room cooler. Susquehanna concluded that the room cooler was inoperable because the pipe could have reasonably contacted and damaged the flexible conduit for the power cable to the room cooler during a seismic event. Additionally, from 7:48 a.m. on December 2, 2017 to 1:35 p.m. on December 3, 2017, maintenance was performed on the Unit 1, division 2 LPCI swing bus motor generator which rendered the division 2 LPCI system inoperable. During this time, Susquehanna did not perform the required actions of LCO 3.0.3 and remained in MODE 1.
    b. Significance/Severity Level: This violation is of very low safety significance (Green), since this finding did not represent a loss of system, a loss of function of at least a single train for greater than its TS allowed outage time, or a loss of a non-TS train.
    c. Corrective Action Reference(s): CR-2017-20227; CR-2018-01717; CR-2018-02250

May 15, 2018 - Letter dated May 15, 2018, the Nuclear Regulatory Commission issued a letter to Senior Vice President, Bryan Hanson of Exelon Generation Company with the subject of: Susquehanna Steam Electric Station, Units 1 and 2 information request for the cyber-security inspection notification to perform inspection 05000387/2018403 and 05000388/2018403

On October 15, 2018, the U.S. Nuclear Regulatory Commission (NRC) will begin a team inspection in accordance with Inspection Procedure (IP) 71130.10P "Cyber-Security," issued May 15, 2017 at your Susquehanna Steam Electric Station, Units 1 and 2 (Susquehanna). 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, *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

October 15-19, 2018, and October 29 – November 2, 2018. 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 July 23, 2018. The inspection team will review this information and, by
August 20, 2018, 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 September 17, 2018.

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, October 15, 2018.

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 Jigar Patel. We understand that our regulatory contact for this inspection is Mr. Charlie Manges of your organization.

November 19, 2018 - Letter dated November 19, 2018, the Nuclear Regulatory Commission issued a letter to Senior Vice President, Bryan Hanson of Exelon Generation Company with the subject of: Susquehanna Steam Electric Station – Evaluated emergency preparedness exercise inspection report 05000387/2018501 and 05000388/2018501

On October 19, 2018, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Susquehanna Steam Electric Station (SSES), Units 1 and 2. The NRC inspectors discussed the results of this inspection with you and members of your staff on October 30, 2018. The results of this inspection are documented in the enclosed report.

No NRC-identified or self-revealing findings were identified during this inspection. NRC inspectors documented one licensee-identified violation which was determined to be of very low safety significance in this report. The NRC is 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 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 at SSES.

Inspection results – licensee identified non-cited violation

1. Violation: 10 CFR 50.54(q)(2) requires, in part, that a licensee shall follow and maintain the effectiveness of an emergency plan that meets the requirements in Appendix E to this Part and, for nuclear power reactor licensees, the planning standards of §50.47(b). 10 CFR 50.47(b)(4) requires, in part, that a standard emergency classification and action level (EAL) scheme is in use by the licensee.
    a. Contrary to the above, from December 2016 to the present, Susquehanna did not have sufficient guidance contained in procedures to assess the availability of the main condenser to support the containment barrier such that a site area emergency would be consistently declared in a timely manner upon loss of two fission product barriers.
    b. Significance/Severity Level: The inspectors assessed the significance of the finding using Inspection Manual Chapter 0609, Appendix B. The inspectors determined that this finding was similar to the example in Table 5.4-1, Significance Examples §50.47(b)(4), which states "[a]n EAL has been rendered ineffective such that any Site Area Emergency would not be declared for a particular off-normal event, but because of other EALs, an appropriate declaration could be made in a degraded manner (e.g., delayed)." Thus, the inspectors determined that the finding was of very low safety significance (Green).
    c. Corrective Action Reference: CR-2018-14650

January 3, 2019 - Letter dated January 3, 2019, the Nuclear Regulatory Commission issued a letter to Senior Vice President, Bryan Hanson of Exelon Generation Company with the subject of: Susquehanna Steam Electric Station, Units 1 and 2 – safety evaluation regarding implementation of hardened containment vents capable of operation under severe accident conditions related to order EA-13-109 (CAC Nos. MF4364 and MF4365; EPID No. L-2014-JLD-0055)

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 26, 2014 (ADAMS Package Accession No. ML14178A619), Susquehanna Nuclear, LLC (the licensee) submitted its Phase 1 OIP for Susquehanna Steam Electric Station, Units 1 and 2 (SSES, Susquehanna) 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 Susquehanna, including the combined Phase 1 and Phase 2 OIP in its letter dated December 23, 2015 (ADAMS Accession No. ML15362A528). 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 April 1, 2015 (Phase 1) (ADAMS Accession No. ML15090A300), August 25, 2016 (Phase 2) (ADAMS Accession No. ML16231A509), and October 5, 2017 (ADAMS Accession No. ML17272A733), the NRC issued Interim Staff Evaluations (ISEs) and an audit report, respectively, on the licensee's progress. By letter dated June 26, 2018 (ADAMS Accession No. ML18179A221), the licensee reported that Susquehanna, Units 1 and 2 are in full compliance with the requirements of Order EA-13-109 and submitted a Final Integrated Plan (FIP) for Susquehanna, which was supplemented by letter dated November 27, 2018 (ADAMS Accession No. ML18332A263).

The enclosed safety evaluation provides the results of the NRC staff's review of Susquehanna's hardened containment vent design and water management strategy for Susquehanna. The intent of the safety evaluation is to inform Susquehanna 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 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 26, 2014 [Reference 2], Susquehanna Nuclear, LLC (the licensee) submitted a Phase 1 Overall Integrated Plan (OIP) for Susquehanna Steam Electric Station, Units 1 and 2 (SSES, Susquehanna) in response to Order EA-13-109. By letters dated December 23, 2014 [Reference 3], June 23, 2015 [Reference 4], December 23, 2015 (which included the combined Phase 1 and Phase 2 OIP) [Reference 5], June 29, 2016 [Reference 6], December 19, 2016 [Reference 7], June 15, 2017 [Reference 8], and December 12, 2017 [Reference 9], the licensee submitted 6-month updates to its OIP. By letters dated May 27, 2014 [Reference 10], and August 10, 2017 [Reference 11], 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 NRG Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC- 111, "Regulatory Audits" [Reference 12]. By letters dated April 1, 2015 (Phase 1) [Reference 13], August 25, 2016 (Phase 2) [Reference 14], and October 5, 2017 [Reference 15], the NRG issued Interim Staff Evaluations (ISEs) and an audit report, respectively, on the licensee's progress. By letter dated June 26, 2018 [Reference 16], the licensee reported that full compliance with the requirements of Order EA-13-109 was achieved and submitted its Final Integrated Plan (FIP), which was supplemented by letter dated November 27, 2018 [Reference 17].

Safety Evaluation 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 April 1, 2015 [Reference 13], an ISE for implementation of Phase 2 requirements on August 25, 2016 [Reference 14], and an audit report on the licensee's responses to the ISE open items on October 5, 2017 [Reference 15]. The licensee reached its final compliance date on April 30, 2018 and has declared in letter dated June 26, 2018 [Reference 16] that Susquehanna Steam Electric Station, Units 1 and 2 are 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-13-1 09.

February 8, 2019 - Email dated February 8, 2019 from Jennifer Tobin, Project Manager, Office of Nuclear Reactor Regulation, US Nuclear Regulatory Commisson to Kevin Cimorelli with the subject of: Acceptance Review: Susquehanna Units 1 and 2 License Amendment Request for Emergency Service Water

Subject of Letter: SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 – ACCEPTANCE OF REQUESTED LICENSING ACTION RE: LICENSE AMENDMENT REQUEST TO REVISE EMERGENCY SERVICE WATER PIPING TECHNICAL SPECIFICATIONS DURING REPLACEMENT (EPID: L-2019-LLA-0004)

By letter dated January 9, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19009A431), Susquehanna Nuclear, LLC (the licensee) submitted a license amendment request (LAR) for Susquehanna Steam Electric Station (SSES), Units 1 and 2. The proposed LAR would allow temporary changes to TS 3. 7.1, "Residual Heat Removal Service Water (RHRSW) System and the Ultimate Heat Sink (UHS)," and TS 3.7.2,

"Emergency Service Water (ESW) System." Additionally, Susquehanna is proposing an administrative change to the TS Table of Contents (TOC). The proposed amendment would permit one division of the ESW and RHRSW systems to be inoperable for a total of 14 days to address piping degradation. The proposed amendment would also remove the TOC from the TS and place it under licensee control.

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 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 (including the TSs) 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 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 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

licensing request will take approximately 424 hours to complete. The NRC staff expects to complete this review in approximately 9 months, which is October 2019. 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.

March 4, 2019 – Letter dated March 4, 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 ANNUAL ASSESSMENT LETTER FOR SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 (REPORTS 05000387/2018006 AND 05000388/2018006)

The U.S. Nuclear Regulatory Commission (NRC) has completed its end-of-cycle performance assessment of Susquehanna Steam Electric Station (Susquehanna), Units 1 and 2, reviewing performance indicators (PIs), 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 Susquehanna, Units 1 and 2 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 your facility.

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 conduct Temporary Instruction 2515/191, "Inspection of Licensee's Responses to Order EA-12-049, EA-12-051, and Emergency Preparedness Info Request," in June 2019; Inspection Procedure 60855.1, "Operation of an ISFSI at Operating Plant," in September 2019; and Inspection Procedure 71003, "Post Approval Site Inspection for License Renewal," (Unit 1) in April 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
4th Quarter Site Support 10						
1, 2	10/01/2018	02/12/2019	IP 71152	000748	Problem Identification and Resolution	
1, 2	10/01/2018	02/12/2019	IP 71153	000749	Followup of Events and Notices of Enforcement Discretion	
ACCESS CONTROL, EQUIP PERF, TRAINING 3						
1, 2	02/11/2019	02/15/2019	IP 71130.02	000734	Access Control	
1, 2	02/11/2019	02/15/2019	IP 71130.04	000736	Equipment Performance, Testing, and Maintenance	
1, 2	02/11/2019	02/15/2019	IP 71130.07	000739	Security Training	
1, 2	02/11/2019	02/15/2019	IP 71151	001338	Performance Indicator Verification	
HP 1						
1, 2	02/25/2019	03/01/2019	IP 71124.01	000725	Radiological Hazard Assessment and Exposure Controls	
1, 2	02/25/2019	03/01/2019	IP 71124.02	000726	Occupational ALARA Planning and Controls	
INSERVICE INSPECTION - UNIT 2 1						

2 03/31/2019 03/31/2019 IP 71111.08G 000701 Inservice Inspection Activities (BWR)
HP 1
1, 2 04/08/2019 04/12/2019 IP 71124.01 000725 Radiological Hazard Assessment and Exposure Controls
1, 2 04/08/2019 04/12/2019 IP 71124.02 000726 Occupational ALARA Planning and Controls
REMP 1
1, 2 05/20/2019 05/24/2019 IP 71124.07 000731 Radiological Environmental Monitoring Program
TI-191 FUKUSHIMA LESSONS-LEARNED 3
1, 2 06/24/2019 06/28/2019 TI 2515/191 000509 Inspection of Licensee's Responses to Order EA-12-049, EA-12-051 & EP Info Request March 12, 2012
SUSQUEHANNA INITIAL OL EXAM 4
1, 2 07/28/2019 08/02/2019 OV 000956 VALIDATION OF INITIAL LICENSE EXAMINATION (OV)
1, 2 08/25/2019 09/06/2019 EXAD 000500 LICENSE EXAM ADMINISTRATION (EXAD)
ISFSI OPERATIONAL INSPECTION 2
1, 2 09/09/2019 09/13/2019 IP 60855.1 000590 Operation of an Independent Spent Fuel Storage Installation at Operating Plants
DESIGN BASES ASSURANCE INSP - TEAMS 6
1, 2 09/15/2019 09/21/2019 IP 71111.21M 000713 Design Bases Assurance Inspection (Teams)

This report does not include INPO and OUTAGE activities.
This report shows only on-site and announced inspection procedures.

Susquehanna

01/01/2019 - 12/31/2020

IP 22 Inspection Activity Plan Report

Unit	Start	End	Activity	CAC	Title	Staff Count
DESIGN BASES ASSURANCE INSP - TEAMS 6						
1, 2	09/29/2019	10/05/2019	IP 71111.21M	000713	Design Bases Assurance Inspection (Teams)	
EP PROGRAM INSPECTION 2						
1, 2	10/21/2019	10/25/2019	IP 71114.02	000717	Alert and Notification System Testing	
1, 2	10/21/2019	10/25/2019	IP 71114.03	000718	Emergency Response Organization Staffing and Augmentation System	
1, 2	10/21/2019	10/25/2019	IP 71114.05	000720	Maintenance of Emergency Preparedness	
1, 2	10/21/2019	10/25/2019	IP 71151	001397	Performance Indicator Verification	
HP 1						
1, 2	11/04/2019	11/08/2019	IP 71124.01	000725	Radiological Hazard Assessment and Exposure Controls	
1, 2	11/04/2019	11/08/2019	IP 71124.02	000726	Occupational ALARA Planning and Controls	

1, 2 11/04/2019 11/08/2019 IP 71124.03 000727 In-Plant Airborne Radioactivity Control and Mitigation						
1, 2 11/04/2019 11/08/2019 IP 71124.04 000728 Occupational Dose Assessment						
1, 2 11/04/2019 11/08/2019 IP 71124.05 000729 Radiation Monitoring Instrumentation						
1, 2 11/04/2019 11/08/2019 IP 71151 000746 Performance Indicator Verification						
SQ REQUAL INSP WITH P/F RESULTS 2						
1, 2 11/17/2019 11/22/2019 IP 71111.11A 000703 Licensed Operator Requalification Program and Licensed Operator Performance (Annual)						
1, 2 11/17/2019 11/22/2019 IP 71111.11B 000704 Licensed Operator Requalification Program and Licensed Operator Performance (Biennial)						
Radwaste 1						
1, 2 03/09/2020 03/13/2020 IP 71124.08 000732 Radioactive Solid Waste Processing and Radioactive Material Handling, Storage, and Transportation						
License Renewal Phase 1 - Unit 1 1						
1 04/05/2020 04/11/2020 IP 71003 000687 Post-Approval Site Inspection for License Renewal						
ISI - UNIT 1 1						
1 04/05/2020 04/11/2020 IP 71111.08G 000701 Inservice Inspection Activities (BWR)						
HP 1						
1, 2 04/13/2020 04/17/2020 IP 71124.01 000725 Radiological Hazard Assessment and Exposure Controls						
1, 2 04/13/2020 04/17/2020 IP 71124.02 000726 Occupational ALARA Planning and Controls						

This report does not include INPO and OUTAGE activities.
This report shows only on-site and announced inspection procedures.

Susquehanna

01/01/2019 - 12/31/2020

IP 22 Inspection Activity Plan Report

Unit	Start	End	Activity	CAC	Title	Staff Count
Access Control, Protective Strategy, TSR 4						
1, 2 06/15/2020 06/19/2020 IP 71130.02 000734 Access Control						
1, 2 06/15/2020 06/19/2020 IP 71130.05 000737 Protective Strategy Evaluation						
1, 2 06/15/2020 06/19/2020 IP 71130.14 000743 Review of Power Reactor Target Sets						
1, 2 06/15/2020 06/19/2020 IP 71151 001338 Performance Indicator Verification						
HP 1						
1, 2 07/06/2020 07/10/2020 IP 71124.01 000725 Radiological Hazard Assessment and Exposure Controls						
1, 2 07/06/2020 07/10/2020 IP 71124.02 000726 Occupational ALARA Planning and Controls						
1, 2 07/06/2020 07/10/2020 IP 71124.03 000727 In-Plant Airborne Radioactivity Control and Mitigation						
1, 2 07/06/2020 07/10/2020 IP 71124.04 000728 Occupational Dose Assessment						
1, 2 07/06/2020 07/10/2020 IP 71124.05 000729 Radiation Monitoring Instrumentation						
PI&R Biennial Team Inspection 4						
1, 2 07/13/2020 07/17/2020 IP 71152B 000747 Problem Identification and Resolution						

1, 2 07/27/2020 07/31/2020 IP 71152B 000747 Problem Identification and Resolution
TRIENNIAL FIRE PROTECTION 4
1, 2 09/21/2020 09/25/2020 IP 71111.05T 000696 Fire Protection (Triennial)
1, 2 10/05/2020 10/09/2020 IP 71111.05T 000696 Fire Protection (Triennial)
RETS 1
1, 2 10/05/2020 10/09/2020 IP 71124.06 000730 Radioactive Gaseous and Liquid Effluent Treatment
SUSQUEHANNA EP EXERCISE INSPECTION 5
1, 2 10/19/2020 10/23/2020 IP 71114.01 000716 Exercise Evaluation
1, 2 10/19/2020 10/23/2020 IP 71151 001397 Performance Indicator Verification
HP 1
1, 2 12/14/2020 12/18/2020 IP 71124.01 000725 Radiological Hazard Assessment and Exposure Controls
1, 2 12/14/2020 12/18/2020 IP 71124.02 000726 Occupational ALARA Planning and Controls
1, 2 12/14/2020 12/18/2020 IP 71124.03 000727 In-Plant Airborne Radioactivity Control and Mitigation
1, 2 12/14/2020 12/18/2020 IP 71124.04 000728 Occupational Dose Assessment
1, 2 12/14/2020 12/18/2020 IP 71124.05 000729 Radiation Monitoring Instrumentation

This report does not include INPO and OUTAGE activities.
This report shows only on-site and announced inspection procedures.

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March 5, 2019 – Letter dated March 5, 2019 from Jennifer C. Tobin, Project Manager Plant Licensing Branch I Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to Kevin Cimorelli Site Vice President Susquehanna Nuclear with a subject of SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 - CORRECTION TO TECHNICAL SPECIFICATIONS PAGE 3.3-45 (UNIT 1) AND PAGE 3.6-11 (UNIT 2) FOR ADMINISTRATIVE ERRORS INTRODUCED IN THE ISSUANCE OF AMENDMENT NOS. 271 AND 253

On September 26, 2018 (Agencywide Documents Access and Management System Package Accession No. ML18222A203), the U.S. Nuclear Regulatory Commission (NRC) issued Amendment Nos. 271 and 253 to Renewed Facility Operating License Nos. NPF-14 and NPF-22 for the Susquehanna Steam Electric Station (Susquehanna), Units 1 and 2, respectively. The amendments revised the Susquehanna Technical Specifications (TSs) to adopt the NRG-approved Technical Specifications Task Force Traveler (TSTF)-542, Revision 2, "Reactor Pressure Vessel Water Inventory Control."

Subsequent to the issuance of these amendments, the licensee notified the NRC by telephone on February 13, 2019, that administrative errors were made on Susquehanna, Unit 1, TS
page 3.3-45, and Unit 2, TS page 3.6-11, as follows:

- For Unit 1, TS page 3.3-45, Amendment No. 254 was added to the bottom right of the page in the list of stricken amendment numbers. Number 254 should not have been added and is now removed in the attached corrected page.
- For Unit 2, TS page 3.6-11, a separating line between paragraphs D. and E. was omitted and is now added in the attached corrected page.

The NRC staff has determined that these errors were made inadvertently. The corrections do not change any of the conclusions associated with the issuance of Amendment Nos. 271 and 253, and do not affect the associated notice to the public. The inadvertent changes were neither addressed in the notice for the amendments nor reviewed as part of the license amendment request.

April 16, 2019 – Email dated April 16, 2019 from Jennifer Tobin to Kevin Cimorelli of Talen Energy cc Melisa Krick, Jason Jennings, Shane Jurek and Tanya Hood with a subject of Acceptance Review: Susquehanna Units 1 and 2 License Amendment Request for TSTF-439 (EPID: L-2019-LLA- 0066)

SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 – ACCEPTANCE OF REQUESTED LICENSING ACTION RE: LICENSE AMENDMENT REQUEST TO ADOPT TSTF-439, “ELIMINATE SECOND COMPLETION TIMES LIMITING TIME FROM DISCOVERY OF FAILURE TO MEET AN LCO" (EPID: L-2019-LLA-0066)

By letter dated March 28, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19087A208), Susquehanna Nuclear, LLC (the licensee) submitted a license amendment request (LAR) for Susquehanna Steam Electric Station (SSES), Units 1 and 2. The proposed LAR would eliminate second Completion Times limiting time from discovery of failure to meet a Limiting Condition for Operation (LCO). The proposed amendment is consistent with previously NRC-approved TS Task Force (TSTF) Traveler TSTF-439, Revision 2, "Eliminate Second Completion Times Limiting Time from Discovery of Failure to Meet an LCO."

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 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 (including the TSs) 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 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 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 licensing request will take approximately 200 hours to complete. The NRC staff expects to complete this review within the normal 12 months, which is April 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.

May 21, 2019 – Letter dated May 21, 2019 from Jennifer Tobin, Project Manager Plant Licensing Branch I Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to Susquehanna Nuclear, LLC with a subject of SUMMARY OF MAY 7, 2018, PRE-SUBMITTAL MEETING WITH SUSQUEHANNA NUCLEAR, LLC TO DISCUSS A POTENTIAL LICENSE AMENDMENT REQUEST RE: USE OF ATRIUM 11 FUEL (EPID L-2019-LRM-0022)

On May 7, 2019, a pre-submittal meeting was held between the U.S. Nuclear Regulatory Commission (NRC) staff and representatives of Susquehanna Nuclear, LLC (Susquehanna or the licensee) at NRC Headquarters, One White Flint North, 11555 Rockville Pike, Rockville, Maryland. The purpose of the meeting was to discuss a proposed license amendment request to allow the use of Atrium 11 fuel for the Susquehanna Steam Electric Station, Units 1 and 2.

The licensee's presentation material can be found in the Agencywide Documents Access and Management System at Accession No. ML19099A016.

During the meeting, the licensee presented information regarding the schedule to submit, review, and approve the license amendment request. Specifically, Susquehanna plans to submit a license amendment request to revise the list of approved methodologies in Technical Specification 5.6.5, "Reporting Requirements - Core Operating Limits Report (COLR)," to reference Advanced Framatome Methodologies in order to support loading of Framatome Atrium 11 fuel in 2021 and 2022, respectively. Staff focused discussion on the transition to advanced Framatome methodologies, including adopting Technical Specifications Task Force (TSTF) Traveler TSTF-535, "Revise Shutdown Margin

Definition to Address Advanced Fuel Designs," and removal of two analysis penalties that are no longer applicable.

Questions from the U.S. Nuclear Regulatory Commission (NRC) staff concentrated on:

- Differences between the Atrium 10 fuel currently in use at Susquehanna and the Atrium 11 fuel the licensee plans to use going forward. The questions specifically concentrated on debris protection features and thermal hydraulic properties of the new fuel compared to the existing fuel.
- How Susquehanna and Framatome intend to model the cores. The analyses prepared for the amendment will use an equilibrium cycle of only Atrium 11 fuel. The reload specific analyses, which will be submitted to the NRC for information during the application process as they become available, will document the behavior of the mixed cores (i.e., fresh Atrium 11 and once- or twice-burned Atrium 10).
- Changes to the fuel property models used in the stability method currently employed by Susquehanna and their comparison to the fuel property models used in the application for advanced Framatome methodologies the NRC currently has under review (Brunswick Steam Electric Plant).
- Impacts on accident analyses, specifically, dose consequences thereof.

No regulatory decisions were made during the meeting.

July 16, 2019 – Letter dated July 16, 2019 from Jennifer Tobin, Project Manager Plant Licensing Branch I Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to Kevin Cimorelli Site Vice President Susquehanna Nuclear with a subject of SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 273 AND 255 RE: ADOPT TSTF-439, REVISION 2, "ELIMINATE SECOND COMPLETION TIMES LIMITING TIME FROM DISCOVERY OF FAILURE TO MEET AN LCO" (EPID L-2019-LLA-0066)

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 273 to Renewed Facility Operating License No. NPF-14 and Amendment No. 255 to Renewed Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station (Susquehanna), Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated March 28, 2019.

The changes revise TS Section 1.3 to alter the discussion contained in Example 1.3-3 to eliminate second completion times. Consistent with these changes, the second completion times associated with TS 3.8.1, "AC [Alternating Current] Sources - Operating," Required Actions A.3 and 8.4, and TS 3.8.7, "Distribution Systems - Operating," Required Actions A.1 and 8.1, are deleted. The changes are consistent with Technical Specifications Task Force (TSTF} Traveler TSTF-439, Revision 2, "Eliminate Second Completion Times Limiting Time from Discovery of Failure to Meet an LCO [Limiting Condition for Operation]," dated June 20, 2005.

A copy of the related safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's Biweekly *Federal Register* Notice.

SUSQUEHANNA NUCLEAR, LLC
ALLEGHENY ELECTRIC COOPERATIVE, INC. DOCKET NO. 50-387
SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1 AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 273 Renewed License No. NPF-14

1. The U.S. Nuclear Regulatory Commission (NRC or the Commission) has found that:

   1. The application for the amendment filed by Susquehanna Nuclear, LLC, dated March 28, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 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.
   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. NPF-14 is hereby amended to read as follows:

      (2) Technical Specifications and Environmental Protection Plan

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 273, and the Environmental Protection Plan contained in Appendix 8 are hereby incorporated in the license. Susquehanna Nuclear, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days.

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 Susquehanna Steam Electric Station, Units 1 and 2 – integrated inspection report 05000387/2019002 and 05000388/2019002

On June 30, 2019, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Susquehanna Steam Electric Station, Units 1 and 2. On July 11, 2019, the NRC inspectors discussed the results of this inspection with you and other members of your staff. The results of this inspection are documented in the enclosed report.

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 an integrated inspection at Susquehanna Steam Electric Station, Units 1 and 2 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 19, 2019 – email dated August 19, 2019 from Tanya Hood to Kevin Cimorelli (Talen Energy) cc Melisa Krick and Shane Jurek with a subject of Acceptance Review: Susquehanna Units 1 and 2 License Amendment Request for TSTF-535 with a subject of SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 – ACCEPTANCE OF REQUESTED LICENSING ACTION RE: LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT TSTF-535, REVISION 0, "REVISE SHUTDOWN MARGIN DEFINITION TO ADDRESS ADVANCED FUEL DESIGNS." (EPID: L-2019- LLA-0154)

By letter dated July 15, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19196A270), Susquehanna Nuclear, LLC (the licensee) submitted a license amendment request (LAR) for Susquehanna Steam Electric Station (SSES), Units 1 and 2. The proposed LAR would revise the Technical Specification definition of "Shutdown Margin" (SDM) to require calculation of the SDM at a reactor moderator temperature of 68°F or a higher temperature that represents the most reactive state throughout the operating cycle. The proposed changes are based on Technical Specification Task Force (TSTF) Traveler TSTF-535, Revision 0, "Revise Shutdown Margin Definition to Address Advanced Fuel Designs."

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 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 (including the TSs) 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 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 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 licensing request will take approximately 176 hours to complete. The NRC staff expects to complete this review in approximately 6 months, which is February 29, 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.

November 12, 2019 – Letter dated November 12, 2019 from Mel Gray, Chief Engineering Branch 1 Division of Reactor Safety to Brad Berryman President and Chief Nuclear Officer Susquehanna Nuclear with the subject of Susquehanna Steam Electric units 1 and 2 – design basis assurance inspection (teams) inspection report 050 On

October 10, 2019, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Susquehanna Steam Electric Station, Units 1 and 2 and discussed the results of this inspection with Mr. Kevin Cimorelli, Site Vice President and other members of your staff. The results of this inspection are documented in the enclosed report.

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 design basis assurance inspection (teams) inspection at Susquehanna Steam Electric Station, Units 1 and 2 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.

November 18, 2019 – Letter dated November 18, 2019 from Sujata Goetz, Project Manager, Plant licensing Branch 1, Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to Talen Energy with subject line of: Summary of the October 29, 2019 meeting with Talen Energy regarding a future license amendment request related to revising the dose consequence analysis for a loss of coolant accident at Susquehanna Steam Electric Station, Units 1 and 2 (EPID L-2019-LRM-0069).

On October 29, 2019, a Category 1 public teleconference meeting was held between the U.S. Nuclear Regulatory Commission (NRC) staff and the representatives of Talen Energy (the licensee) at NRC Headquarters, One White Flint North, 11555 Rockville Pike, Rockville, Maryland. The purpose of the meeting was to discuss a future license amendment request to revise the dose consequence analysis for a loss-of-coolant accident for Susquehanna Steam Electric Station (Susquehanna), Units 1 and 2. The meeting notice and agenda, dated October 7, 2019, are available in the Agencywide Documents Access and Management System (ADAMS) at Accession No. ML19282A701. A list of attendees is provided as an enclosure.
The licensee’s presentation materials can be found at ADAMS Accession No. ML19298A955.

The licensee’s staff expects to submit the proposed license amendment request (LAR) for Susquehanna, Units 1 and 2 by the end of December 2019.

The licensee discussed a number of planned changes to the dose consequence analysis for a loss-of-coolant accident (LOCA). The changes being considered include:

- Upgrade to ORIGEN-ARP for source term.
- Reduce assumed engineered safety feature leakage to align with Regulatory
- Guide 1.183.
- Increase assumed Secondary Containment inleakage.
- Increase assumed Control Room Habitability Envelope inleakage.
- Eliminate Control Room Habitability Envelope continuous occupancy areas to align with Regulatory Guide 1.183.
- Upgrade to RADTRAD Version 3.10.

The NRC staff's discussion included:

- The future LAR will not be related to the Atrium 11 LAR that the licensee has already submitted to the NRC staff. The licensee clarified that the planned LAR and Atrium 11 LAR are two separate licensing actions and each is independent of the other.
- The licensee plans to submit the LAR by the end of December 2019 and would need this amendment prior to the next outage in 2021.
- The LAR will not include modification to primary system piping.
- The LAR does not involve an update to the neutron fluence calculation method or
- the neutron fluence estimates.
- The licensee was advised by the NRC staff that it should address how Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36 is still being met for the proposed changes to its technical specifications (TSs). If the licensee is proposing a change to the administrative controls section of its TSs, then the licensee should explain how the proposed change continues to assure operation of the facility in a safe manner in accordance with 10 CFR 50.36(c)(5).
- The licensee is updating its current licensing basis accident source term to reflect the conversion from ATRIUM 10 to ATRIUM 11 fuel. The current licensing basis accident source term utilized the computer code SAS2H/ORIGEN-S to compute the reactor core isotopic inventory. With the conversion to ATRIUM 11, the licensee has chosen to compute the reactor core isotopic inventory utilizing the OrigenArp computer code and considers this a method change.
- With respect to analyses which assume continuous occupancy in rooms not required by Regulatory Guide 1.183, the NRC staff indicated a need for the licensee to describe why continuous occupancy has been assumed for these rooms and to provide a justification as to why continuous occupancy no longer applies. The NRC staff indicated that a justification for the change simply based on Regulatory Guide 1.183 not requiring continuous occupancy assumption is not acceptable. The staff also requested that a clear and detailed basis be included for the proposed increase in secondary containment and control room envelope inleakage values.
- In order to improve the efficiency of the NRC staff's review, it is important for licensees to explicitly identify the areas that are - and are not – affected by the license amendment request with respect to the design-basis accident loss-of-coolant accident dose analysis. The licensee should consider providing a matrix that includes information for each input parameter; the current licensing basis vale, the proposed value, justification for the proposed value, and reference to the applicable updated final safety analysis report section and/or TS.
- In order to accomodate the NRC staff's review, the various RadTrad 3.10 input files would be preferred to improve the efficiency to confirm the licensees analysis. Otherwise, the RadTrad output file would be sufficient for confirmatory analyses.

Members of the public were not in attendance. Public meeting feedback forms were not received.

Please direct any inquiries to me at 301-415-8004 or Sujata.Goetz@nrc.gov.

January 7, 2020 – Letter dated January 7, 2020 from Daniel S. Collins, Director Division of Reactor Projects to Brad Berryman, President and Chief Nuclear Officer Susquehanna Nuclear with a subject of Susquehanna Steam Electric Station – NRC Investigation Report Number 1-2018-011 and Notice of Violation.

This letter refers to an investigation conducted by the U.S. Nuclear Regulatory Commission (NRC) Office of Investigations (01) at the Susquehanna Nuclear, LLC (Susquehanna Nuclear) Susquehanna Steam Electric Station (SSES). The investigation, which was completed on April 19, 2019, was conducted to evaluate potential violations of NRC fitness for duty (FFD) requirements by contract workers. Based on the evidence gathered during the 01 investigation, the NRC determined that one Severity Level IV (SL IV) violation of NRC requirements occurred. Specifically, the NRC identified that a contract ironworker who worked at SSES through BHI Energy provided inaccurate information that was material to the NRC about past and current substance abuse on applications for unescorted access authorization, thereby creating a false record. As a result, Susquehanna Nuclear granted unescorted access to the ironworker based, in part, on the inaccurate information and without having the opportunity to review and resolve the potentially disqualifying FFD information. The NRC notes that Susquehanna Nuclear terminated the contract ironworker's employment due to separate FFD concerns and that this issue did not result in any actual safety or security impacts at the site.

After considering the factors set forth in Section 2.3.2.a of the Enforcement Policy, this violation is being cited in the enclosed Notice of Violation (Notice). Namely, the violation was identified by the NRC. You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. If you have additional information that you believe the NRC should consider, you may provide it in your response to the Notice. The NRC review of your response to the Notice will also determine whether further enforcement action is necessary to ensure compliance with regulatory requirements. In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be made available electronically for public inspection in the NRC Public Document Room and from the NRC's Agency-wide Documents Access and Management 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 or proprietary information so that it can be made available to the Public without redaction.

Please note that final NRC investigation documents, such as the 01 report described above, may be made available to the public under the Freedom of Information Act (FOIA}, subject to redaction of information appropriate under the FOIA. Requests under the FOIA should be made in accordance with 10 CFR 9.23, "Requests for Records." Additional information is available on the NRC website at http://www.nrc.gov/readinq-rm/foia/foia-privacy.html.

This enforcement action will be administratively tracked under NRC Inspection Report No. 05000387; 05000388/2019090. Should you have any questions regarding this letter, please contact Mr. Jon Greives at 610-337-5337.

Notice of Violation:

During an NRC investigation conducted between April 20, 2018 and April 19, 2019, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below.

10 CFR 50.9(a) requires that information required by the Commission's regulations to be maintained by the licensee shall be complete and accurate in all material respects.

10 CFR 26.713(a)(1) requires, in part, that licensees retain records of self-disclosures that are required under 10 CFR 26.59, that result in the granting of authorization, for at least 5 years after the licensee terminates or denies an individual's authorization.

10 CFR 26.59(a)(1) requires that, in order to grant authorization to an individual whose authorization has been interrupted for a period of more than 30 days but no more than 365 days and whose last period of authorization was terminated favorably, the licensee shall ensure that a self-disclosure has been obtained and reviewed under the applicable requirements of 10 CFR 26.61.

10 CFR 26.61 requires, in part, that before granting authorization, the licensee shall ensure that a written self-disclosure has been obtained from the individual who is applying for authorization. The written self-disclosure must state whether the individual has used, sold, or possessed illegal drugs; and whether the individual has abused legal drugs or alcohol. 10 CFR 26.5, in part, defines potentially disqualifying FFD information as information demonstrating that an individual has used, sold, or possessed illegal drugs or has abused legal drugs.·

Contrary to the above, from January 14, 2017, through April 19, 2018, Susquehanna Nuclear, LLC maintained information that was required by the Commission's regulations to be maintained that was not complete and accurate in all material respects. Specifically, on January 14, 2017, and January 31, 2018, an individual applying for access authorization submitted written self-disclosures on which the individual stated that he had not used, sold, or possessed illegal drugs and had not abused legal drugs or alcohol. However, the individual had used and possessed illegal drugs and had abused legal drugs. The information was material to the NRC because the inaccuracies involved potentially disqualifying fitness for duty information and based in part on this inaccurate information, the licensee granted unescorted access authorization to the individual until April 19, 2018.

This is a Severity Level IV violation (Enforcement Policy Sections 2.2.4 and 6.9).

Pursuant to the provisions of 10 CFR 2.201, Susquehanna Nuclear, LLC is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001 with a copy to the Regional Administrator, Region I, 2100 Renaissance Blvd., Suite 100, King of Prussia, PA 19406, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as "Reply to a Notice of Violation; EA-19-050," and should include: (1) the eason for the violation, or, if contested, the basis the basis for disputing the violation or severity level, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken, and (4) the date when full compliance will be achieved. In particular, the NRC requests that your response include discussion of the actions being taken to evaluate employee standards related to procedural adherence.

Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C., 20555-0001.

January 9, 2020 – Letter dated January 9, 2020 from Sujata Goetz, Project Manager, Susquehanna Steam Electric Station to Shane Jurek of Talen Energy

By letter dated July 15, 2019, Talen Energy submitted a license amendment request (LAR} for Susquehanna Steam Electric Station, Units 1 and 2 (Susquehanna) to allow application of the Framatome analysis methodologies necessary to support a planned transition to ATRIUM 11 fuel under the currently licensed Maximum Extended Load Line Limit Analysis (MELLLA) operating domain (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19196A270).

The Nuclear Regulatory Commission's (NRC) staff has reviewed your LAR and has determined that additional information is needed to complete its review. The specific questions are in the attachment to this letter.

Your response to these questions is due by February 6, except for question 2, which is due to the NRC by March 6, 2020.

Request for additional information for Susquehanna Steam Electric Station, Units 1 and 2, to support review of the license amendment request regarding application of framatome methodologies to support transition to atrium 11 fuel.

1. Containment
    a. Describe the analysis done to justify (redacted)
    b. Provide quantitative results for the containment pressure and suppression pool temperature response changes due to the change in fuel type. Describe the analysis performed to confirm the ATRIUM-10 analysis bounds the ATRIUM 11 fuel transition.
2. Anticipated operational occurences (AOOS) and ATWS
    a. Regulatory Basis -10 CFR 50, GDCs 10, 13, 15, 20, 25, 26, and ATWS acceptance criteria

2.1 ANP-3753P and ANP-3783P provide a subset of the events analyzed in the Susquehanna Chapter 15 Updated Final Safety Analysis Report (UFSAR) and covered by the AURORA-B AOO/ATWS methodology. To ensure the methodology is implemented appropriately for the events not covered in ANP-3753P and ANP-3783P, provide the following:

a. Describe how each Chapter 15 UFSAR event (that is covered by the AURORA-B AOO/ATWS methodology) will be analyzed in the AURORA-B AOO methodology framework (e.g., a table identifying UFSAR Section/Event Name/Disposition)
b. Describe how the methodology is implemented (including steps prior to the execution of the uncertainty analysis) to ensure nuclear power plant - specific options are covered in the analyses.
c. Void quality correlation uncertainties are discussed in Section 6.1 of ANP-3753P. Provide information about which parameters are sampled and which parameters are biased. How is a conservative approach ensured regarding the sampled and biased parameters?

2.2 To ensure there is appropriate coverage of the parameters used in the uncertainty analysis and to ensure there is no significant trends with respect to the uncertainty parameters in the results such that the Susquehanna implementation of the AURORA-B methodology is sufficient, provide the following for the load rejection no bypass/turbine trip without bypass event at 100% power/ 108% flow, main steam isolation valve closure ATWS event at 100% power and 99% flow, and high pressure coolant injection event at 100% power / 108% flow:

1. The sampled values of the uncertainty parameters for all cases executed in the set
2. The figure of merit results for all cases executed in the set

2.3 Please provide the schedule for Reload Safety Analysis Report (RSAR) submittal. Discuss how the information in the RSAR is used to confirm the AURORA-B limitations and conditions in ANP-2637P, "Boiling Water Reactor Licensing Methodology Compendium, Rev. 8", are appropriately applied.

2.4 Section 5.4 of ANP-3753P describes the safety limit minimum critical power ratio methodology at SUSQUEHANN. This methodology is used to determine that 99.9% of the fuel rods are expected to avoid boiling transition during normal reactor operation and anticipated operation occurrences. The analysis provided by the licensee shows that (redacted) Please provide the approach used to confirm the bounds will be checked in the appropriate assemblies of the core for future reloads. What process is applied if (redacted)

2.5 In the AOO event analysis in ANP-3753P, the load rejection no bypass event is combined with the turbine trip without bypass event even though plant systems may respond differently for each event. Justify that one event bounds the other without doing explicit analysis for both events. Confirm that the bounding analysis can be determined by combining these two events.

3. Fuel: Introduction of atrium 11 fuel to Susqeuhanna

REGULATORY BASIS-10 CFR 50, GDCS 10, 13, 15, 20, 25, 26, AND ATWS ACCEPTANCE CRITERIA

GDC 10 requires that specified acceptable fuel design limits are not exceeded during normal operation including the effects of AOOs. Oxidation and hydriding are two specified acceptable fuel design limits that ensure components maintain strength and ductility. Section 3.5.1 of ANP-3762P mentions that water chemistry is controlled to reduce oxidation in the fuel channel. Please describe what process is used to control the water chemistry and what are the key figures-of-merit monitored to ensure satisfactory performance of ATRIUM 11 fuel and the 248 water channel.

4. Loss of Coolant Accident (LOCA)

REGULATORY BASIS-10 CFR 50, GDCS 10, 13, 15, 20, 25, 26, AND ATWS ACCEPTANCE CRITERIA

The regulatory bases for the following LOCA related requests for additional information are the requirements contained in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," insofar as they establish the requirements and acceptance criteria for emergency core cooling system (ECCS) design, and for the evaluation models used to evaluate ECCS performance during a hypothetical LOCA. Specific considerations include:

- 10 CFR 50.46(a)(1)(i) requires the use of an acceptable evaluation models to evaluate ECCS performance under the conditions of a hypothetical LOCA, and 10 CFR 50.46(a)(1)(ii) allows for the development of an evaluation models that conforms to the required and acceptable features specified in Appendix K to 10 CFR 50.
- 10 CFR 50.46(a}(1)(i) also requires ECCS cooling performance to be calculated for a number of postulated LOCAs of different sizes, locations, and other properties sufficient to provide assurance that the most severe hypothetical LOCAs are calculated.
- Acceptance criteria set forth in paragraph (b) of 10 CFR 50.46, and the results of the ECCS evaluation must show that the acceptance criteria are met. Among others, these include requirements related to peak cladding temperature (PCT), maximum cladding oxidation, and maximum hydrogen generation.

For licensed operating domain and equipment-out-of-service, please provide justification to assure that the LOCA analysis has been performed conservatively to cover Susquehanna licensed operating domain and equipment out-of-service conditions.:.

For, limiting PCT: Explain why the limiting PCT of (redacted) of exposure-dependent LOCA analysis.

For local Cladding Oxidation (Table 9.1 of ANP-3784P): Explain why the change of local cladding oxidation from the assembly average planar exposure of (redacted)

Linear heat generation rate (LHGR) and maximum average planar LGHR (MAPLHGR) Data Used in Exposure-Dependent Analysis

- What is the process for determining the LHGR used, for both U02 and Gd203-U02 pellets during exposure-dependent analysis, in the AURORA-B LOCA analysis? Specifically, are the LHGR limit curves presented in Figures 2.2 and 2.3 shown in ANP-3784P, "Susquehanna ATRIUM 11 Introduction - Exposure-Dependent LOCA Analysis," (redacted)
- Please demonstrate the analysis margin for the MAPLHGR limit in Figure 2.1 of ANP-3784P, (redacted)

Please address how the implementation of Atrium 11 fuel affects the aging degradation on the reactor vessel pressure and reactor pressure internal components.

REGULATORY BASIS-10 CFR 50, GDCS 10, 13, 15, 20, 25, 26, AND ATWS ACCEPTANCE CRITERIA

If the neutron fluence values associated with Atrium 11 are higher than the Atrium 10 fuel, the licensee should provide a technical explanation how it intends to manage the aging degradation related to irradiation embrittlement, irradiation-assisted stress corrosion cracking, and, irradiation stress relaxation at Susquehanna units in the current licensing period.

January 13, 2020 – Letter dated January 13, 2020 from Sujata Goetz, Project Manager Plant Licensing Branch 1, Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to Kevin Cimorelli, Site Vice President Susquehanna Nuclear, LLC with subject line of: Susquehanna steam electric station,, units 1 and 2 – issuance of amendment nos 274 and 256 to adopt technical specifications task force traveler, TSTF-535, revision 0, revise shutdown margin definition to address advanced fuel designs)EPID L-2019-LLA-0154).

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 274 to Renewed Facility Operating License No. NPF-14 and Amendment No. 256 to Renewed Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Units 1 and 2, respectively. These amendments consist of changes to the technical specifications (TSs) in response to your application dated July 15, 2019.

The license amendment requests proposed changes to adopt Technical Specifications Task Force (TSTF) Traveler TSTF-535, Revision 0, "Revise Shutdown Margin Definition to Address Advanced Fuel Designs," and revise TS 5.6.5b to allow application of Advanced Framatome methodologies for loading Framatome fuel type ATRIUM 11 The enclosed amendments are for the TSTF-535 portion of the application. If approved, the adoption of Framatome fuel type ATRIUM 11 will be addressed in separate amendments and issued later.

The enclosed amendments revise the TS definition of "shutdown margin" to require its calculation at a reactor moderator temperature of 68 degrees Fahrenheit or a higher temperature that represents the most reactive state throughout the operating cycle.

A copy of the related safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's Biweekly *Federal Register* Notice.

Findings:

1. The U.S. Nuclear Regulatory Commission (NRC or the Commission) has found that:

    1. The application for the amendment filed by Susquehanna Nuclear, LLC, dated July 15, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 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, 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. NPF-14 is hereby amended to read as follows:

    (2) Technical Specifications and Environmental Protection Plan

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 274, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. Susquehanna Nuclear, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance. Once approved, Amendment No. 274 for Unit 1 will be implemented prior to loading ATRIUM 11 fuel into the core during the spring 2022 refueling outage.

February 13, 2020 – Letter dated February 13, 2020 from Jonathan E. Greives, Chief Reactor Projects Branch 4 Division of Reactor Projects with the subject of: Susquehanna Steam Electric Station, Units 1 and 2 – Integrated Inspection Report 05000387/2019004 and 05000388/2019004

On December 31, 2019, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Susquehanna Steam Electric Station, Units 1 and 2. Two findings of very low safety significance (Green) are documented in this report. Both of these findings involved violations of NRC requirements.

List of Findings and Violations

1. Untimely Identification and Correction of Breaker Stab Misalignment Results In Subsequent Safety Bus Fault
    a. The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action." Specifically, Susquehanna failed to identify and correct a condition adverse to quality, associated with misalignment of Class 1E breaker 0B136-044 to safety- related electrical bus 0B136, that resulted in a repeat electrical bus (0B136) fault.
2. Inadequate procedural adherence for 'C' Emergency Service Water (ESW) pump flow surveillance
    a. The inspectors documented a self-revealing Green finding and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," when the licensee failed to accomplish an activity affecting quality in accordance with procedures and a violation of Technical Specification 3.7.2, Condition C, for exceeding the allowed outage time.

March 3, 2020 – Letter dated March 3, 2020 from Jonathan E. Greives, Chief Reactor Projects Branch 4 Division of Reactor Projects to Brad Berryman, President and Chief Nuclear Officer, Susquehanna Nuclear, LLC with a subject of ANNUAL ASSESSMENT LETTER FOR SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 (REPORT 05000387/2019006 AND 05000388/2019006).

Dear Mr. Berryman:

The NRC has completed its end-of-cycle performance assessment of Susquehanna Steam Electric Station, Units 1 and 2, 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 Susquehanna Steam Electric Station, Units 1 and 2 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 your facility.

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 conduct Inspection Procedure (IP) 71003, "Post Approval Site Inspection for License Renewal," for Unit 1 in April 2020, and the NRC will schedule an additional inspection per a revised version of 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). Licensees will be individually notified when the NRC schedules these inspections.

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 me at (610) 337-5337 with any questions you have regarding this letter.

Susquehanna

01/01/2020 - 12/31/2021

IP 22 Inspection Activity Plan Report

Unit	Start	End	Activity	CAC	Title	Staff Count
HP 1						
1, 2	01/06/2020	01/09/2020	IP 71124.01	000725	Radiological Hazard Assessment and Exposure Controls	
1, 2	01/06/2020	01/10/2020	IP 71124.02	000726	Occupational ALARA Planning and Controls	
1, 2	01/06/2020	01/10/2020	IP 71124.03	000727	In-Plant Airborne Radioactivity Control and Mitigation	
1, 2	01/06/2020	01/10/2020	IP 71124.04	000728	Occupational Dose Assessment	
1, 2	01/06/2020	01/10/2020	IP 71124.05	000729	Radiation Monitoring Instrumentation	
TI-194 Open Phase Condition Inspection-SQ 2						
1, 2	02/03/2020	02/07/2020	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)	
License Renewal Phase 1 - Unit 1 1						
1	04/05/2020	04/11/2020	IP 71003	000687	Post-Approval Site Inspection for License Renewal	
ISI - UNIT 1 1						
1	04/05/2020	04/11/2020	IP 71111.08G	000701	Inservice Inspection Activities (BWR)	
HP 3						
1, 2	04/13/2020	04/17/2020	IP 71124.01	000725	Radiological Hazard Assessment and Exposure Controls	

1, 2 04/13/2020 04/17/2020 IP 71124.03 000727 In-Plant Airborne Radioactivity Control and Mitigation
1, 2 04/13/2020 04/17/2020 IP 71124.08 000732 Radioactive Solid Waste Processing and Radioactive Material Handling, Storage, and Transportation
Access Control, Protective Strategy, TSR-SUS 4
1, 2 06/15/2020 06/19/2020 IP 71130.02 000734 Access Control
1, 2 06/15/2020 06/19/2020 IP 71130.05 000737 Protective Strategy Evaluation
1, 2 06/15/2020 06/19/2020 IP 71130.09 001656 Security Plan Changes
1, 2 06/15/2020 06/19/2020 IP 71130.14 000743 Review of Power Reactor Target Sets
1, 2 06/15/2020 06/19/2020 IP 71151 001338 Performance Indicator Verification
HP 1
1, 2 07/06/2020 07/10/2020 IP 71124.05 000729 Radiation Monitoring Instrumentation
1, 2 07/06/2020 07/10/2020 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.

3/2/2020 10:25:12 AM

Enclosure

Susquehanna

01/01/2020 - 12/31/2021

IP 22 Inspection Activity Plan Report

Unit	Start	End	Activity	CAC	Title	Staff Count
PI&R Biennial Team Inspection - SQ 4						
1, 2 07/13/2020 07/17/2020 IP 71152B 000747 Problem Identification and Resolution						
1, 2 07/27/2020 07/31/2020 IP 71152B 000747 Problem Identification and Resolution						
FIRE PROTECTION-SUSQUEHANNA 3						
1, 2 09/21/2020 09/25/2020 IP 71111.21N.05 001646 Fire Protection Team Inspection (FPTI)						
1, 2 10/05/2020 10/09/2020 IP 71111.21N.05 001646 Fire Protection Team Inspection (FPTI)						
EP EXERCISE INSPECTION - SUSQUEHANNA 5						
1, 2 10/19/2020 10/23/2020 IP 71114.01 000716 Exercise Evaluation						
1, 2 10/19/2020 10/23/2020 IP 71114.04 000719 Emergency Action Level and Emergency Plan Changes						
1, 2 10/19/2020 10/23/2020 IP 71151 001397 Performance Indicator Verification						
INSERVICE INSPECTION 1						
2 03/21/2021 03/27/2021 IP 71111.08G 000701 Inservice Inspection Activities (BWR)						
HP 1						
1, 2 04/05/2021 04/09/2021 IP 71124.01 000725 Radiological Hazard Assessment and Exposure Controls						
1, 2 04/05/2021 04/09/2021 IP 71124.02 000726 Occupational ALARA Planning and Controls						
HEAT SINK INSPECTION - BIENNIAL 1						

1, 2 05/16/2021 05/22/2021 IP 71111.21N 000714 Design Bases Assurance Inspection (Programs)						
FY21 Susquehanna Initial Examination 4						
1, 2 05/23/2021 05/28/2021 OV 000956 VALIDATION OF INITIAL LICENSE EXAMINATION (OV)						
1, 2 06/20/2021 07/02/2021 EXAD 000500 LICENSE EXAM ADMINISTRATION (EXAD)						
REMP 1						
1, 2 05/24/2021 05/28/2021 IP 71124.07 000731 Radiological Environmental Monitoring Program						
RETS 1						
1, 2 06/21/2021 06/25/2021 IP 71124.06 000730 Radioactive Gaseous and Liquid Effluent Treatment						
FORCE-ON-FORCE PLANNING AND EXERCISE WEEKS - SQ 6						
1, 2 06/28/2021 07/02/2021 IP 71130.03 000735 Contingency Response - Force-On-Force Testing						
1, 2 07/19/2021 07/23/2021 IP 71130.03 000735 Contingency Response - Force-On-Force Testing						

This report does not include INPO and OUTAGE activities.
This report shows only on-site and announced inspection procedures.

Susquehanna

01/01/2020 - 12/31/2021

IP 22 Inspection Activity Plan Report

Unit	Start	End	Activity	CAC	Title	Staff Count
EP Program Inspection - Susuquehanna 1						
1, 2 07/19/2021 07/23/2021 IP 71114.02 000717 Alert and Notification System Testing						
1, 2 07/19/2021 07/23/2021 IP 71114.03 000718 Emergency Response Organization Staffing and Augmentation System						
1, 2 07/19/2021 07/23/2021 IP 71114.04 000719 Emergency Action Level and Emergency Plan Changes						
1, 2 07/19/2021 07/23/2021 IP 71114.05 000720 Maintenance of Emergency Preparedness						
1, 2 07/19/2021 07/23/2021 IP 71151 001397 Performance Indicator Verification						
HP 1						
1, 2 08/23/2021 08/27/2021 IP 71124.02 000726 Occupational ALARA Planning and Controls						
1, 2 08/23/2021 08/27/2021 IP 71124.04 000728 Occupational Dose Assessment						
1, 2 08/23/2021 08/27/2021 IP 71151 000746 Performance Indicator Verification						
Evaluations of Changes, Tests and Experiments 3						
1, 2 09/12/2021 09/18/2021 IP 71111.17T 000709 Evaluations of Changes, Tests, and Experiments						
Access Control, Equipment Testing and Maintenance, Training, SPR 3						
1, 2 10/18/2021 10/22/2021 IP 71130.02 000734 Access Control						
1, 2 10/18/2021 10/22/2021 IP 71130.04 000736 Equipment Performance, Testing, and Maintenance						
1, 2 10/18/2021 10/22/2021 IP 71130.07 000739 Security Training						
1, 2 10/18/2021 10/22/2021 IP 71130.09 001656 Security Plan Changes						
1, 2 10/18/2021 10/22/2021 IP 71151 001338 Performance Indicator Verification						
Design Basis Assurance Inspection - Programs - Power Operated Valves - Susquehanna Units 1 and 2 3						

1, 2 10/25/2021 10/29/2021 IP 71111.21N.02 001645 Design-Basis Capability of Power-Operated Valves Under 10 CFR 50.55a Requirements	
1, 2 11/08/2021 11/12/2021 IP 71111.21N.02 001645 Design-Basis Capability of Power-Operated Valves Under 10 CFR 50.55a Requirements	
SQ Requal Inspection with P/F Results 2	
1, 2 11/15/2021 11/19/2021 IP 71111.11A 000703 Licensed Operator Requalification Program and Licensed Operator Performance	
1, 2 11/15/2021 11/19/2021 IP 71111.11B 000704 Licensed Operator Requalification Program and Licensed Operator Performance	

This report does not include INPO and OUTAGE activities.
This report shows only on-site and announced inspection procedures.

March 19, 2020 – Letter dated March 19, 2020 from Sujata Goetz, Project Manager Plant Licensing Branch I Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to Talen Energy with a subject line of SUMMARY OF FEBRUARY 27, 2020, MEETING WITH TALEN ENERGY REGARDING FUTURE LICENSE AMENDMENT REQUEST RELATED TO CREATING NEW CONDITION FOR INOPERABLE MANUAL SYNCHRONIZATION CIRCUIT (EPID L-2019-LRM-0069)

On February 27, 2020, a Category 1 public teleconference was held between the U.S. Nuclear Regulatory Commission (NRC) staff and representatives of Talen Energy (the licensee) at NRC Headquarters, One White Flint North, 11555 Rockville Pike, Rockville, Maryland. The purpose of the meeting was to discuss a future license amendment request to create a new technical specification (TS) condition for an inoperable manual synchronization circuit and institute a completion time (CT) of 14 days to restore the circuit to an operable status for Susquehanna Steam Electric Station (Susquehanna), Units 1 and 2. The meeting notice dated February 10, 2020, is available in the Agencywide Documents Access and Management System (ADAMS) at Accession No. ML20042E722. A list of attendees is enclosed.

Talen Energy provided presentation slides (ADAMS Accession No. ML20058B167). The licensee expects to submit the license amendment request by the end of March 2020.

The proposed amendments would modify existing Limiting Condition for Operation (LCO) 3.8.1, “AC [Alternating Current] Sources – Operating,” to create a new condition for an inoperable manual synchronization circuit and institute a CT of 14 days to restore the circuit to an operable status. The premise for this request is based on failure of a synchronization selector switch. The licensee stated that the synchronization switch circuit impacts the bus transfer scheme for preferred power sources and onsite power sources for both Susquehanna units and results in both units entering TS-related LCO 3.0.3, which requires unit shutdown immediately. However, the licensee noted that the safety functions of the offsite and onsite power systems, as assumed in Susquehanna’s safety analyses, are not adversely impacted.

Susquehanna is proposing that the required action of a dual unit shutdown is not commensurate with the overall risk of the configuration. The licensee explained that the circuits associated with the synchronization hand switch are shared by all diesel generators and the 4.16 kilovolt (kV) and 13.8 kV safe shutdown buses on both units. However, the automatic transfers associated with a unit trip or diesel generator powering the safety buses following loss-of-offsite-power (LOOP) event are not impacted by an inoperable manual synchronization circuit. Therefore, all assumptions of the accident analyses are met for this condition.

During the meeting with Talen Energy, the NRC staff discussed the loss of synchronization capability for diesel generator testing and restoration of plant buses to the preferred power source after recovering from a LOOP event. The staff noted that safety bus voltage and

frequency monitoring circuits also provided input to the synchronization circuits, and therefore, the staff needed to understand the consequences of switch failures on the associated circuits. The NRC staff requested a logic diagram, as well as simplified drawings, to help understand the circuit and the impact on the onsite and offsite power systems.

The licensee discussed the potential delay in TSs currently applicable for monthly surveillance of the diesel generators and the safety significance of the ability to reconnect the safety buses to the preferred power source (after recovery from a LOOP event) if an entry into the proposed LCO is implemented during plant operation. The NRC staff also stated that a 14-day CT appears long for switch replacement, considering a 6-day CT associated with a diesel generator outage. The proposed CT should be commensurate with the time required to restore the synchronization circuit to an operable status. A justification based on past operating experience should be provided to support the proposed CT.

No regulatory decisions were made during the meeting. Members of the public were not in attendance. Public meeting feedback forms were not received.

Please direct any inquiries to me at 301-415-8004 or Sujata.Goetz@nrc.gov.

LIST OF ATTENDEES
FEBRUARY 27, 2020, MEETING WITH TALEN ENERGY REGARDING FUTURE LICENSE AMENDMENT REQUEST
RELATED TO CREATING NEW CONDITION FOR INOPERABLE MANUAL SYNCHRONIZATION CIRCUIT
SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2

Name

Sujata Goetz - U.S. Nuclear Regulatory Commission (NRC) NRC

Gurcharan Matharu* - NRC
Roy Mathew - NRC
Tarico Sweat* - NRC

Shane Jurek* - Talen Energy
Katie Brown*- Talen Energy
Melisa Krick*- Talen Energy
Jason Lada*- Talen Energy
Jeff Oswald*- Talen Energy
Rob Peterson* - Talen Energy

*participated by teleconference

SUBJECT:

SUMMARY OF FEBRUARY 27, 2020, MEETING WITH TALEN ENERGY REGARDING FUTURE LICENSE AMENDMENT REQUEST RELATED TO CREATING NEW CONDITION FOR INOPERABLE MANUAL SYNCHRONIZATION CIRCUIT (EPID L-2019-LRM-0069)

DATED MARCH 19, 2020

May 28, 2020 – Letter from Glenn T. Dentel, Chief Engineering Branch 2 Division of Reactor Safety to Brad Berryman President and Chief Nuclear Officer Susquehanna Nuclear, LLC with subject of SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 – 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 Susquehanna Steam Electric Station, Units 1 and 2, starting in September 2020. 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: September 8 – 10, 2020
- • Onsite Inspection: Weeks of September 21, 2020 and October 5, 2020

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 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 August 14, 2020. 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. 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 August 14, 2020:

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, etc.
7. Fire Protection Design Basis Document(s), if available
8. List of all safety evaluations performed to support any plant modifications since September 2017. Include a short description and/or title of each evaluation.
9. List of all 10 CFR 50.59 applicability determinations and screens (date, number, title) performed since September 2017. Include a short description and/or title of each document.

10. List of all fire protection system impact screening reviews for any design changes, modifications, or temporary modifications completed since September 2017 (e.g., a Generic Letter 86-10 review, LS-AA-128 review, etc.). Include a short description and/or title of each review.
11. List of fire protection system, post-fire safe shutdown, or alternative shutdown design changes completed since September 2017. Include a short description and/or title of each change.
12. List of the top 25 highest fire CDF scenarios, if available
13. List of the top 25 highest fire LERF scenarios, if available
14. From your most recent site-specific PRA, including external events and fires (if available):
    - Two risk rankings of components: one sorted by Risk Achievement Worth (RAW) and the other sorted by Birnbaum Importance
    - A list of the top 100 cut sets
15. Risk ranking of operator actions and/or recovery actions from your site-specific PRA sorted

    by Risk Achievement Worth

16. List of current fire protection system impairments, including description
17. List of time critical operator actions and associated program procedure
18. One-line diagram of the electrical distribution system
19. Copy of the Updated Final Safety Analysis Report
20. Copy of the Technical Requirements Manual
21. Copy of the Quality Assurance Program Manual (including specific fire protection Quality Assurance Manual, if applicable)
22. Copy of the Corrective Action Program Procedure(s)
23. List of station procedures used to respond to fire (i.e., Emergency Operating Procedures, Abnormal Operating Procedures, and Annunciator Response Procedures). Include the procedure number, title, and current revision
24. List of open and closed condition reports for post-fire safe shutdown or alternative shutdown issues since September 2017 (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.

25. List of open and closed condition reports for fire protection system issues (e.g., fire pumps, detection, suppression, etc.) since September 2017. Include the issue report number and a brief description.

26. List of open and closed condition reports related to the fire brigade or fire drills since July 2017. Include the issue report number and a brief description.
27. Copies of the following condition reports:

CR-2017-13710 CR-2017-14574 CR-2017-14743 CR-2017-14748 CR-2018-12310

CR-2017-14802 CR-2017-15307 CR-2018-12310 CR-2017-14813 CR-2017-15491 CR-2017-15590 CR-2017-14957 CR-2017-15529 CR-2017-15656 CR-2017-15036 CR-2017-15531 CR-2018-12308

28. Copies of any self-assessments performed, and corrective action documents generated, in preparation for this fire protection team inspection.

June 26, 2020 – email from Sujata Goetz to Shane Jurek with subject of Acceptance Review For Susquehanna - Revise Technical Specification 3.8.1 To Create A New Condition For An Inoperable Manual Synchronization Circuit (EPID L-2020-LIa-0118)

By letter dated May 26, 2020 (Agencywide Document and Access Management System (ADAMS) Accession No. ML20148L497, Susquehanna Nuclear, LLC submitted a license amendment request (LAR) for Susquehanna Unit 1 and Unit 2. The LAR would create a new technical specification Action for an inoperable manual synchronization circuit requiring restoration within 14 days. The proposed amendment is necessary to reduce the potential for an unnecessary dual unit shutdown. Based on the configuration of the AC power sources at Susquehanna, an inoperable manual synchronization circuit currently results in entry into Limiting Condition for Operation 3.0.3 for both units, which is not commensurate with the risk associated with having an inoperable manual synchronization circuit.

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 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.

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. 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 in approximately 12 months which is May 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, 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.

If you have any questions, please contact me.

August 11, 2020 – Letter from Jonathan E. Greives, Chief Reactor Projects Branch 4 Division of Reactor Projects to Mr. Brad Berryman Senior Vice President and Chief Nuclear Officer Susquehanna Nuclear, LLC with a subject of SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 – INTEGRATED INSPECTION REPORT 05000387/2020002 AND 05000388/2020002

On June 30, 2020, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Susquehanna Steam Electric Station, Units 1 and 2. On July 30, 2020, the NRC inspectors discussed the results of this inspection with Mr. Kevin Cimorelli 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 Susquehanna Steam Electric Station, Units 1 and 2.

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 Susquehanna Steam Electric Station, Units 1 and 2.

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: 05000387 and 05000388
License Numbers: NPF-14 and NPF-22
Report Numbers: 05000387/2020002 and 05000388/2020002 I-2020-002-0040
Licensee: Susquehanna Nuclear, LLC
Facility: Susquehanna Steam Electric Station, Units 1 and 2
Location: Berwick, PA
Inspection Dates: April 1, 2020 to June 30, 2020

Inspectors:	E. Dipaolo, Senior Reactor Inspector D. Kern, Senior Reactor Inspector J. Kulp, Senior Reactor Inspector L. Micewski, Senior Resident Inspector R. Rolph, Resident Inspector M. Rossi, Senior Resident Inspector, Acting
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 Susquehanna Steam Electric Station, Units 1 and 2, 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

Turbine trip stop and control valve closure scram inoperable during plant start up

Cornerstone

Mitigating Systems

Significance

Green
NCV 05000387/2020002-01 Open/Closed

Additional Tracking Items

Issue Number Title

Cross-Cutting Aspect
[H.11] - Challenge the Unknown

Report Section 71111.22

NRC inspectors identified a Green finding and associated non-cited violation (NCV) of Technical Specification (TS) 5.4.1, "Procedures," when control room operators raised reactor power above 25 percent while the reactor protection system (RPS) scram bypass function was inoperable, as indicated by the turbine control fast closure and stop valve trip bypass alarm being annunciated contrary to procedure GO-100-002, "Plant Startup, Heatup, and Power Operations."

Type

Report Section Status

LER	05000388/2020-001-00	LER 2020-001-00 for Susquehanna Steam Electric Station, Unit 2, Manual Reactor Scram Due to Rising Main Condenser Backpressure	71153	Closed

PLANT STATUS

Unit 1 began the inspection period shutdown for a planned refueling outage. Following the completion of refueling and maintenance activities, operators commenced a reactor startup on April 23, 2020. On April 25, 2020, while at 16 percent power, operators returned the unit to Mode 5 following a malfunction of the electrohydraulic control unit system. Operators commenced plant startup operations on April 30, 2020. On May 3, 2020, the unit experienced an automatic scram from 76 percent power due to a failure of a current transformer, which resulted in a main turbine trip. Operators commenced startup operations on May 9, 2020, and reached approximately 77 percent on May 14, 2020, when they reduced power to 57 percent for a rod pattern adjustment. The station achieved 81 percent power on May 16, 2020, when operators lowered power to 57 percent for a rod pattern adjustment, and commenced power ascension on the same day. The unit was at 97 percent on May 20, 2020, when operators lowered power to 58 percent for a rod pattern adjustment, returning to 94 percent power the following day. On June 17, 2020, power was reduced from 100 percent to 57 percent power due to a feedwater heater extraction steam isolation, and the unit was returned to 100 percent power the following day. The station remained at or near 100 percent power for the remainder of the inspection period.

Unit 2 began the inspection period at 99 percent power. The station had been requested by grid operator on November 21, 2019, to reduce power to approximately 98 percent maximum facility output, during a planned distribution line outage. This request was lifted on June 15, 2020. On April 1, 2020, operators lowered power to 69 percent at the request of the grid operator for planned line work, and returned to full power the following day. On April 15, 2020, operators lowered power to 70 percent for retrieval of a foreign material from the cooling tower basin, and the unit was returned to full power the same day. On May 25, 2020, operators lowered power to 70 percent for water box cleaning, returning the unit to full power the same day. On June 20, 2020, operators lowered power to 63 percent for a rod sequence exchange, returning to full power the same day. On June 21, 2020, operators lowered power to 72 percent to repair a leak on the electrohydraulic control system. While performing this repair, an additional component failure required operators to lower power to 17 percent on June 25, 2020.

Operators commenced power ascension the same day. On June 27, 2020, operators lowered power from 84 percent to 64 percent for a rod pattern adjustment, returning the unit to full power the same day. On June 30, 2020, operators lowered power to 83 percent for a rod pattern adjustment and returned to full power the same day.

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 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 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 the summer readiness of the offsite primary and backup alternating current systems and for the unit main condensers on May 11, 2020.

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 impending severe weather anticipated strong winds and thunderstorms on April 9, 2020.

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. (1) Unit 1, division 2 residual heat removal (RHR) and core spray during division 1 outage on April 1, 2020
2. (2) Spent fuel pool cooling system on April 15, 2020
3. (3) Unit 1, containment instrument gas (CIG) system during 'B' CIG compressor

   maintenance on June 1, 2020

4. (4) Unit Common, 'D' emergency diesel generator (EDG) and 'B' loop emergency service

   water (ESW) during 'B' EDG repair and testing on June 9, 2020

Complete Walkdown Sample (IP Section 03.02) (1 Sample)

(1) The inspectors evaluated system configurations during a complete walkdown of the Unit 1 'B' loop core spray system on April 15, 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 1, drywell general area (fire zone 1-4F) on April 1, 2020
2. (2) Unit 1, main steam tunnel area (fire zone 1-4G) on April 9, 2020
3. (3) Unit 1, spent fuel pool cooling heat exchanger and pump room (fire zone 1-5D) on

   April 15, 2020

4. (4) Unit 1, equipment space elevation 683', (fire zone 1-3C) on April 21, 2020
5. (5) Unit 2, control structure upper relay room elevation 754', (fire zone 0-27A) on

   May 26, 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, remote shutdown panel on May 28, 2020

71111.07A - Heat Sink Performance
Annual Review (IP Section 03.01) (1 Sample)

The inspectors evaluated readiness and performance of:

(1) Residual heat removal service water (RHRSW) heat exchanger 1A inspection on April 22, 2020

71111.08G - Inservice Inspection Activities (BWR)

BWR Inservice Inspection Activities Sample - Nondestructive Examination and Welding Activities (IP Section 03.01) (1 Sample)

(1) The inspectors verified that the reactor coolant system boundary, reactor vessel internals, risk-significant piping system boundaries, and containment boundary are appropriately monitored for degradation and that repairs and replacements were appropriately fabricated, examined and accepted by reviewing the following activities from April 6, 2020 to April 10, 2020:

03.01.a - Nondestructive Examination and Welding Activities.

1. ASME IWL General Visual (VT3) inspection of Drywell Exterior Concrete Surfaces in Zones 13, 14, 15, 16 (VT-20-014)
2. Radiographic Test on Reactor Water Cleanup Pipe Weld (DBB122-1 FW-4A) (BOP-RT-20-001) in conjunction with EC 2269140 “Install Stainless Steel Piping for DBB122-1 Downstream of FEG332N040”
3. Liquid Penetrant Test on Radiographic Test on Reactor Water Cleanup Pipe Weld (DBB122-1 FW-4A) (BOP-PT-20-22) in conjunction with EC 2269140 “Install Stainless Steel Piping for DBB122-1 Downstream of FEG332N040”
4. Magnetic Particle Examination of Residual Heat Removal Piping Lugs (GBB1151-HW-5A, B, C and D) (MT-20-001, 002, 003 and 004)
5. Ultrasonic Examination of the H4 Core Shroud Weld (CNF-SSES1-2)
6. In-vessel Visual Inspection Enhanced Visual Test (EVT-1) of N2H Jet Pump Riser Welds RS-1A and RS-2 (1-AUG9.1340 and 1-AUG9.1341)
7. Ultrasonic Examination of the A and B Inboard Main Steam Isolation Valve (MSIV) upstream weld VNBB212-FW-A4 and B4 (UT-20-020 and UT-20-005)
8. Replacement of Reactor Water Cleanup System Carbon Steel Piping with Flow Accelerated Corrosion Resistant Stainless Steel Piping (EC 2269140)

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 Unit 1 reactor startup following a refueling outage on April 23, 2020.

Licensed Operator Requalification Training/Examinations (IP Section 03.02) (1 Sample)
(1) The inspectors observed and evaluated simulator training on May 18, 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 1, CIG system reliability and corrective actions for CIG header loss of pressure on April 16, 2020 (CR 2020-05868)

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. (1) Unit 2, yellow risk during division 1 ESW pipe replacement on April 3, 2020
2. (2) Unit 2, protected equipment during 1B210 outage on April 3, 2020
3. (3) Unit 1, yellow shutdown risk during common RHR piping suction maintenance on April 15, 2020
4. (4) Unit 2, yellow risk during automatic depressurization system (ADS) level calibrations and testing on April 28, 2020

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 1, broken pivot stud on high pressure containment isolation (HPCI) turbine fulcrum bracket spring mount identified on March 31, 2020
2. (2) Unit Common, ‘A’ and ‘B’ phase breaker stabs misaligned on feeder to motor operated valve for RHRSW/RHR crosstie valve on April 28, 2020
3. (3) Unit 1, functionality determination on refueling hoist lowering without command on April 30, 2020
4. (4) Unit 1, 125V direct current engineered safeguards system (ESS) distribution panel breaker failed as found short and long time delay testing on May 5, 2020
5. (5) Unit 1, RHR pressure relief valve failed as found testing May 12, 2020
6. (6) Unit Common, 'E' EDG failed to fully sequence during testing on June 17, 2020
7. (7) Unit 1, suppression pool vacuum breaker support not torqued to specification

June 29, 2020

71111.18 - Plant Modifications

Temporary Modifications and/or Permanent Modifications (IP Section 03.01 and/or 03.02) (3 Samples)

The inspectors evaluated the following temporary or permanent modifications:

1. (1) Engineering Change (EC) 1958923 Unit 1, ESW Loop 'A' supply and return piping replacement, on April 15, 2020
2. (2) Design Equivalent Change (DEC) 2247720, Feedwater Check Valve Replacement Graphite Cover Gasket, on April 20, 2020
3. (3) Removal of CT-7 from Main Transformer 1X102 on May 4, 2020

71111.19 - Post-Maintenance Testing
Post-Maintenance Test Sample (IP Section 03.01) (10 Samples)

The inspectors evaluated the following post maintenance test activities to verify system operability and functionality:

1. (1) Unit Common, division 1 ESW buried piping replacement on April 6, 2020
2. (2) Unit 1, 1X210 ESS transformer replacement on April 6, 2020
3. (3) Unit 1, RHRSW Loop 'A' after scheduled maintenance on April 6, 2020
4. (4) Unit 1, 'A' loop RHR after valve work on April 8, 2020
5. (5) Unit 1, 'B' loop RHRSW after valve replacement on April 13, 2020
6. (6) Unit 1, 'B' loop RHR after valve maintenance on April 13, 2020
7. (7) Unit 1, suppression pool vacuum breaker on April 13, 2020
8. (8) Unit 1, hydrostatic leak testing following system restoration on April 19, 2020
9. (9) Unit 1, electro-hydraulic control system after repairing speed control system on May 1, 2020
10. (10) Unit 1, 'B' RHRSW pump after lift setting adjustment on May 13, 2020

71111.20 - Refueling and Other Outage Activities Refueling/Other Outage Sample (IP Section 03.01) (1 Sample)

(1) The inspectors evaluated Unit 1 refueling outage 21 activities from March 23, 2020 to May 14, 2020.

71111.22 - Surveillance Testing
The inspectors evaluated the following surveillance tests: Surveillance Tests (other) (IP Section 03.01) (2 Samples)

1. (1) Unit 2, division 2 core spray flow verification on April 17, 2020
2. (2) Unit 1, turbine trip bypass logic from start up on May 10, 2020

Inservice Testing (IP Section 03.01) (1 Sample)
(1) Unit 1, 'A' loop RHR flow surveillance on June 4, 2020

Containment Isolation Valve Testing (IP Section 03.01) (1 Sample)
(1) Unit 1, main steam isolation valve as-found leak rate testing on April 27, 2020

FLEX Testing (IP Section 03.02) (1 Sample)
(1) Unit 1, hardened containment vent valve local leak rate testing on April 7, 2020

71114.06 - Drill Evaluation
Drill/Training Evolution Observation (IP Section 03.02) (1 Sample)

The inspectors evaluated:

(1) Observation of training evolution and Emergency Action Level (EAL) classification – Anticipated Transient without a Scram (ATWS) and radiological release (simulator training) on May 18, 2020

RADIATION SAFETY

71124.01 - Radiological Hazard Assessment and Exposure Controls Instructions to Workers (IP Section 03.02) (1 Sample)

(1) The inspectors evaluated radiological protection-related instructions to plant workers.

Radiological Hazards Control and Work Coverage (IP Section 03.04) (1 Sample)

The inspectors evaluated in-plant radiological conditions during facility walkdowns and observation of radiological work activities.

(1) Reactor Water Clean Up Heat Exchanger Room Piping Replacement under radiation work permit (RWP) 20201126. This work involved high dose rates and high contamination controls.

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 1 (January 1, 2019- December 31, 2019)
2. (2) Unit 2 (January 1, 2019- December 31, 2019)

BI02: RCS Leak Rate Sample (IP Section 02.11) (2 Samples)

1. (1) Unit 1 (January 1, 2019- December 31, 2019)
2. (2) Unit 2 (January 1, 2019- December 31, 2019)

71152 - Problem Identification and Resolution Semiannual Trend Review (IP Section 02.02) (1 Sample)

(1) The inspectors reviewed Susquehanna's corrective action program (CAP) for trends that might be indicative of a more significant safety issue.

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) Condition Report (CR) CR-2019-03732, Multiple Unit 2 Control Rods Failed to Settle After Reset from Planned Scram on March 23, 2019, due to Control Rod Friction

71153 - Followup of Events and Notices of Enforcement Discretion Event Followup (IP Section 03.01)

Unplanned plant transient from 100 percent to 64 percent reactor power due to 104C feedwater heater level control valve failure and extraction steam isolation on June 16, 2020

Event Report (IP Section 03.02) (1 Sample)

The inspectors evaluated the following licensee event reports (LERs):

(1) LER 2020-001-00, Manual scram due to rising condenser back pressure (ADAMS Accession No. ML20098F706). The inspectors determined that the cause of the condition described in the LER was not reasonably within the licensee's ability to foresee and correct and therefore was not reasonably preventable. No performance deficiency nor violation of NRC requirements was identified.

INSPECTION RESULTS

Turbine trip stop and control valve closure scram inoperable during plant start up

Cornerstone

Mitigating Systems

Significance

Green
NCV 05000387/2020002-01 Open/Closed

Cross-Cutting Aspect
[H.11] - Challenge the Unknown

Report Section 71111.22

NRC inspectors identified a Green finding and associated non-cited violation (NCV) of Technical Specification (TS) 5.4.1, "Procedures," when control room operators raised reactor power above 25 percent while the reactor protection system (RPS) scram bypass function was inoperable, as indicated by the turbine control fast closure and stop valve trip bypass alarm being annunciated contrary to procedure GO-100-002, "Plant Startup, Heatup, and Power Operations."
Description: The turbine stop valve and control valve closure inputs are two of the reactor scram signals generated by the RPS in the event of a turbine trip at reactor thermal power (RTP) >26%. Turbine stop valve closure inputs to the RPS come from position switches mounted on the four turbine stop valves. Each switch opens before the valve is more than 10% closed to provide the earliest positive indication of closure. Either of the two channels associated with one stop valve can signal valve closure. The logic is arranged so that closure of three or more valves initiates a scram. Turbine control valve fast closure inputs to the RPS come from oil line pressure switches on each of four fast acting control valve hydraulic mechanisms. These hydraulic mechanisms are part of the turbine control and are used to affect fast closure of the turbine control valves. These pressure switches provide signals to the RPS. If hydraulic oil line pressure is lost, a turbine control valve fast closure scram is initiated. Diversity of trip initiation for increases in reactor vessel pressure due to termination of steam flow by turbine stop valve or control valve closure is provided by reactor vessel high pressure trip signals. A closure of the turbine stop valves or control valves at steady state conditions would result in an increase in reactor vessel pressure. If a scram was not initiated from these closures, a scram would occur from high reactor vessel pressure. Reactor vessel high pressure is an independent variable for this condition and provides diverse protective action. The turbine stop valve and control valve closure scram is an anticipatory trip to prevent a rapid power increase resultant from increased reactor pressure. The turbine stop valve closure scram and turbine control valve fast closure scram are automatically bypassed during low power operation. TS Limiting Condition for Operation (LCO) 3.3.1.1 allows the trip to be bypassed during lower power operations to prevent inadvertent scram signals, e.g. plant start up. Closure of these turbine valves below a low

initial power level does not threaten the integrity of any radioactive material release barrier. Turbine stop valve closure and turbine control valve fast closure trip bypass is affected by four pressure switches associated with the turbine first stage. Any one channel in a bypass state produces a control room annunciation. The switches are arranged so that no single failure can prevent a turbine stop valve closure scram or turbine control valve fast closure scram. In addition, this bypass automatically clears as power is raised above the setpoint for the pressure switches, but per plant TSs this bypass must clear when greater than or equal to 26% of RTP.

Susquehanna's procedure GO-100-002, Revision 113, "Plant Startup, Heatup, and Power Operation," defines actions required at various RTP levels during startup. In the notes to Step 5.74 it is stated that the bypass clears at approximately 22% RTP, and Step 5.77 specifies actions required at or before reaching 25% RTP, including, but not limited to, ensuring that the stop and control valve bypass annunciators have cleared, and to record the power at which this occurs. Alarm Response Procedure AR-103-001, Revision 58, specifies control room operator actions. In section 2, "Operator Actions" for annunciator E03, "Turbine control fast closure and stop valve trip bypass," step 4 specifies that operators "ensure alarm clears prior to 26% RTP."

On May 2, 2020, operators performed GO-100-002, and the bypass signal cleared at 20.8% RTP. This value is consistent with prior start ups, and the approximate target value established by Instrumentation and Control (I&C) procedure IC-158-002, "Calibration of Turbine First Stage Pressure Channels PSH-C72-1N003A, B, C, D, (Turbine Valve Closure Scram and EOC/RPT Bypass)."

On May 9, 2020, while performing procedure GO-100-002, Revision 113, "Plant Startup, Heatup, and Power Operation," operators were in the process of plant start up. The plant was steady for a period of approximately 10 minutes at 24.1% RTP with the turbine stop valve and control valve closure RPS scram still in bypass. Operators noted this condition and decided to proceed with power maneuvers. Operators then withdrew additional control rods and the plant stabilized at 28.5% RTP. During this time, the RPS scram bypass signal did not clear, and control room operators entered LCO 3.3.1.1 for RPS trip capability, and evaluated the applicability of LCO 3.3.4.1, EOC/RPT. It was later determined that LCO 3.3.4.1 did not require entry into action statements since the reactor was operating within the specified limits. The station promptly downpowered to approximately 23%, exiting the condition of applicability, and the station investigated why the scram bypass did not clear. Upon investigation, the station identified that a 1-inch line had severed, resulting in the K9A and K9B relays of the turbine trip logic not sensing sufficient pressure to disable the bypass. The bypass logic uses two divisions (A and B), which are fed by the four relays (A&C for division A, B&D for division B). Under normal conditions, with the bypass disabled, a trip on either channel within a division results in a scram signal from that division. The normal scram logic is either A or C concurrent with either B or D. With both the A and B relays unable to clear the bypass, only one channel in each division was enabled, that is, a scram signal from both C and D would be required to scram the reactor on a stop or control valve closure signal at >26% RTP.

Corrective Actions: The unit was promptly maneuvered to below 26% RTP, repaired the severed steam line, and identified a corrective action to modify the GO-100-002 procedure to preclude recurrence.

Corrective Action References: CR-2020-07186, CR-2020-07187, CR-2020-07351

Performance Assessment: Performance Deficiency: The failure to implement station procedures on May 9, 2020, was a performance deficiency because it was within Susquehanna's ability to reasonable foresee and correct and should have been prevented. Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the Human 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. This determination was informed by IMC 0612,

Appendix E, Example 2.f, because the performance deficiency was not administrative in nature and adversely impacted the mitigating systems cornerstone. Specifically, due to operators failing to execute procedural steps as written, power exceeded the allowed value (26% RTP) while the RPS logic function was degraded due to loss of multiple channels on two RPS functions.

Significance: The inspectors assessed the significance of the finding using Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." This finding was determined to be Green because it did not affect an RPS trip signal and the function of the redundant or diverse methods of reactor shutdown, it did not involve unintentional positive reactivity changes, and it did not result in mismanagement of reactivity by operators.

Cross-Cutting Aspect: H.11 - Challenge the Unknown: Individuals stop when faced with uncertain conditions. Risks are evaluated and managed before proceeding. Step 5.77 of the GO-100-002 procedure specifies actions required to be completed at less than or equal to 25% RTP. Step 4 of AR-103-001 specifies that operators ensure the turbine trip bypass annunciator clears prior to 26% RTP. On May 9, 2020, operators did note that while stable at 24.1% RTP, the bypass alarm had not cleared. Previous startups indicated this alarm cleared at approximately 21% RTP, including a startup performed on May 2, 2020. Rather than suspend reactivity manipulations and perform an investigation, operators proceeded with withdrawing control rods while expecting the alarm to clear between 24% and 26% RTP. If operators had halted plant maneuvers to investigate and address the cause when not receiving an expected response, the station would not have entered into the condition of applicability with the stop and control valve RPS scram function inoperable.

Enforcement:

Violation: Susquehanna's TS 5.4.1, "Procedures," specifies, in part, that written procedures shall be established, implemented, and maintained as described in Regulatory Guide 1.33, Revision 2, Appendix A, which includes, but is not limited to general plant operating procedures, start-up procedures, and procedures for abnormal, off normal, or alarm conditions.

Susquehanna procedure GO-100-002, Revision 113, "Plant Startup, Heatup, and Power Operation," defines actions required at various RTP levels during startup. Step 5.77 requires that when reactor power is =25%, the licensee shall ensure, in part, that when the turbine control valve fast closure and stop valve trip bypass annunciator clears, the percent of core thermal power is recorded. The stated acceptance criteria is =26%.

Susquehanna procedure AR-103-001, Revision 58, specifies control room operator actions for the turbine control valve fast closure and stop valve trip bypass annunciator alarm.

Step 2.4 specifies that operators shall ensure the alarm clears prior to 26% RTP.

Contrary to the above, on May 9, 2020, Susquehanna failed to implement procedures as described in Regulatory Guide 1.33, Revision 2, Appendix A, pertaining to plant start up and alarm conditions. Specifically, during plant startup, while at 24.1% RTP, the turbine stop valve and control valve closure trip bypass alarm remained annunciated. However, rather than ensuring that the annunciator cleared, the licensee resumed power ascension to 28.5% RTP .

Enforcement Action: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.

Observation: Multiple Unit 2 Control Rods Failed to Settle After Reset from 71152 Planned Scram on March 23, 2019 due to Control Rod Friction

The inspectors performed an in-depth review of Susquehanna's actions following the failure of multiple controls rods to settle on Unit 2 after resetting the scram signal following a planned reactor scram on March 23, 2019, in preparation for a refueling outage. The licensee documented the issue in CR-2019-03732. The licensee performed an apparent cause evaluation on the issue which included an extent-of-condition on Unit 1. Similar failures of control rods to settle were experienced on Unit 1 on March 28, 2020, following a planned reactor scram for a refueling outage (CR-2020-04294). The inspectors reviewed the cause analysis, technical evaluations performed, and the corrective actions taken and planned. The inspectors assessed Susquehanna's problem identification threshold, prioritization of the issue, apparent cause analysis, use of operating experience, and timeliness of corrective actions.

The inspectors observed that Susquehanna's apparent cause evaluation for the issue provided a thorough and detailed evaluation of the event. The direct cause for the control rods failing to settle following scram reset was determined to be due to fuel channel deformation causing friction between the fuel channel and the associated control rod. Fuel channel bowing resulting in channel-to-control rod friction is a known phenomenon in boiling water reactors. The evaluation included a detailed technical review of factors that affect fuel channel deformation including comparisons of unit fuel cycles and between Units 1 and 2. The apparent cause was due to Susquehanna's channel management program threshold not properly predicting channel deformation for the fuel cycles. A change in channel supply vendors (Veridiam to Kobe Steel Limited) for some fuel cycles resulted in unpredicted fuel channel bowing magnitude in the channels manufactured by Kobe Steel Limited. The channel manufacturers utilized different manufacturing techniques that apparently resulted in a change in predicted fuel channel bowing.

The inspectors observed that Susquehanna appropriately evaluated the issue, performed a thorough review of operating experience, and performed or planned timely corrective
actions. The inspectors verified that Susquehanna implemented corrective actions to resolve control rod friction issues due to Kobe Steel Limited channels during subsequent fuel cycles, to the extent practicable. For Unit 2 Cycle 20, which began operation in the Spring of 2019, the licensee revised the core design to place most of the effected fuel bundles in un-rodded pseudo-cells in the core periphery. Four fuel bundles were loaded

in rodded core locations. However, the fuel bundles were orientated to result in channel bowing away from the control rod. For Unit 1 Cycle 22, which began operation in the Spring of 2020, the effected fuel bundles placed in un-rodded pseudo-cells. Susquehanna planned actions to work with the fuel vendor to better understand Kobe Steel Limited fuel channel bowing rates and to determine further channel management and friction monitoring thresholds.

Observation: Semi-annual trend review 71152

The inspectors performed a semi-annual review of site issues to identify trends that might indicate the existence of more significant safety concerns. As part of this review, the inspectors included repetitive or closely related issues documented by Susquehanna in the CAP database, trend reports, site performance indicators, major equipment problem lists, system health reports, maintenance rule assessments, and maintenance or CAP

backlogs. The inspectors also reviewed how Susquehanna's CAP evaluated and responded to individual issues identified by the NRC inspectors during routine plant walkdowns and daily CR reviews.

Negative Human Performance Trend
The inspectors noted a marked increase in the number of human performance errors spanning the first two quarters of 2020. Specifically, the station documented 233 CRs with the Human Performance trend code in the first two quarters of 2019 and has documented 430 CRs with the Human Performance trend code in the first two quarters of 2020. While the station did lower its threshold for applying human performance trend codes to CRs, inspectors determined that this likely did not account for the entire increase and that the potential adverse trend warranted monitoring. Susquehanna recognized this adverse trend, specifically procedure use and adherence, as documented in CR-2020-03025. Inspectors noted several examples of the procedural adherence attribute of this issue.

- The inspectors reviewed the circumstances and corrective actions related to component damage caused by use of improper tools, as documented in CR-2020- 02590. On February 19, 2020, station personnel identified damage to the 1P105 vacuum pump motor shaft due to the use of a pipe wrench for manual shaft rotation. Upon review of the work instructions REWL S5049, it is clearly stated that this activity is to be completed with a strap wrench. Station corrective actions included a prompt investigation, additional oversight of preventive maintenance activities, and communicating station expectations to operations personnel for performing maintenance activities. The inspectors determined that the damage incurred was the result of a minor performance deficiency for failing to meet the requirements of the work instructions in REWL S5049, which specify the correct tools.
- The inspectors reviewed the circumstances and corrective actions related to component mispositioning, as documented in CR-2020-04283. On March 27, 2020, while performing SM-054-001, plant operators opened the incorrect link for testing. This mispositioned component revealed itself when operators proceeded through the test and the plant response was not as expected. Station corrective actions included a status control investigation and a crew clock reset for the

human performance error. The inspectors determined that this mispositioning was a minor performance deficiency for failing to meet the requirements of testing procedure SM-054-001, and a minor violation of 10 CFR Part 50, Appendix B, Criterion V, "Procedures." This violation was determined to be minor because it did not adversely affect the cornerstone objective for mitigating systems, since the error was discovered during the process of testing.

- The inspectors reviewed the circumstances and corrective actions related to foreign material exclusion and control, as documented in CR-2020-05548. On April 12, 2020, while disconnecting temporary piping sections associated with draining the Unit 1 cooling tower basin for the refueling outage, a section of piping fell into the Unit 2 cooling tower spillway, requiring a downpower to facilitate retrieval. Station corrective actions included revision of the maintenance activity to prevent recurrence. The inspectors determined that this event was a minor performance deficiency for failure to meet maintenance procedure ME-142-001.

• The inspectors reviewed the circumstances and corrective actions related to improper component alignment, as documented in CR-2020-06465. On April 24, 2020, while attempting a plant start up, Unit 1 operators received an electrohydraulic control (EHC) malfunction alarm and subsequent control valve oscillations. Upon further investigation is was discovered that the speed probe in the EHC system was not correctly installed, resulting in a gap in the probe which exceeded allowable tolerances. Station corrective actions included adding independent verification to EHC work instructions and assessed for additional training requirements. The inspectors determined this was a minor performance deficiency for operators failing to correctly install the speed probe according to work instructions because it did not adversely affect the initiating events cornerstone objective.

EXIT MEETINGS AND DEBRIEFS

The inspectors verified no proprietary information was retained or documented in this report.

• On July 30, 2020, the inspectors presented the integrated inspection results to Mr. Kevin Cimorelli and other members of the licensee staff.

August 31, 2020 – Letter from Jonathan E. Greives, Chief Reactor Projects Branch 4 Division of Reactor Projects to Brad Berryman President and Chief Nuclear Officer Susquehanna Nuclear, LLC with subject line of SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 – UPDATED INSPECTION PLAN (INSPECTION REPORTS 05000387/2020005 AND 05000388/2020005)

The enclosed inspection plan lists the inspections scheduled through June 30, 2022, for Susquehanna Steam Electric Station, Units 1 and 2. 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. 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.

In addition to baseline inspections, the NRC plans to conduct an inspection per Inspection Procedure 71003, “Post-Approval Site Inspection for License Renewal,” 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
HP Rad Instruments 1						
1, 2 06/29/2020 07/31/2020 IP 71124.01 000725 Radiological Hazard Assessment and Exposure Controls						
1, 2 07/06/2020 07/10/2020 IP 71124.05 - 2019 000729 2019 Radiation Monitoring Instrumentation						
PI&R Biennial Team Inspection - SQ 4						
1, 2 07/13/2020 07/17/2020 IP 71152B 000747 Problem Identification and Resolution						
1, 2 07/20/2020 07/24/2020 IP 71152B 000747 Problem Identification and Resolution						
1, 2 07/27/2020 07/31/2020 IP 71152B 000747 Problem Identification and Resolution						
FIRE PROTECTION-SUSQUEHANNA 3						
1, 2 09/21/2020 09/25/2020 IP 71111.21N.05 001646 Fire Protection Team Inspection (FPTI)						
1, 2 10/05/2020 10/09/2020 IP 71111.21N.05 001646 Fire Protection Team Inspection (FPTI)						
EP EXERCISE INSPECTION - SUSQUEHANNA 5						
1, 2 10/19/2020 10/23/2020 IP 71114.01 000716 Exercise Evaluation						
1, 2 10/19/2020 10/23/2020 IP 71114.04 000719 Emergency Action Level and Emergency Plan Changes						
1, 2 10/19/2020 10/23/2020 IP 71151 001397 Performance Indicator Verification						
ANNUAL SAMPLE - Internal Flooding - Susquehanna Unit 1 and 2 1						
1, 2 11/15/2020 11/21/2020 IP 71152 000748 Problem Identification and Resolution						
Susquehanna Rad Safety Inspection 1						
1, 2 11/16/2020 11/20/2020 IP 71124.01 000725 Radiological Hazard Assessment and Exposure Controls						
1, 2 11/16/2020 11/20/2020 IP 71124.03 000727 In-Plant Airborne Radioactivity Control and Mitigation						
1, 2 11/16/2020 11/20/2020 IP 71151 000746 Performance Indicator Verification						
Protective Strategy, Drill, SUS 2						
1, 2 01/11/2021 01/15/2021 IP 71130.05 000737 Protective Strategy Evaluation						
INSERVICE INSPECTION 1						
2 03/29/2021 04/02/2021 IP 71111.08G 000701 Inservice Inspection Activities (BWR)						
HP 1						
1, 2 04/05/2021 04/09/2021 IP 71124.01 000725 Radiological Hazard Assessment and Exposure Controls						
1, 2 04/05/2021 04/09/2021 IP 71124.02 000726 Occupational ALARA Planning and Controls						

This report does not include INPO and OUTAGE activities.
This report shows only on-site and announced inspection procedures.

Susquehanna

07/01/2020 - 06/30/2022

IP 22 Inspection Activity Plan Report

Unit	Start	End	Activity	CAC	Title	Staff Count
HEAT SINK INSPECTION 1						
1, 2 05/16/2021 05/22/2021 IP 71111.07T 000700 Heat Sink Performance						
FY21 Susquehanna Initial Examination 4						
1, 2 05/23/2021 05/28/2021 OV 000956 VALIDATION OF INITIAL LICENSE EXAMINATION (OV)						
1, 2 07/25/2021 08/06/2021 EXAD 000500 LICENSE EXAM ADMINISTRATION (EXAD)						
REMP 1						
1, 2 05/24/2021 05/28/2021 IP 71124.07 - 2019 000731 2019 Radiological Environmental Monitoring Program						
RETS 1						
1, 2 06/21/2021 06/25/2021 IP 71124.06 - 2019 000730 2019 Radioactive Gaseous and Liquid Effluent Treatment						
FORCE-ON-FORCE PLANNING AND EXERCISE WEEKS - SQ 6						
1, 2 06/28/2021 07/02/2021 IP 71130.03 000735 Contingency Response - Force-On-Force Testing						
1, 2 07/19/2021 07/23/2021 IP 71130.03 000735 Contingency Response - Force-On-Force Testing						
EP Program Inspection - Susuquehanna 1						
1, 2 07/19/2021 07/23/2021 IP 71114.02 000717 Alert and Notification System Testing						
1, 2 07/19/2021 07/23/2021 IP 71114.03 000718 Emergency Response Organization Staffing and Augmentation System						
1, 2 07/19/2021 07/23/2021 IP 71114.04 000719 Emergency Action Level and Emergency Plan Changes						
1, 2 07/19/2021 07/23/2021 IP 71114.05 000720 Maintenance of Emergency Preparedness						
1, 2 07/19/2021 07/23/2021 IP 71151 001397 Performance Indicator Verification						
HP 1						
1, 2 08/23/2021 08/27/2021 IP 71124.08 000732 Radioactive Solid Waste Processing & Radioactive Material Handling, Storage, & Transportation						
1, 2 08/23/2021 08/27/2021 IP 71151 000746 Performance Indicator Verification						
Evaluations of Changes, Tests and Experiments 3						
1, 2 09/12/2021 09/18/2021 IP 71111.17T 000709 Evaluations of Changes, Tests, and Experiments						
Access Control, Equipment Testing and Maintenance, Training, SPR 3						
1, 2 10/18/2021 10/22/2021 IP 71130.02 000734 Access Control						
1, 2 10/18/2021 10/22/2021 IP 71130.04 000736 Equipment Performance, Testing, and Maintenance						
1, 2 10/18/2021 10/22/2021 IP 71130.07 000739 Security Training						
1, 2 10/18/2021 10/22/2021 IP 71130.09 001656 Security Plan Changes						

This report does not include INPO and OUTAGE activities.
This report shows only on-site and announced inspection procedures.

Susquehanna

07/01/2020 - 06/30/2022

IP 22 Inspection Activity Plan Report

Unit	Start	End	Activity	CAC	Title	Staff Count
Access Control, Equipment Testing and Maintenance, Training, SPR						3
1, 2	10/18/2021	10/22/2021	IP 71151	001338	Performance Indicator Verification	
Design Basis Assurance Inspection - Programs - Power Operated Valves - Susquehanna Units 1 and 2						3
1, 2	11/01/2021	11/05/2021	IP 71111.21N.02	001645	Design-Basis Capability of Power-Operated Valves Under 10 CFR 50.55a Requirements	
1, 2	11/15/2021	11/19/2021	IP 71111.21N.02	001645	Design-Basis Capability of Power-Operated Valves Under 10 CFR 50.55a Requirements	
SQ Requal Inspection with P/F Results						2
1, 2	11/15/2021	11/19/2021	IP 71111.11A	000703	Licensed Operator Requalification Program and Licensed Operator Performance	
1, 2	11/15/2021	11/19/2021	IP 71111.11B	000704	Licensed Operator Requalification Program and Licensed Operator Performance	
License Renewal Phase 2						4
1, 2	01/10/2022	01/28/2022	IP 71003	000687	Post-Approval Site Inspection for License Renewal	
Access Authorization, FFD- SUS						2
1, 2	01/24/2022	01/28/2022	IP 71130.01	000733	Access Authorization	
1, 2	01/24/2022	01/28/2022	IP 71130.08	000740	Fitness For Duty Program	
License Renewal - Phase 1 - Susquehanna Unit 1						2
1	03/28/2022	04/01/2022	IP 71003	000687	Post-Approval Site Inspection for License Renewal	
Inservice Inspection - Susquehanna Unit 1						1
1	03/28/2022	04/01/2022	IP 71111.08G	000701	Inservice Inspection Activities (BWR)	

September 22, 2020 – Letter from Gregory F. Suber, Deputy Director Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to Kevin Cimorelli Site Vice President Susquehanna Nuclear, LLC with subject of SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 – EXEMPTION FROM CERTAIN REQUIREMENTS OF 10 CFR PART 73, APPENDIX B, "GENERAL CRITERIA FOR SECURITY PERSONNEL" (EPID L-2020-LLE-0094 [COVID-19])

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has approved the 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 Susquehanna Steam Electric Station, Units 1 and 2 (Susquehanna). This action is in response to Susquehanna Nuclear, LLC's (the licensee) application dated August 18, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20231A750 (non-public, withheld under 10 CFR 2.390)), that requested a temporary exemption from 10 CFR Part 73, Appendix B, Section VI, subsection C.3.(l)(1), regarding annual force-on-force (FOF) exercises at Susquehanna.

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 exercises is to ensure that the site security force maintains its contingency response readiness. Participation in these exercises also supports the requalification of security force members.

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.

Susquehanna Nuclear, LLC's August 18, 2020, application states the following:

- • This temporary exemption supports isolation restrictions (e.g., social distancing, group size limitations, self-quarantining, etc.) necessary to protect required site personnel in response to the COVID-19 virus.
- • This exemption is needed to ensure personnel are isolated from the COVID-19 virus and remain capable of maintaining plant security.
- • Susquehanna began implementing isolation restrictions for site personnel on March 4, 2020, and implemented additional isolation restrictions after the Governor of the Commonwealth of Pennsylvania issued a disaster declaration on March 6, 2020.
- • Susquehanna 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.
- • Susquehanna will ensure contingency response readiness of security personnel not participating in an annual FOF exercise by conducting a scenario-based tabletop exercise.
- • Susquehanna will complete the FOF exercise within the time period in this request (i.e., prior to 90 days after the PHE is ended, or December 31, 2020, whichever occurs first) when isolation restrictions are ended.
- • Susquehanna will begin implementing COVID-19 PHE controls for managing personnel performing security program duties upon NRC approval of Susquehanna Nuclear, LLC's August 18, 2020, exemption request.

1. This temporary exemption is specific to Susquehanna security personnel who have previously demonstrated proficiency and are currently qualified in accordance with the requirements of
10 CFR Part 73, Appendix B, Section VI. Susquehanna Nuclear, LLC stated that given the rigorous nature of Susquehanna's nuclear security personnel training programs, which consist of regularly scheduled training activities to include weapons training, contingency response drills and exercises, and demonstrated acceptable performance of day-to-day job activities (e.g., detection and

assessment, patrols, searches, and defensive operations), it is reasonable to conclude that security personnel will continue to maintain their proficiency even though the requalification periodicity is temporarily exceeded. Additionally, the August 18, 2020, request identified site-specific COVID-19 PHE (ADAMS Accession No. ML20105A483). Susquehanna Nuclear, LLC requested that the duration of the exemption be in effect for 90 days after the PHE is ended or until December 31, 2020, whichever occurs first, consistent with the NRC staff's April 20, 2020 letter.

Pursuant to 10 CFR 73.5, "Specific exemptions," the Commission may, upon application by any interested person or on its own initiative, grant exemptions from the requirements of 10 CFR Part 73 when the exemptions are authorized by law, 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 Susquehanna Nuclear, LLC will implement for the duration of the exemption, including a tabletop exercise, and completing the annual FOF exercise within the time period for this exemption, the NRC staff has reasonable assurance that the security force at Susquehanna will maintain its proficiency and 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 annual 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.(l)(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 Susquehanna, 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 this 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. 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 facility 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; 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 Susquehanna from the annual FOF exercise requalification requirement of security personnel in subsection C.3.(l)(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 Susquehanna project manager, Sujata Goetz, at 301-415-8004 or Sujata.Goetz@nrc.gov.

October 27, 2020 – Letter from Glenn T. Dentel, Chief Engineering Branch 2 Division of Reactor Safety to Brad Berryman President and Chief Nuclear Officer Susquehanna, LLC with subject of SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 – TRIENNIAL FIRE PROTECTION INSPECTION REPORT 05000387/2020012 AND 05000388/2020012

On October 8, 2020, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Susquehanna Steam Electric Station, Units 1 and 2 and discussed the results of this inspection with you 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 Susquehanna Steam Electric Station, Units 1 and 2.

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: 05000387 and 05000388
License numbers: NPF-14 and NPF-22
Report numbers: 05000387/2020012 and 05000388/2020012

Enterprise numbers: I-2020-012-0006
Licensee: Susquehanna, LLC
Facility: Susquehanna Steam Electric Station, Units 1 and 2 Berwick, PA
Inspection dates: September 21, 2020 to October 8, 2020

Inspectors: C. Bickett, Senior Reactor Inspector

E. DiPaolo, Senior Reactor Inspector

D. Kern, Senior Reactor Inspector

Approved by: Glenn T. Dentel, Chief Engineering Branch 2 Division of Reactor Safety

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting a triennial fire protection inspection at Susquehanna Steam Electric Station, Units 1 and 2, 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

Inadequate Diesel-Driven Fire Pump 3-Hour Fire Barrier

Cornerstone

Mitigating Systems

Significance

Green
NCV 05000387,05000388/2020012-01 Open/Closed

Additional Tracking Items

None.

Aspect None (NPP)

Section 71111.21N.05

The team identified a finding of very low safety significance (Green) involving a non-cited violation (NCV) of Unit 1 License Condition 2.C.(6) and Unit 2 License Condition 2.C.(3) for failure to implement and maintain in effect all provisions of the approved fire protection program as described in the Fire Protection Review Report (FPRR) for the facility and as approved by the NRC. Specifically, Susquehanna failed to enclose the diesel-driven fire pump within a 3-hour rated fire enclosure as described in the FPRR.

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 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

71111.21N.05 - Fire Protection Team Inspection (FPTI)

Structures, Systems, and Components (SSCs) Credited for Fire Prevention, Detection, Suppression, or Post-Fire Safe Shutdown Review (IP Section 03.01) (4 Samples)

The inspectors verified that the following systems credited in the approved fire protection program could perform their licensing basis function:

1. (1) Fire Protection Water Supply System
2. (2) Residual Heat Removal System
3. (3) Automatic Depressurization System/Safety-Relief Valves
4. (4) Fire Barrier System

Fire Protection Program Administrative Controls (IP Section 03.02) (2 Samples)

The inspectors verified that the following fire protection program administrative controls were implemented in accordance with the current licensing basis:

1. (1) Combustible Control Program
2. (2) Fire Watch Program

Fire Protection Program Changes/Modifications (IP Section 03.03) (1 Sample)

The inspectors reviewed the following changes to ensure that they did not constitute an adverse effect on the ability to safely shutdown post-fire and to verify that fire protection program documents and procedures affected by the changes were updated:

(1) Engineering Change 1544685, Fire Pump Replacement 3

INSPECTION RESULTS

Inadequate Diesel-Driven Fire Pump 3-Hour Fire Barrier

Cornerstone

Mitigating Systems

Significance

Green
NCV 05000387,05000388/2020012-01 Open/Closed

Cross-Cutting Aspect
None (NPP)

Report Section 71111.21N.05

The team identified a finding of very low safety significance (Green) involving a non-cited violation (NCV) of Unit 1 License Condition 2.C.(6) and Unit 2 License Condition 2.C.(3)

for failure to implement and maintain in effect all provisions of the approved fire protection program as described in the Fire Protection Review Report (FPRR) for the facility and as approved by the NRC. Specifically, Susquehanna failed to enclose the diesel-driven fire pump within a 3-hour rated fire enclosure as described in the FPRR.
Description: Both the motor-driven and diesel-driven fire pumps are located in close proximity within the Circulating Water Pump House. The diesel-driven fire pump is located inside an enclosure that was described by the SSES FPRR to be 3-hour fire rated to prevent both fire pumps from being damaged by a single fire, thus affecting their capability to provide fire protection water during an event. During a tour of the Circulating Water Pump House, the team observed a gap, approximately 1-inch wide, in the junction of the diesel-driven fire pump enclosure cinder block wall and the Circulating Water Pump House interior wall. The gap was filled with an unknown board-type material that resembled polystyrene. The team questioned whether the gap and material were depicted on fire barrier drawings of the enclosure. Construction drawings of the Circulating Water Pump House showed the diesel-driven fire pump enclosure walls were 3-hour fire rated but did not depict or note the gap or the filler material. The team noted that the drawings specified the room's ceiling was a poured concrete slab with a gap between the slab and Circulating Water Pump House wall. That gap was specified to be filled with a compressible material. Susquehanna sampled the filler material, concluded it was most likely polystyrene, and that the material was combustible. As a result, the team concluded that Susquehanna failed to enclose the diesel-driven fire pump within a 3-hour rated fire enclosure as described in the FPRR, and both fire pumps were not assured protection from being damaged by a single fire. Corrective Actions: Susquehanna declared the diesel-driven fire pump room fire barrier non- functional and established an hourly fire watch as a compensatory measure. Susquehanna performed an evaluation of the configuration per NRC Generic Letter 86-10. That evaluation concluded the deficient condition did not negatively impact the barrier's ability to prevent a single fire from disabling both the motor-driven and diesel-driven fire pumps. Corrective Action References: CR-2020-13254 and CR-2020-13799

Performance Assessment:

Performance Deficiency: The team determined that the failure to enclose the diesel-driven fire pump within a 3-hour fire rated enclosure, as described in Section 4.1 of the SSES FPRR, was a performance deficiency that was within Susquehanna's ability to foresee and correct. As a result, both fire pumps (i.e., the motor-driven fire pump and the diesel-driven fire pump) were not assured protection from being damaged by a single fire.

Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the Protection Against External Factors 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, failing to enclose the diesel-driven fire pump within a 3-hour rated enclosure did not assure that both fire pumps were protected from being damaged by a single fire which could have affected their capability to provide fire protection water during a fire event. In addition, this finding is more than minor because it was similar to Example 3.g of IMC 0612, Appendix E, "Examples of Minor Issues." Regardless of the final operability or functionality, the as-found condition was such that there was reasonable doubt with respect to the availability, reliability or capability of systems.

Significance: The inspectors assessed the significance of the finding using Appendix F, "Fire Protection and Post - Fire Safe Shutdown SDP." Appendix F was applicable in this case because the finding was associated with fire water supply systems. This issue screened as Green in Step 1.4.3 because adequate fire water capacity was still available for protection of equipment important to safe shutdown in the most limiting location onsite. Susquehanna performed a detailed evaluation of the arrangement in the

Circulating Water Pump House. The evaluation concluded that the deficient condition did not negatively impact the degraded fire barrier’s ability to prevent a single fire from disabling both the motor-driven and diesel- driven fire pumps. In addition, the site is equipped with a backup diesel-driven fire pump, remote to the Circulating Water Pump House, that could be placed in service per plant operating procedures. Cross-Cutting Aspect: Not Present Performance. No cross-cutting aspect was assigned to this finding because the inspectors determined the finding did not reflect present licensee performance. The installation of the non-fire rated material used as a fire barrier was performed during the construction of the diesel-driven fire pump enclosure. Although a periodic inspection is performed on Circulating Water Pump House fire barriers, the scope of the inspection was to visually identify degradation that could prevent barriers from meeting their design function. The inspectors concluded that identification of incorrect fire barrier material was beyond the scope of the periodic inspection.
Enforcement: Violation: Unit 1 License Condition 2.C.(6) and Unit 2 License Condition 2.C.(3), in part, requires Susquehanna Nuclear, LLC to implement and maintain in effect all provisions of the approved fire protection program as described in the FPRR for the facility and as approved by the NRC. SSES FPRR, Section 4.1, Fire Protection Water Supply System, stated that the diesel-driven fire pump is enclosed within a 3-hour fire rated enclosure which prevents both fire pumps (i.e., the motor-driven fire pump and the diesel-driven fire pump) from being damaged by a single fire. Contrary to the above, since July 17, 1982, the effective date of the Unit 1 operating license, until September 24, 2020, when the issue was entered into the corrective action program, Susquehanna failed to enclose the diesel-driven fire pump within a 3-hour fire rated fire enclosure which prevents both fire pumps from being damaged by a single fire. Enforcement Action: This violation is being treated as a non-cited violation, 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 October 8, 2020, the inspectors presented the triennial fire protection inspection results to Mr. Brad Berryman and other members of the licensee staff.

November 5, 2020 – Letter from Jonathan E. Greives, Chief Reactor Projects Branch 4 Division of Reactor Projects to Brad Berryman Senior Vice President and Chief Nuclear Officer Susquehanna Nuclear, LLC with subject of SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 – INTEGRATED INSPECTION REPORT 05000387/2020003 AND 05000388/2020003

On September 30, 2020, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Susquehanna Steam Electric Station, Units 1 and 2. On October 15, 2020, the NRC inspectors discussed the results of this inspection with Mr. Kevin Cimorelli, 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: 05000387 and 05000388
License numbers: NPF-14 and NPF-22
Report numbers: 05000387/2020003 and 05000388/2020003

Enterprise identifiers: I-2020-003-0038
Licensee: Susquehanna Nuclear, LLC
Facility: Susquehanna Steam Electric Station, Units 1 and 2

Location: Berwick, PA
Inspection dates: July 1, 2020, to September 30, 2020

Inspectors: M. Hardgrove, Senior Resident Inspector
C. Highley, Senior Resident Inspector
M. Rossi, Resident Inspector
H. Anagnostopoulos, Senior Health Physicist

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 Susquehanna Steam Electric Station, Units 1 and 2, 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 1 began the inspection period at 100 percent power. On July 31, 2020, operators reduced power to approximately 60 percent for a rod sequence exchange. The unit was returned to full power on August 4, 2020. The unit remained at or near 100 percent power for the remainder of the inspection period.

Unit 2 began the inspection period at 100 percent power. On September 11, 2020, operators reduced power to approximately 60 percent for a rod sequence exchange. The unit was returned to 100 percent power on September 15, 2020, and remained at or near 100 percent power 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
External Flooding Sample (IP Section 03.03) (1 Sample)

(1) The inspectors evaluated readiness to cope with external flooding on August 17, 2020.

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 2, 'B' loop residual heat removal during 'A' loop maintenance on August 3, 2020
2. (2) Unit Common, 'E' emergency diesel generator (EDG) during 'D' EDG system outage

   window on August 3, 2020

3. (3) Unit 1, 'B' loop residual heat removal service water during 'A' loop heat exchanger

   maintenance on September 22, 2020

Complete Walkdown Sample (IP Section 03.02) (1 Sample)

(1) The inspectors evaluated system configurations during a complete walkdown of Units 1 and 2 residual heat removal service water system on July 7, 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 2, 683' elevation equipment space (fire zone 2-3C) on July 14, 2020

2. (2) Unit Common, diesel generator bay 'B' during planned maintenance window (fire

   zone 0-41B) on July 29, 2020

3. (3) Unit 2, 719' elevation containment instrument gas compressor area (fire zone

   2-4A-N/W/S) on August 26, 2020

4. (4) Unit Common, 'D' diesel generator bay following 'D' EDG system outage window

   (fire zone 0-41D) on August 31, 2020

5. (5) Unit 2, 754' elevation upper relay room (fire zone 0-27A) on September 22, 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 2 683' and 670' elevations of the reactor building general areas on September 1 to 2, 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 Unit 1 rod sequence exchange and pattern adjustment on July 31, 2020.

Licensed Operator Requalification Training/Examinations (IP Section 03.02) (1 Sample)
(1) The inspectors observed and evaluated simulator training on August 4, 2020.

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, refueling bridge and hoist on August 18, 2020
2. (2) Unit 1, main steam isolation valve leakage trending and maintenance planning on

September 17, 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 2, 2A emergency service water core spray room cooler throttle valve leaking through wall on July 31, 2020

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. (1) Units 1 and 2, yellow risk during automatic depressurization system (ADS) drywell pressure bypass timer and ADS timer permissive testing on July 8, 2020
2. (2) Units 1 and 2, yellow risk during reactor pressure vessel level functional tests with 2A core spray room cooler isolated and 2A core spray declared inoperable for maintenance on July 21, 2020
3. (3) Unit Common, 'D' EDG system outage window - 5-year overhaul on August 10, 2020
4. (4) Unit 2, temporary design change to provide main steam pressure to main steam

   line A instrument indication on September 8, 2020

71111.15 - Operability Determinations and Functionality Assessments
Operability Determination or Functionality Assessment (IP Section 03.01) (5 Samples)

The inspectors evaluated the licensee's justifications and actions associated with the following operability determinations and functionality assessments:

1. (1) Unit 1, reactor core isolation cooling system one of two anti-rotational bolts found loose on July 20, 2020
2. (2) Units 1 and 2, fuel pool cooling pump bearing degradation on August 11, 2020 5

(3) Unit 1, 'B' residual heat removal service water pump failed to start on demand on August 19, 2020

3. (4) Unit 1, 'D' main steam isolation valve pressure switch stuck on September 2, 2020
4. (5) Unit 2, 'D' low pressure coolant injection permissive switch found out of tolerance on

   September 16, 2020

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. (1) Unit 2, EC 2365089, temporary design change to provide main steam pressure to main steam line A instrument indication on September 3, 2020
2. (2) Unit 1, EC 2276120, permanent modification for 'A' loop residual heat removal valve actuator replacement on September 14, 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 2, ventilation exhaust rad monitoring system (VERMS) 'A' bypass pump failed on July 14, 2020
2. (2) Unit 2, 'A' core spray room cooler emergency service water piping valve replacements on July 22, 2020
3. (3) Unit 2, residual heat removal 'A' loop motor cooler maintenance window on August 3, 2020
4. (4) 'C' EDG repair due to fuel oil seepage on August 6, 2020
5. (5) Unit Common, 'D' EDG scheduled outage window on August 10, 2020
6. (6) Unit 2, 'A' main steam line low pressure instrumentation temporary modification

   installation on September 10, 2020

71111.22 - Surveillance Testing

The inspectors evaluated the following surveillance tests: Inservice Testing (IP Section 03.01) (1 Sample)

(1) Unit 2, high-pressure coolant injection quarterly start and flow surveillance on September 10, 2020

RCS Leakage Detection Testing (IP Section 03.01) (1 Sample)
(1) Unit 2, drywell floor drain sump level channels monthly test on August 3, 2020

71114.06 - Drill Evaluation
Drill/Training Evolution Observation (IP Section 03.02) (1 Sample)

The inspectors evaluated:

(1) Integrated on-site and off-site fire response drill on September 17, 2020

RADIATION SAFETY

71124.01 - Radiological Hazard Assessment and Exposure Controls Contamination and Radioactive Material Control (IP Section 03.03) (2 Samples)

The inspectors evaluated licensee processes for monitoring and controlling contamination and radioactive material.

1. (1) Observed the egress of personnel and material from the Unit 2 radiation protection control point.
2. (2) Observed the unconditional release of gas bottles from the Unit 2 turbine building truckbay.

Radiological Hazards Control and Work Coverage (IP Section 03.04) (2 Samples)

The inspectors evaluated in-plant radiological conditions during facility walkdowns and observation of radiological work activities.

1. (1) Pre-job briefing, initial entry radiation survey, and worker entry into the Unit 2 spent fuel pool cooling pump room to collect pump vibration data.
2. (2) Briefing and unconditional release of gas bottles from the Unit 2 turbine building truckbay.

High Radiation Area and Very High Radiation Area Controls (IP Section 03.05) (2 Samples)

The inspectors evaluated licensee controls of the following high radiation areas and very high radiation areas:

1. (1) Units 1 and 2 spent fuel pool
2. (2) Unit 2 spent fuel pool cooling pump room

Radiation Worker Performance and Radiation Protection Technician Proficiency (IP Section 03.06) (1 Sample)

(1) The inspectors evaluated radiation worker and radiation protection technician performance as it pertains to radiation protection requirements for initial entry into the Unit 2 spent fuel pool cooling pump room.

71124.05 - Radiation Monitoring Instrumentation
Walkdowns and Observations (IP Section 03.01) (5 Samples)

The inspectors evaluated the following radiation detection instrumentation during plant walkdowns:

1. (1) Personnel contamination monitors at the main access point
2. (2) Portal monitors at the main access point
3. (3) Tool monitors at the main access point
4. (4) Whole body counter in the dosimetry office

5. (5) Telepole in the ready-for-issue rack in the calibration laboratory

Calibration and Testing Program (IP Section 03.02) (10 Samples)

The inspectors evaluated the calibration and testing of the following radiation detection instruments:

1. (1) AMP100 AMP1-0064
2. (2) Fluke FMFM-0093
3. (3) Ludluum 43-93 L236-006
4. (4) Ludluum 43-2 L200-0013
5. (5) AMS4 AMS4-0039
6. (6) FUJI NSN3 FUJI-0007
7. (7) FUJI NSN3 FUJI-0008
8. (8) Ludluum Model 3 LUD3-0051
9. (9) Telepole POLE-0019
10. (10) Canberra Fastscan Dosimetry Office

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. (1) Standby gas treatment vent low range radiation monitor (VERMS)
2. (2) Unit 2 reactor building vent low range radiation monitor nobel gas channel (VERMS)

OTHER ACTIVITIES – BASELINE

71151 - Performance Indicator Verification
The inspectors verified licensee performance indicators submittals listed below:
MS05: Safety System Functional Failures (SSFFs) Sample (IP Section 02.04) (2 Samples)

1. (1) Unit 1 for the period of July 1, 2019, through June 30, 2020
2. (2) Unit 2 for the period of July 1, 2019, through June 30, 2020

MS06:

(1) (2)

MS07:

(1) (2)

MS08:

(1) (2)

Emergency AC Power Systems (IP Section 02.05) (2 Samples)

Unit 1 for the period of July 1, 2019, through June 30, 2020 Unit 2 for the period of July 1, 2019, through June 30, 2020

High Pressure Injection Systems (IP Section 02.06) (2 Samples)

Unit 1 for the period of July 1, 2019, through June 30, 2020 Unit 2 for the period of July 1, 2019, through June 30, 2020

Heat Removal Systems (IP Section 02.07) (2 Samples)

Unit 1 for the period of July 1, 2019, through June 30, 2020 Unit 2 for the period of July 1, 2019, through June 30, 2020

71152
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) Human performance error trend and operator crew clock resets

INSPECTION RESULTS

Observation: Human Performance Error Trend and Operator Crew Clock Resets 71152

- Problem Identification and Resolution

The inspectors reviewed Susquehanna's evaluations and corrective actions to address human performance errors as documented in various condition reports (CRs) from January 1, 2020. This time period covered the spring outage for Unit 1 and the COVID-19 pandemic. Additionally, the inspectors reviewed Level 1 through Level 3 evaluations with an associated human performance error dating back to August 2019. The inspectors determined during their review of the Level 1 through Level 3 evaluations impacted by human performance errors that Susquehanna conducted an appropriate review of each issue and implemented appropriate and timely corrective actions to address the human performance errors. The inspectors determined during their review of the selected CRs that Susquehanna was appropriately screening and implementing adequate corrective actions to address the human performance errors. Susquehanna identified a rise in human performance errors during the spring outage for Unit 1 and saw the human performance error trend decline following the outage. Overall, the inspectors determined that Susquehanna is adequately trending and addressing human performance errors through the corrective action program. The inspectors documented a negative human performance trend in Susquehanna Steam Electric Station, Units 1 and 2 – Integrated Inspection Report 05000387/2020002 and 05000388/2020002, with the licensee documenting the trend in CR-2020-03025. The corrective actions taken to address the human performance errors included procedure revisions, revisions to various maintenance work instruction revisions, training, operating experience communications to departments, and a larger management presence in the field providing observations.

The inspectors focused on human performance errors within the operations department related to crew clock resets. During the time period of review, 20 CRs captured a human performance error within the operations department without a crew clock reset, with 12 CRs being questioned by the inspectors whether there should have been a crew clock reset. The inspectors discussed with the operations department on how crew clock resets are applied and the thresholds for operating crews, as it was not clear to the inspectors when reviewing the CRs. From this discussion with the inspectors, the operations department determined 3 of the 12 CRs should have been classified as operator crew clock resets related to discretionary and non-discretionary thresholds, which has subsequently been changed. These CRs were related to missing a swap of a breaker during Unit 1 4kV A bus restoration (CR-2020-05018); performance of a high-pressure coolant injection valve stroke testing while local leak rate testing was in progress (CR-2020-05851); and missed fire drills during the first and second quarters of 2019 (CR-2020-00579). The changes made to these three CRs constitute a minor performance deficiency in accordance with IMC 01612, Appendix B, "Additional Issue Screening Guidance," because all more-than-minor questions were answered no. Crew clock resets are determined under licensee procedures NDAP-00-0032, "Human Performance (HuP) - Standards for Error and Event Prevention," Attachment C and OP-AD-300, "Administration of Operations," Attachment I. The reclassification of the crew clock resets was captured in Susquehanna's corrective action program and addressed.

The inspector's assessment of Susquehanna appropriately applying procedures, standards, and thresholds for operations crew clock is overall adequate, however, as discussed above there were some areas where the licensee had to reassess after further engagement from the inspectors.

EXIT MEETINGS AND DEBRIEFS

The inspectors verified no proprietary information was retained or documented in this report.

- On October 15, 2020, the inspectors presented the integrated inspection results to Mr. Kevin Cimorelli, Site Vice President, and other members of the licensee staff.

November 6, 2020 – Letter from Craig G. Erlanger, Director Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to Kevin Cimorelli Site Vice President Susquehanna Nuclear, LLC with subject of SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 – 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-0153 [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* (10 CFR) Part 50, Section IV.F.2.c, for Susquehanna Steam Electric Station (Susquehanna), Units 1 and 2. This action is in response to your application dated September 28, 2020, as supplemented by letter from the Pennsylvania Emergency Management Agency (PEMA) dated October 9, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML20272A020 and ML20283A772, respectively), that requested a one-time exemption from 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.

Susquehanna Nuclear, LLC (the licensee) holds Renewed Facility Operating License Nos. NPF-14 and NPF-22, which authorize operation of Susquehanna, Units 1 and 2, respectively. These licenses are subject to the rules, regulations, and orders of the Commission. The facility consists of two boiling-water reactors located in Luzerne County, Pennsylvania.

By letter dated September 28, 2020, the licensee submitted a request for temporary exemption from Appendix E to 10 CFR Part 50, Section IV.F.2.c, regarding the performance 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 ORO personnel in response to the COVID-19 PHE. These activities are needed to ensure that supporting State and local government personnel are isolated from the COVID-19 virus and remain capable of executing the functions of the emergency response organization, as described in the Susquehanna Emergency Plan, as well as other non-nuclear health and safety functions for the benefit of the public. In June 2020, the OROs notified PEMA of their concerns with supporting the biennial EP exercise and maintaining protection of offsite staff during the current COVID-19 PHE response. Based on these concerns, the needed response to the PHE, and the uncertainty of the future in this matter, seeking a one-time exemption regarding the ORO participation in the CY 2020 biennial EP exercise was determined to be the most appropriate action.

- • The threat of COVID-19 spread resulted in the inability to safely conduct, with ORO participation, the biennial EP exercise held on October 20, 2020, due to implementation of isolation activities (e.g., social distancing, group size limitations, self-quarantining, etc.). In addition, the Commonwealth of Pennsylvania and the Counties of Columbia and Luzerne, which are in the plume exposure EP zone, informed the licensee that the current COVID-19 PHE response impacted their ability to prepare for the scheduled exercise and that they would be challenged to participate in the exercise by putting an undue burden on staff and volunteers. Columbia and Luzerne Counties and the Commonwealth of Pennsylvania will maintain their current emergency plans and remain able to respond to an emergency. The exemption would not hinder the ability of Susquehanna, Units 1 and 2; Columbia and Luzerne Counties; and the Commonwealth of Pennsylvania to respond should an actual emergency occur.
- • This one-time schedular exemption to not conduct the ORO participation portion of 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 emergency response organization and ORO personnel in response to the COVID-19 PHE.
- • The last biennial EP exercise was conducted on October 16, 2018. Since that time, the licensee has conducted drills, exercises, and other training activities that have exercised its emergency response strategies, in coordination with offsite authorities. These activities included, in part, a full participation exercise on August 28, 2019, and limited participation (i.e., taking Emergency Offsite Notification Reports) exercises on June 25, July 9, August 13, September 4, and October 29, 2019, and July 28, 2020.

[1] CDC, "How to Protect Yourself and Others," April 18, 2020 (ADAMS Accession No. ML20125A069).

- • The licensee will continue to conduct drills and exercises as evidenced by its conduct of the onsite participation portion of the CY 2020 biennial EP exercise required under Appendix E to 10 CFR Part 50, Section IV.F.2.c on October 20, 2020.
- • The licensee made a reasonable effort to reschedule ORO participation in the biennial EP exercise during CY 2020 with the respective OROs but was unsuccessful. As addressed in ORO letters included in the exemption request, it was agreed that it was not feasible to schedule ORO participation in the exercise in CY 2020 or in CY 2021 due to uncertainty of COVID-19 isolation actions and conflicts with other NRC inspections.
- • The licensee also noted that the ORO will maintain its current emergency plans and remain able to respond to an emergency during the COVID-19 PHE. The exemption from ORO participation in the CY 2020 biennial EP exercise does not obviate the ability to respond should an actual emergency occur. Specifically, the letters from PEMA and Columbia and Luzerne Counties state that they remain committed to maintaining their radiological emergency plans and that they are fully prepared for and can handle any emergency throughout the COVID-19 PHE, including an actual incident at Susquehanna.

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 the Federal Emergency Management Agency (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 their 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 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 offsite participation portion of the 10 CFR Part 50, Appendix E, Section IV.F.2.c requirement for biennial EP exercises in CY 2020, with the next performance of the exercise with offsite participation 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.

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.

[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

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 biennial EP exercise with offsite participation 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."

Both PEMA and the plume exposure pathway emergency planning zone counties have informed the licensee that they support the request for a one-time exemption to exclude the participation of the OROs in the biennial EP exercise for CY 2020.

PEMA 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 Susquehanna, Units 1 and 2, with the next performance of the exercise with offsite participation 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 Susquehanna, Units 1 and 2. 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 offsite participation portion of 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 Susquehanna project manager, Sujata Goetz, at 301-415-8004 or by e-mail to Sujata.Goetz@nrc.gov.

November 18, 2020 – Letter from Fred L. Bower, III, Chief Plant Support Branch Division of Reactor Safety to Brad Berryman Senior Vice President and Chief Nuclear Officer Susquehanna Nuclear, LLC with subject of SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 – EMERGENCY PREPAREDNESS BIENNIAL EXERCISE INSPECTION REPORT 05000387/2020501 AND 05000388/2020501

On October 23, 2020, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Susquehanna Steam Electric Station, Units 1 and 2. On October 29, 2020, the NRC inspectors discussed the results of this inspection with Kevin Cimorelli, 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 number: 05000387 and 05000388
License numbers: NPF-14 and NPF-22
Report numbers: 05000387/2020501 and 05000388/2020501

Enterprise identifier: I-2020-501-0007
Licensee: Susquehanna Nuclear, LLC
Facility: Susquehanna Steam Electric Station, Units 1 and 2

Location: Berwick, PA
Inspection dates: October 19, 2020 to October 23, 2020

Inspectors: J. Ambrosini, Senior Emergency Preparedness Inspector

D. Johnson, Senior Emergency Preparedness Specialist

J. Rady, Emergency Preparedness Inspector
S. Seeley, Health Physicist

D. Silk, Senior Operations Engineer

Approved by: Fred L. Bower, III, Chief Plant Support Branch Division of Reactor Safety

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting an emergency preparedness biennial exercise inspection at Susquehanna Steam Electric Station, Units 1 and 2, 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. The inspection was conducted to assess the exercise performed to meet the regulatory requirements in 10 CFR Part 50, Appendix E, paragraph IV.F.2.b (onsite). The regulatory requirements of 10 CFR Part 50, Appendix E, paragraph IV.F.2.c (offsite) are being addressed in separate docketed correspondence (ML20272A020). 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 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 that were available for performance (ML20294A291).

REACTOR SAFETY

71114.01 - Exercise Evaluation
Inspection Review (IP Section 02.01-02.09) (1 Sample)

(1) The inspectors evaluated the biennial emergency plan exercise. The exercise scenario simulated a seismic event, core spray flooding, a reactor core isolation cooling (RCIC) steam leak, and an anticipated transient without scram (ATWS) on October 20, 2020.

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.

• Evaluations 2020-03-05-01, 2020-03-05-02, 2020-03-05-03, and 2020-03-05- 04, EP-RM-004, EAL Classification Bases, Revision 17

This evaluation does not constitute NRC approval.

71114.08 - Exercise Evaluation Scenario Review Inspection Review (IP Section 02.01 - 02.04) (1 Sample)

(1) The inspectors reviewed and evaluated the proposed scenario for biennial emergency plan exercise on September 16, 2020.

OTHER ACTIVITIES – BASELINE

71151 - Performance Indicator Verification
The inspectors verified licensee performance indicators submittals listed below: EP01: Drill/Exercise Performance (IP Section 02.12) (1 Sample)

(1) July 1, 2019 - September 30, 2020
EP02: ERO Drill Participation (IP Section 02.13) (1 Sample)

(1) July 1, 2019 - September 30, 2020
EP03: Alert & Notification System Reliability (IP Section 02.14) (1 Sample)

(1) July 1, 2019 - September 30, 2020

INSPECTION RESULTS

No findings were identified.

EXIT MEETINGS AND DEBRIEFS

The inspectors verified no proprietary information was retained or documented in this report.

• On October 29, 2020, the inspectors presented the emergency preparedness biennial exercise inspection results to Mr. Kevin Cimorelli, Site Vice President and other members of the licensee staff.

November 30, 2020 – Email from Sujata Goetz to Shane Jurek with subject of Susquehanna -Acceptance Review Replacement Of Unit 1 Transformers

By letter dated November 5, 2020 (Agencywide Document and Access Management System (ADAMS) Accession No. ML20310A231, Susquehanna Nuclear, LLC submitted a license amendment request for Susquehanna Unit 1 and Unit 2. The license amendment would temporarily extend the Completion Time for Technical Specification (TS) 3.8.7 to June 15, 2024 and be applicable only for replacement of Transformers 1X230 and 1X240.

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 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.

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. 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 300 hours to complete. The NRC staff expects to complete this review in approximately 12 months which is November 5, 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, 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.

If you have any questions, please contact me.

December 17, 2020 – Letter from Craig G. Erlanger, Director Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to Kevin Cimorelli Site Vice President Susquehanna Nuclear, LLC with subject of SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 – EXEMPTION FROM ANNUAL FORCE-ON-FORCE EXERCISE REQUIREMENT OF 10 CFR PART 73, APPENDIX B, "GENERAL CRITERIA FOR SECURITY PERSONNEL," SUBSECTION VI.C.3.(I)(1) (EPID L-2020-LLE-0222 [COVID-19])

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has approved the requested exemption from a specific requirement 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 Susquehanna Steam Electric Station, Units 1 and 2 (Susquehanna) for calendar year (CY) 2020. This action is in response to Susquehanna Nuclear, LLC's (the licensee) application dated November 24, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20329A335), that requested an exemption from 10 CFR Part 73, Appendix B, Section VI, subsection C.3.(I)(1), regarding the annual force-on-force (FOF) exercises for CY 2020 at Susquehanna.

The requirements in 10 CFR Part 73, Appendix B, Section VI, subsection C.3.(I)(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 exercises is to ensure that the site security force maintains its contingency response readiness. Participation in these exercises also supports the requalification of security force members.

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. On September 21, 2020 (ADAMS Accession No. ML20232C272), the NRC granted the licensee's previous request for temporary exemption from 10 CFR Part 73, Appendix B, Section VI, subsection C.3.(l)(1). That exemption is set to expire on December 31, 2020. As such, the licensee is required to conduct any missed annual licensee conducted FOF exercises by December 31, 2020.

The licensee's application dated November 24, 2020, stated the following:

- • The PHE has not ended and continues to impact Susquehanna's ability to conduct annual FOF exercises. Based on the licensee's current data, the trend in positive cases and quarantines is consistent with the trends in the Commonwealth of Pennsylvania and the local area of Luzerne County. As of November 9, 2020, the number of cases in the Commonwealth of Pennsylvania is on the increase.
- • A review of impacted security positions and other locations necessary for briefings and critiques has determined that annual exercises cannot be conducted due to the limited space available in most locations to allow for appropriate social distancing.
- • Approval of this exemption will continue to support the isolation protocols necessary to protect essential site personnel. These restrictions are necessary to ensure personnel are isolated from the COVID-19 disease and remain capable of maintaining plant security.
- • Impacted security personnel continue to maintain proficiency with the knowledge, skills and abilities required to effectively implement the protective strategy to protect the station against the design basis threat as described in 10 CFR 73.1, Purpose and Scope, because Susquehanna has continued to conduct the following training requalification requirements of Section VI. of Appendix B to Part 73:

    ○ Quarterlytacticalresponsedrills(Tabletopdrills,Timelinedrills,Limited-scope tactical response drills)

    ○ Annualfirearmsfamiliarization
    ○ Annualdaylightqualificationcourse
    ○ Annualnightfirequalificationcourse
    ○ Annualtacticalqualificationcourse
    ○ Annualphysicalexamination
    ○ Annualphysicalfitnesstest
    ○ Weaponsrangeactivity(4-monthperiodicity) ○ Annualwrittenexam

• In addition, and in accordance with the approved temporary exemption, Susquehanna conducted tabletop exercises and a lessons-learned review of past exercises with all impacted security personnel.

This exemption is specific to CY 2020 and Susquehanna security personnel who have previously demonstrated proficiency and are currently qualified in accordance with the requirements of 10 CFR Part 73, Appendix B, Section VI. The licensee stated that given the exemption does not change physical security plans or the defensive strategy; impacted security proposed personnel continue to maintain proficiency with the knowledge, skills and abilities required to effectively implement the protective strategy to protect the station against the design basis threat because Susquehanna has continued to conduct other training requalification requirements; and security personnel will continue to be monitored regularly by supervisory personnel and have implemented controls as identified in the temporary exemption granted on September 21, 2020, granting the requested exemption will not endanger or compromise the

Additionally, the November 24, 2020, request identified the site-specific actions listed above that have occurred, or will continue to occur, at Susquehanna to maintain contingency response readiness, consistent with the NRC staff's October 13, 2020, letter (ADAMS Accession
No. ML20273A117).

Pursuant to 10 CFR 73.5, "Specific exemptions," the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of
10 CFR Part 73 when the exemptions are authorized by law, 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 in accordance with the security requirements outlined in 10 CFR Part 73, Appendix B, Section VI. Based on this fact, and its review of the controls that the licensee will implement for the duration of the exemption, including conducting quarterly tactical response drills and other security requalification requirements, the NRC staff has reasonable assurance that the security force at Susquehanna will maintain its proficiency and 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. The NRC staff finds that the exemption from the annual FOF exercise requirement in 10 CFR Part

73, Appendix B,
Section VI, subsection C.3.(l)(1), for CY 2020 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 exemption for CY 2020 is in the public interest because it allows the licensee to maintain the required security posture at Susquehanna, 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 common defense or security, or safeguarding Susquehanna

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 facility 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; 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 NRC hereby grants the licensee's request to exempt Susquehanna from the annual FOF exercise requalification requirement of security personnel in subsection C.3.(l)(1) of 10 CFR Part 73, Appendix B, Section VI. This exemption applies only to those FOF exercises required during CY 2020.

If you have any questions, please contact the Susquehanna project manager, Sujata Goetz, at 301-415-8004 or by e-mail to Sujata.Goetz@nrc.gov.

January 21, 2021 – Letter from Sujata Goetz, Project Manager Plant Licensing Branch I Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to Kevin Cimorelli Site Vice President Susquehanna Nuclear, LLC with subject of SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 278 AND 260 TO ALLOW APPLICATION OF ADVANCED FRAMATOME ATRIUM 11 FUEL METHODOLOGIES (EPID L-2019-LLA-0153)

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 278 to Renewed Facility Operating License No. NPF-14 and Amendment No. 261 to Renewed Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Units 1 and 2, respectively. The amendments consist of changes to the technical specifications in response to your application dated July 15, 2019, as supplemented by letters dated February 6, 2020, and April 1, 2020.

The amendments allow application of the Framatome analysis methodologies necessary to support a planned transition to ATRIUM 11 fuel under the currently licensed Maximum Extended Load Line Limit Analysis operating domain.

A copy of the related safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's monthly *Federal Register* notice.

SUSQUEHANNA NUCLEAR, LLC
ALLEGHENY ELECTRIC COOPERATIVE, INC. DOCKET NO. 50-387
SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1 AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 278 Renewed License No. NPF-14

1. The U.S. Nuclear Regulatory Commission (NRC or the Commission) has found that:

1. The application for amendment filed by Susquehanna Nuclear, LLC, dated July 15, 2019, as supplemented by letters dated February 6, 2020, and April 1, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 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, 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. NPF-14 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 278, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. Susquehanna Nuclear, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to loading ATRIUM 11 fuel into the core during the spring 2022 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation

ATTACHMENT TO LICENSE AMENDMENT NO. 278 SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1 RENEWED FACILITY OPERATING LICENSE NO. NPF-14 DOCKET NO. 50-387

Replace the following pages of the Renewed Facility Operating License with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE Page 3 Page 18

INSERT Page 3 Page 18

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE INSERT 2.0-1 2.0-1

22. 5.0-22 5.0-22
23. 5.0-23 5.0-23

3. (3) Susquehanna Nuclear, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70,
to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed neutron sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

4. (4) Susquehanna Nuclear, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and
5. (5) Susquehanna Nuclear, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

1. (1) Maximum Power Level

   Susquehanna Nuclear, LLC is authorized to operate the facility at reactor core power levels not in excess of 3952 megawatts thermal in accordance with the conditions specified herein. The preoperational tests, startup tests and other items identified in License Conditions 2.C.(36), 2.C.(37), 2.C.(38), and 2.C.(39) to this license shall be completed as specified.

2. (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 278, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. Susquehanna Nuclear, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

   For Surveillance Requirements (SRs) that are new in Amendment 178 to Facility Operating License No. NPF-14, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 178. For SRs that existed prior to Amendment 178, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 178.

result of the test, the test failure shall be addressed in accordance with corrective action program requirements and the provisions of the power ascension test program prior to continued operation of the SSES Unit above 3489 MWt.

(b) Unless the NRC issues a letter notifying the licensee that the tests specified by License Condition 2.C.(37)(a) adequately demonstrate that a single condensate pump trip will not result in a loss of all feedwater while operating at the full CPPU power level of 3952 MWt, the operating licensee shall perform the transient test on either SSES unit (whichever unit is first to achieve the following specified operating conditions) specified by License Condition 2.C.(37)(a) during the power ascension test program while operating at 3872 MWt to 3952 (98% to 100% of the full CPPU power level) with feedwater and condensate flow rates stabilized. The test shall be performed within 90 days of operating at greater than 3733 MWt and within 336 hours of achieving a nominal power level of 3872 MWt with feedwater and condensate flow rates stabilized. The operating licensee will demonstrate through performance of transient testing on either Susquehanna Unit 1 or Unit 2 (whichever unit is first to achieve the specified conditions) that the loss of one condensate pump will not result in a complete loss of reactor feedwater. The operating licensee shall confirm that the plant response to the transient is as expected in accordance with the acceptance criteria that are established. If a loss of all feedwater occurs as a result of the test, the test failure shall be addressed in accordance with corrective action program requirements and the provisions of the power ascension test program prior to continued operation of either SSES Unit above 3733 MWt.

(38) Neutronic Methods

1. (a) Not Used
2. (b) Not Used

Renewed Operating License No. NPF-14

Amendment No. 278

1. 2.0 SAFETY LIMITS (SLs)
2. 2.1 SLs

2.1.1 Reactor Core SLs

PPL Rev. 5

Safety Limits (SLs) 2.0

1. 2.1.1.1 With the reactor steam dome pressure < 575 psig or core flow < 10 million lbm/hr:

   THERMAL POWER shall be ≤ 23% RTP.

2. 2.1.1.2 With the reactor steam dome pressure ≥ 575 psig and core flow

   ≥ 10 million lbm/hr:
   MCPR shall be ≥ 1.09 for two recirculation loop operation or ≥ 1.12 for

single recirculation loop operation.

3. 2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL
Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations
With any SL violation, the following actions shall be completed within 2 hours:

1. 2.2.1 Restore compliance with all SLs; and
2. 2.2.2 Insert all insertable control rods.

The approved analytical methods are described in the following documents, the approved version(s) of which are specified in the COLR.

1. XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company.
2. XN-NF-85-67(P)(A), "Generic Mechanical Design for Exxon Nuclear Jet pump BWR Reload Fuel," Exxon Nuclear Company.
3. EMF-85-74(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," Siemens Power Corporation.
4. ANF-89-98(P)(A), "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation.
5. XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors," Exxon Nuclear Company.
6. EMF-2158(P)(A), "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2," Siemens Power Corporation.
7. EMF-2361(P)(A), "EXEM BWR-2000 ECCS Evaluation Model," Framatome ANP.
8. EMF-2292(P)(A), "ATRIUMTM-10: Appendix K Spray Heat Transfer Coefficients," Siemens Power Corporation
9. Not used
10. Notused
11. Notused
12. ANF-1358(P)(A),"TheLossofFeedwaterHeatingTransientinBoiling Water Reactors," Advanced Nuclear Fuels Corporation.
13. EMF-2209(P)(A),"SPCBCriticalPowerCorrelation,"SiemensPower Corporation.
14. EMF-CC-074(P)(A),"BWRStabilityAnalysis-AssessmentofSTAIF with Input from MICROBURN-B2," Siemens Power Corporation.

5.6 Reporting Requirements 5.6.5 COLR (continued)

15. Notused
16. NEDO-32465-A,“BWROGReactorCoreStabilityDetectandSuppress

    Solutions Licensing Basis Methodology for Reload Applications.

17. BAW-10247PA,“RealisticThermal-MechanicalFuelRodMethodology for Boiling Water Reactors.”
18. ANP-10340P-A,“IncorporationofChromia-DopedFuelPropertiesin AREVA Approved Methods.”
19. ANP-10335P-A,“ACE/ATRIUM-11CriticalPowerCorrelation.”
20. ANP-10300P-A,“AURORA-B:AnEvaluationModelforBoilingWater

    Reactors; Application to Transient and Accident Scenarios.”

21. ANP-10332P-A,“AURORA-B:AnEvaluationModelforBoilingWater Reactors; Application to Loss of Coolant Accident Scenarios.”
22. ANP-10333P-A,“AURORA-B:AnEvaluationModelforBoilingWater Reactors; Application to Control Rod Drop Accident (CRDA).”
23. ANP-10307PA,“AREVAMCPRSafetyLimitMethodologyforBoiling Water Reactors.”
24. BAW-10247P-ASupplement2P-A,“RealisticThermal-MechanicalFuel Rod Methodology for Boiling Water Reactors, Supplement 2: Mechanical Methods.”

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

Reporting Requirements 5.6

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SUSQUEHANNA NUCLEAR, LLC
ALLEGHENY ELECTRIC COOPERATIVE, INC. DOCKET NO. 50-388
SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2 AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 260 Renewed License No. NPF-22

1. The U.S. Nuclear Regulatory Commission (NRC or the Commission) has found that:

1. The application for amendment filed by Susquehanna Nuclear, LLC, dated July 15, 2019, as supplemented by letters dated February 6, 2020, and April 1, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 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, 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. NPF-22 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 260, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. Susquehanna Nuclear, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to loading ATRIUM 11 fuel into the core during the spring 2021 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation

ATTACHMENT TO LICENSE AMENDMENT NO. 260 SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2 RENEWED FACILITY OPERATING LICENSE NO. NPF-22 DOCKET NO. 50-388

Replace the following pages of the Renewed Facility Operating License with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE Page 3 Page 14

INSERT Page 3 Page 14

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE INSERT 2.0-1 2.0-1

22. 5.0-22 5.0-22
23. 5.0-23 5.0-23

(3) Susquehanna Nuclear, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed neutron sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Susquehanna Nuclear, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Susquehanna Nuclear, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

1. (1) Maximum Power Level

Susquehanna Nuclear, LLC is authorized to operate the facility at reactor core power levels not in excess of 3952 megawatts thermal in accordance with the conditions specified herein. The preoperational tests, startup tests and other items identified in License Conditions 2.C.(20), 2.C.(21), 2.C.(22), and 2.C.(23) to this license shall be completed as specified.

2. (2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 260, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. Susquehanna Nuclear, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

For Surveillance Requirements (SRs) that are new in Amendment 151 to Facility Operating License No. NPF-22, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 151. For SRs that existed prior to Amendment 151, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 151.

22. (22) Neutronic Methods
    1. (a) Not Used
    2. (b) Not Used
23. (23) Containment Operability for EPU

The operating licensee shall ensure that the CPPU containment analysis is consistent with the SSES 1 and 2 operating and emergency procedures. Prior to operation above CLTP, for each respective unit, the operating licensee shall notify the NRC project manager that all appropriate actions have been completed.

24. (24) Primary Containment Leakage Rate Testing Program

Those primary containment local leak rate program tests (Type B – leakage boundary and Type C - containment isolation valves) as modified by approved exemptions, required by 10 CFR Part 50, Appendix J, Option B and Technical Specification 5.5.12, are not required to be performed at the CPPU peak calculated containment internal pressure of 48.6 psig (Amendment No. 224 to this Operating License) until their next required performance.

25. (25) Critical Power Correlation Additive Constants

AREVA NP has submitted EMF-2209(P), Revision 2, Addendum 1 (ML081260442) for NRC review to correct the critical power correlation additive constants due to a prior Part 21 notification (ML072830334). The report is currently under NRC review.

The license shall apply additional margin to the cycle specific OLMCPR, consistent in magnitude with the non-conservatism reported in the Part 21 report, thus imposing the appropriate MCPR penalty on the OLMCPR. This compensatory measure is to be applied until the approved version of

Renewed Operating License No. NPF-22

Amendment No. 260

1. 2.0 SAFETY LIMITS (SLs)
2. 2.1 SLs

2.1.1 Reactor Core SLs

≥ 10 million lbm/hr:
MCPR shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.11

for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL
Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations
With any SL violation, the following actions shall be completed within 2 hours:

1. 2.2.1 Restore compliance with all SLs; and
2. 2.2.2 Insert all insertable control rods.

PPL Rev. 5 Safety Limits (SLs)

2.0

1. 2.1.1.1 With the reactor steam dome pressure < 575 psig or core flow < 10 million lbm/hr:

   THERMAL POWER shall be ≤ 23% RTP.

2. 2.1.1.2 With the reactor steam dome pressure ≥ 575 psig and core flow

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The approved analytical methods are described in the following documents, the approved version(s) of which are specified in the COLR.

1. XN-NF-81-58(P)(A), “RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model,” Exxon Nuclear Company.
2. XN-NF-85-67(P)(A), “Generic Mechanical Design for Exxon Nuclear Jet pump BWR Reload Fuel,” Exxon Nuclear Company.
3. EMF-85-74(P)(A), “RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model,” Siemens Power Corporation.
4. ANF-89-98(P)(A), “Generic Mechanical Design Criteria for BWR Fuel Designs,” Advanced Nuclear Fuels Corporation.
5. XN-NF-80-19(P)(A), “Exxon Nuclear Methodology for Boiling Water Reactors,” Exxon Nuclear Company.
6. EMF-2158(P)(A), “Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN- B2,” Siemens Power Corporation.
7. EMF-2361(P)(A), “EXEM BWR-2000 ECCS Evaluation Model,” Framatome ANP.
8. EMF-2292(P)(A), “ATRIUMTM-10: Appendix K Spray Heat Transfer Coefficients,” Siemens Power Corporation.
9. Not used
10. Not used
11. Not used
12. ANF-1358(P)(A), “The Loss of Feedwater Heating Transient in Boiling Water Reactors,” Advanced Nuclear Fuels Corporation.
13. EMF-2209(P)(A), “SPCB Critical Power Correlation,” Siemens Power Corporation.
14. EMF-CC-074(P)(A), “BWR Stability Analysis - Assessment of STAIF with Input from MICROBURN-B2,” Siemens Power Corporation.

15. Not used

16. NEDO-32465-A, “BWROG Reactor Core Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications.”
17. BAW-10247PA, “Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors.”
18. ANP-10340P-A, “Incorporation of Chromia-Doped Fuel Properties in AREVA Approved Methods.”

19. ANP-10335P-A, “ACE/ATRIUM-11 Critical Power Correlation.”
20. ANP-10300P-A, “AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios.”
21. ANP-10332P-A, “AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios.”
22. ANP-10333P-A, “AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA).”
23. ANP-10307PA, “AREVA MCPR Safety Limit Methodology for Boiling Water Reactors.”
24. BAW-10247P-A Supplement 2P-A, “Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, Supplement 2: Mechanical Methods.”

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

ENCLOSURE 3

NON-PROPRIETARY SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 278 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-14 AND AMENDMENT NO. 260 TO
RENEWED FACILITY OPERATING LICENSE NO. NPF-22 SUSQUEHANNA NUCLEAR, LLC SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2
DOCKET NOS. 50-387 AND 50-388

Proprietary information pursuant to Section 2.390 of Title 10 of the *Code of Federal Regulations* has been redacted from this document.

Redacted information is identified by blank space enclosed within [[double brackets]]

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1.0

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 278 TO
RENEWED FACILITY OPERATING LICENSE NO. NPF-14
AND AMENDMENT NO. 260 TO
RENEWED FACILITY OPERATING LICENSE NO. NPF-22 SUSQUEHANNA NUCLEAR, LLC
SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 DOCKET NOS. 50-387 AND 50-388

INTRODUCTION

By letter dated July 15, 2019 [1], as supplemented by letters dated February 6, 2020 [2], and April 1, 2020 [3], Susquehanna Nuclear, LLC (the licensee) submitted a license amendment request (LAR) for Susquehanna Steam Electric Station (Susquehanna), Units 1 and 2, to allow application of the Framatome analysis methodologies necessary to support a planned transition to ATRIUM 11 fuel under the currently licensed Maximum Extended Load Line Limit Analysis (MELLLA) operating domain.

The supplemental letters dated February 6, 2020, and April 1, 2020, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on October 22, 2019 (84 FR 56482).

The proprietary information in this document is marked with double brackets and bold font such as **[[ ]]**.

2.0 REGULATORY EVALUATION

The NRC staff reviewed the LAR to evaluate the applicability of the Framatome methodologies to Susquehanna to confirm that the use of the methodologies is within the NRC-approved ranges necessary to support a planned transition to ATRIUM 11 fuel and to verify that the results of the analyses and methodologies are in compliance with the requirements of the following general design criteria (GDC) specified in Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50. In addition, the NRC staff assessed the aging degradation due to irradiation embrittlement in reactor pressure vessel (RPV) base metal and welds to verify compliance with the requirements of the following regulations. Each subsection of this safety evaluation (SE) includes a Regulatory Evaluation section specific to that portion of the review.

GDC 4, “Environmental and dynamic effects design bases*,”* requiring that structures, systems, and components important to safety be designed to accommodate the effects

of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant

accidents.

- • GDC 10, “Reactor design,” requiring that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs).
- • GDC 12, “Suppression of reactor power oscillations,” requiring that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations that can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.
- • GDC 13, “Instrumentation and control,” requiring that instrumentation be provided to monitor variables and systems over their anticipated ranges to assure adequate safety and that appropriate controls be provided to maintain these variables and systems within prescribed operating ranges.
- • GDC 15, “Reactor coolant system design,” requiring that the reactor coolant system (RCS) and associated auxiliary*,* control, and protection systems be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded during any condition of normal operation, including AOOs.

  •

- • GDC 25, “Protection system requirements for reactivity control malfunctions,” requiring that the protection system be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.
- • GDC 26, “Reactivity control system redundancy and capability,” requiring that two independent reactivity control systems of different design principles be provided, one of which can hold the reactor core subcritical under cold conditions.
- • GDC 27, “Combined reactivity control system capability,” requiring that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system (ECCS), of reliably controlling reactivity changes under postulated accident conditions.

GDC 4, “Environmental and dynamic effects design bases,” requiring that structures, systems, and components important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.

GDC 20, “Protection system functions,” requiring that the protection system be designed (1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of AOOs and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

3.0

GDC 28, “Reactivity limits,” requiring that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the RCPB greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other RPV internals to impair significantly the capability to cool the core.

GDC 35, “Emergency core cooling,” requiring that a system to provide abundant emergency core cooling is provided to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

10 CFR Part 50, Appendix G, “Fracture Toughness Requirements,” provides fracture toughness requirements for ferritic materials in the RCPB, including requirements for the Charpy upper-shelf energy (USE) for protecting RPV beltline materials against non-brittle failure and requirements for calculating RCS pressure-temperature (P-T) limits for protection against brittle fracture. Regulatory Guide (RG) 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” contains methodologies for determining the increase in transition temperature and the decrease in USE resulting from neutron radiation.

10 CFR 50.55a imposes the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code (BPV Code), Section XI, for Class 1, 2, and 3 pressure-retaining components and their integral attachments in light-water cooled nuclear power plants. The ASME BPV Code Section XI code of record for the fourth ISI interval at Susquehanna is the ASME BPV Code, Section XI, 2007 Edition through 2008 Addenda.

10 CFR 50.36(c) specifies the categories that are to be included in the TSs, including (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. In 10 CFR 50.36(c)(5), administrative controls are stated to be "the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure the operation of the facility in a safe manner." This also includes the programs established by the licensee and listed in the administrative controls section of the TS for the licensee to operate the facility in a safe manner.

TECHNICAL EVALUATION

In the LAR, the licensee requested a revision to Susquehanna, Units 1 and 2, TS 5.6.5.b to allow application of Advanced Framatome Methodologies for determining core operating limits in support of loading Framatome fuel type ATRIUM 11. The revision would support the transition to ATRIUM 11 fuel in the approved operating domain at Susquehanna, which includes MELLLA conditions. The LAR also requested revisions to TSs 2.1.1.1 and 2.1.1.2 to revise the low-pressure safety limit and remove neutronic methods penalties on oscillation power range monitor (OPRM) amplitude setpoint, the pin power distribution uncertainty, and bundle power correlation coefficient.

A request to implement Technical Specifications Task Force (TSTF) Traveler TSTF-535, "Revise Shutdown Margin Definition to Address Advanced Fuel Designs," was also included in this LAR. This change was reviewed and approved in Amendment Nos. 274 and 256 [4].

This SE includes a detailed review of the following areas of the LAR:

- • applicability of Framatome boiling-water reactor (BWR) methods to Susquehanna with ATRIUM 11 fuel
- • mechanical design of ATRIUM 11 fuel assemblies
- • thermal-hydraulic design of ATRIUM 11 fuel assemblies
- • ATRIUM 11 fuel rod thermal-hydraulic evaluation
- • ATRIUM 11 transient demonstration
- • loss-of-coolant accident (LOCA) analysis for ATRIUM 11 fuel
- • Susquehanna ATRIUM 11 control rod drop accident (CRDA) analyses
- • revision of low-pressure safety limit in TSs 2.1.1.1 and 2.1.1.2
- • removal of neutronic methods penalties for OPRM amplitude setpoint and pin power

  distribution uncertainty and bundle power correlation coefficient

- • aging degradation

  The NRC staff reviewed the LAR in conjunction with the supplemental information and the responses to the NRC staff's requests for additional information (RAIs) [2], [3] to (1) evaluate the acceptability of the Susquehanna transition to Framatome ATRIUM 11 fuel, (2) evaluate the use of the associated

Framatome methodologies for licensing applications, and (3) confirm the adequate technical basis for the proposed TS changes.

3.1 Applicability of Framatome BWR Methods to Susquehanna with ATRIUM 11 Fuel

Applicability of Framatome BWR methods is addressed in the BWR compendium [5], which is referenced as part of ANP-3753P (Enclosure 8 to [1]). While the NRC staff did not review and approve this reference, the staff reviewed it for applicability to the use of ATRIUM 11 fuel at Susquehanna. Many of the methodologies discussed in the compendium have previously been confirmed to be applicable to ATRIUM 10 fuel at Susquehanna and apply to the use of ATRIUM 11 fuel because it is fundamentally an evolutionary fuel design with similar geometry and composition characteristics. When appropriate, the applicability of methodologies to specific safety analyses is addressed in the discussion later in this SE associated with that analysis. Three areas of interest are as follows:

1. ANP-3753P Section 5.4 is dedicated to safety limit minimum critical power ratio (MCPR), specifically related to the methodology to determine the TS limit to ensure that 99.9 percent of fuel rods avoid boiling transition during normal reactor operation and AOOs. The NRC staff's evaluation of this section is provided with the remaining safety limit MCPR evaluation in Section 3.5.2.1 (MCPR Fuel Cladding Integrity Safety Limit) of this SE.

2 ANP-3753P Section 6.4 is dedicated to CRDA, specifically related to the critical heat flux (CHF) correlation used for the CRDA calculations. The NRC staff's evaluation of this

3.2

3.

section is provided with the remaining CRDA evaluation in Section 3.8 (Control Rod Drop Accident (CRDA) of this SE.

ANP-3753P Section 7.0 is dedicated to stability, specifically related to how Susquehanna updated its Option III stability methods to the capture chromia-doped fuel properties in the ATRIUM 11 fuel design. The NRC staff's evaluation of this section is provided in Section 3.4 (Stability) of this SE.

ATRIUM 11 Fuel Assembly/Rod Design

Regulatory Basis

3.2.1

The ATRIUM 11 fuel (assembly/rod) design was developed using the thermal mechanical design bases and limits outlined in ANF-89-98(P)(A) [6], compliance with which ensures that the fuel design meets the fuel system damage, fuel failure, and fuel coolability criteria identified in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP) [7]. The SRP is intended to provide comprehensive guidance for NRC staff review of whether LARs satisfy regulatory requirements, including the evaluation of the safety of light-water nuclear power plants and the review of safety analysis reports.

SRP Section 4.2, “Fuel System Design”; Section 4.3, “Nuclear Design”; and Section 4.4, “Thermal and Hydraulic Design,” provide regulatory guidance for the review of fuel rod cladding materials, the fuel system, the design of the fuel assemblies and control systems, and the thermal and hydraulic design of the core. In addition, the SRP provides guidance for compliance with the applicable GDC in Appendix A to 10 CFR Part 50. In accordance with SRP Section 4.2, the fuel system safety review provides assurance that:

- • the fuel system is not damaged as a result of normal operation and AOOs;
- • fuel system damage is never so severe as to prevent control rod insertion when it

  is required;

- • the number of fuel rod failures is not underestimated for postulated accidents; and
- • coolability is always maintained.

  The NRC staff reviewed the LAR to evaluate the applicability of Framatome BWR methodology to the use of ATRIUM 11 fuel at Susquehanna to confirm that the use of the methodology is within the NRC-approved ranges of its applicability and to verify that the results of the analyses comply with the requirements of the following GDC in Appendix A to 10 CFR Part 50:

- • GDC 10, “Reactor design,” requiring that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs).
- • GDC 27, “Combined reactivity control systems capability,” requiring that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions.

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• GDC 35, “Emergency core cooling,” requiring that a system to provide abundant emergency core cooling is provided to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented, and (2) clad metal-water reaction is limited to negligible amounts.

Technical Evaluation

ANP-3762P (Enclosure 9a to the LAR [1]) provides the mechanical design details, fuel structural analysis results of the ATRIUM 11 fuel assemblies, and fuel channel designs, while ANP-3745P (Enclosure 11a to the LAR [1]) provides the design parameters and design evaluation results of the ATRIUM 11 fuel rods to be used at Susquehanna.

3.2.2.1 Summary of Mechanical Design of ATRIUM 11 Fuel Assemblies for Susquehanna ANP-3762P (Enclosure 9a to the LAR) provides key fuel assembly design details for the

Framatome ATRIUM 11 fuel assembly design planned for use at Susquehanna. **[[**

]] Table 2-1 of ANP-3762P lists the fuel assembly and component description of the ATRIUM 11 fuel assembly design. Further descriptions of the fuel assembly

components are provided in ANP-3762P.

3.2.2.2 Applicability of Methodologies for Analysis of ATRIUM 11 Fuel Assembly Mechanical Design

To perform specific evaluations for the ATRIUM 11 fuel assembly mechanical design, the licensee utilized specific NRC-approved methodologies. NRC approval of these methodologies is conditional on adequately meeting the limitations and conditions listed in the NRC staff's SE for each of the topical reports. A discussion of how these limitations and conditions are met for Susquehanna is provided below for each of the topical reports directly supporting the
ATRIUM 11 fuel assembly mechanical design evaluations, as well as a discussion of the applicability of topical reports already in use at Susquehanna for analysis of the ATRIUM 10 fuel assembly design that may not automatically apply to the ATRIUM 11 fuel assembly design.

- **ANF-89-98(P)(A), "Generic Mechanical Design Criteria for BWR Fuel Designs," Revision 1, and Supplement 1, dated May 1995.**

ANF-89-98(P)(A) provides some generic mechanical design criteria that were approved by the NRC for use with evaluation of Framatome fuel designs. The ATRIUM 11 fuel mechanical design as reported in ANP-3762P, as discussed in Section 3.2.2.3 (Fuel Assembly Mechanical Design Evaluation) of this SE, describes how the design criteria presented in ANF-89-98(P)(A) apply to the ATRIUM 11 fuel assembly mechanical design.

3.2.2

- **EMF-93-177P-A, "Mechanical Design for BWR Fuel Channels," Revision 1, dated August 2005, and Supplement 1P-A, "Mechanical Design for BWR Fuel Channels Supplement 1: Advanced Methods for New Channel Designs," Revision 0, dated September 2013** [8]

The NRC staff's SE for EMF-93-177-NP-A specified several limitations and conditions that have already been shown to be met at Susquehanna for the channels associated with the ATRIUM 10 fuel. Since the ATRIUM 11 channels are very similar, the disposition of the limitations and conditions remains applicable. The two exceptions are the use of

Z4B channels, as approved in Supplement 2P-A [9] and interior milling, which is addressed through the use of the Supplement 1P-A methodology. The Supplement 1P-A methodology was approved with no limitations or conditions.

- • **BAW-10247P-A, Supplement 2P-A, “Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, Supplement 2: Mechanical Methods,” Revision 0, dated August 2018** [10]

  The ATRIUM 11 fuel mechanical design evaluation, as discussed in Section 3.2.2.3 (Fuel Assembly Mechanical Design Evaluation) of this SE, confirms that the **[[**

  ]] and that **[[**

  ]]. The remaining limitations and conditions are met for the ATRIUM 11 fuel assembly design, since the

  channels are constructed of either Zircaloy-4 or Z4B, and the fuel rod materials fall within the range of applicability for the database used to support the fuel rod growth correlations.

3.2.2.3 Fuel Assembly Mechanical Design Evaluation

The objectives of the fuel design are that (i) the fuel assembly (system) is not damaged as a result of normal operation and AOOs, (ii) fuel system damage is never so severe as to prevent control rod insertion when it is required, (iii) the number of fuel rod failures is not underestimated for postulated accidents, (iv) fuel coolability is always maintained [9], (v) the mechanical design of the fuel assemblies shall be compatible with co-resident fuel and the reactor core internals, and (vi) fuel assemblies shall be designed to withstand the loads from handling and shipping. The first four objectives are from SRP Section 4.2 and the latter two are to assure the structural integrity of the fuel and the compatibility with the existing reload fuel (co-resident fuel). This fuel assembly mechanical design evaluation contains only fuel assembly structural analyses, while the fuel rod evaluation, as documented in Enclosure 11a to the LAR [1] is discussed in Section 3.2.2.6 (ATRIUM 11 Fuel Rod Design Evaluation) of this SE.

Stress, Strain, Loading, and Deformation Limits on Assembly Components

The licensee used the *American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code* (BPV)) [11] as a guide to establish the acceptable stress, deformation, and load limits for standard assembly components. These limits are applied to the design and evaluation of the upper tie plate (UTP), lower tie plate (LTP), spacer grids, springs, and load chain components, as necessary and applicable. The fuel assembly structural component criteria under faulted conditions are based on Appendix F of the ASME BPV Code, Section III, with some criteria derived from component tests. Outside of faulted conditions, most structural components are under the most limiting loading conditions during fuel handling. In summary, analyses were performed to determine the mechanical performance of assembly components

during accidents (e.g., seismic events or LOCA events), fuel handling events, or during normal and AOO conditions.

For accident conditions, the dynamic characteristics of the fuel assembly and grids were obtained from testing the assemblies for stiffness, natural frequencies, and damping values, and used as inputs to analytical models for the fuel assembly and fuel channel. These tests were conducted with and without a fuel channel. The test results, when compared with analysis results, have shown the dynamic response of the ATRIUM 11 fuel assembly design to be like other BWR fuel designs that have the same basic channel configuration and weight. The licensee's evaluations of fuel under accident loadings include mechanical fracturing of the fuel rod cladding, assembly structural integrity, and fuel assembly liftoff.

For the fuel handling accident, the primary design criteria given in ANF-89-98(P)(A) is that the fuel assembly and load chain components must be able to withstand an axial tensile force of at least **[[**

]]

For fuel structural characteristics for normal and AOO conditions, the licensee performed evaluations on the stress for ATRIUM 11 fuel channels due to pressure differential and found that the pressure load, including AOO, meets the ASME BPV Code criteria of **[[**

]]. The stress as a result of vertical acceleration is found to be less than allowable. Hence, the deformation during AOO remains within functional limits for normal control blade operation.

Based on the above, the NRC staff finds the licensee's evaluation acceptable because the evaluation is complete and adequate to meet the required design criteria and satisfy the SRP objectives.

Fatigue and Fretting Wear

Fatigue of structural components is generally low because of a small number of cycles (reactor startup) or small amplitudes. The fatigue loads on the fuel channels remain under the fatigue life curve determined by O'Donnell and Langer per Section 2.3 of ANF-89-98(P)(A). While some of the fuel channels will be constructed with Z4B rather than conventional zirconium alloys, **[[ ]]** Therefore, the fatigue life curves remain applicable.

Although there is no specific wear limit for fretting, a general acceptance criterion is that fuel rod failures due to grid-to-rod fretting shall not occur. **[[**

]]. Post-test inspections of the fuel assembly showed no significant wear on fuel rods. Although the testing period is short relative to the time that a fuel assembly will typically spend in the reactor core, this result is sufficient to provide reasonable assurance that structural flaws in the fuel rod cladding would not be expected to lead to widespread fuel rod failures.

The NRC staff finds that based on the fatigue loads, the fuel channels will continue to perform their function and will not interfere with control blade insertion. Furthermore, the NRC staff finds that based on the results of the fretting wear testing, widespread rod failures would not be expected because of fretting effects. The NRC staff notes that isolated rod failures due to localized mechanisms leading to excessive fretting are not explicitly required by regulatory acceptance criteria to be addressed; therefore, the generic testing performed in support of this conclusion was sufficient to establish a regulatory finding.

Rod Bow

A combination of differential expansion between the fuel rods and cage structure, thermal gradients, and flux gradients can result in lateral loads applied to the fuel rods. This load may result in rod bowing in the spans between spacer grids due to creep. Since a reduction in rod pitch may have a detrimental impact on power peaking and local heat transfer, the licensee must address the potential impact on thermal margins. The Framatome design criterion for fuel rod bowing is **[[**

]] The licensee developed a

[[

]] described in BAW-10247P-A, Supplement 2P-A [10].

The NRC has approved the use of the BAW-10247P-A, Supplement 2P-A correlation for all current and future Framatome BWR fuel designs up to an **[[**

]], provided that the change process described in [10], Section 5.0, “Change Process,” is followed.

Axial Irradiation Growth

Rod growth, assembly growth, and fuel channel growth are calculated using correlations that were reviewed and approved by the NRC in BAW-10247P-A, Supplement 2P-A. In accordance with BAW-10247P-A, Supplement 2P-A, **[[**

]] The channel material that will be used in Susquehanna Z4B is within the scope of the NRC approval

of BAW-10247P-A, Supplement 2P-A. Furthermore, the NRC considered and accepted data for the ATRIUM 11 fuel assembly design as part of the basis and applicability for the BAW-10247P-A, Supplement 2P-A methodology.

The NRC staff finds the approach used to address axial irradiation growth to be acceptable based on the use of an NRC-approved methodology within the bounds of applicability of the approval and consistent with the limitations and conditions as discussed above.

Assembly Liftoff

The design criteria for assembly liftoff are no liftoff from fuel support during normal operations (including AOOs) and no disengagement from fuel support during postulated accidents. These

criteria assure control blade insertion is not impaired. For normal operating conditions, the calculated net axial force acting on the assembly due to the addition of the loads from gravity, hydraulic resistance from coolant flow, difference in fluid flow entrance and exit momentum, and buoyancy will be in the downward direction, indicating no assembly liftoff. The licensee confirmed that the calculated net force will be in the downward direction, indicating no assembly liftoff. **[[**

]]

Mixed core conditions for assembly liftoff are considered on a cycle-specific basis as determined by the plant operating conditions and other fuel types. Analyses to date indicate a large margin to assembly liftoff under normal operating conditions.

For faulted (postulated accident) conditions, **[[**

]]. The fuel will not lift under normal or AOO conditions. It will not become disengaged from the fuel support under faulted conditions or block the insertion of the control blade in all operating conditions.

Based on the above, the NRC staff finds the liftoff evaluation acceptable because the evaluation is complete and adequate to meet the required design criteria and satisfy the SRP objectives.

Fuel Channel Irradiation-Induced Changes

The fuel channel was specifically evaluated for changes due to exposure to the reactor environment that may lead to loss of strength or deformation. These types of changes are critical for the fuel channel because the fuel channel typically absorbs most of the load from seismic events and other similar design-basis events and is also the component most likely to interfere with control blade insertion. The proposed fuel channels are constructed of Z4B, which was approved by the NRC as part of EMF-93-177P-A, Revision 1, Supplement 1P-A,

Revision 0 . **[[**

]]. The NRC staff finds this disposition of the potential changes to the fuel channel as a result of irradiation and exposure to the coolant to be acceptable because the use of Z4B material with the EMF-93-177 methodology was reviewed by the NRC in Supplement 2P-A. **[[**

]]

Summary of Sections 3.2.2.1 through 3.2.2.3

Tables 3-1 and 3-2 of Enclosure 9a to the LAR provide a disposition of the specific design criteria evaluated for the ATRIUM 11 fuel assembly design based on the aforementioned tests

and analyses. The NRC staff considerations of the approach used to perform the dispositions are summarized above. As a result, the NRC staff finds that evaluations have been performed acceptably to ensure that the mechanical design criteria for the ATRIUM 11 fuel assembly design are met for use in the Susquehanna reactor cores.

3.2.2.4 Summary of ATRIUM 11 Fuel Rod Thermal-Mechanical Design for Susquehanna

ANP-3745P (Enclosure 11a to the LAR) provides key fuel rod design details for Framatome ATRIUM 11 fuel planned for use at Susquehanna. The ATRIUM 11 fuel rod is conventional in design configuration and is very similar to past designs such as the ATRIUM 10XM and ATRIUM 10 fuel rods. **[[**

]] plenum spring on the upper end of the fuel column assists in maintaining a compact fuel column during shipment and initial reactor operation.

There are two part length fuel rod (PLFR) designs incorporated in the fuel assembly. **[[**

]]. Table 3-1 of ANP-3745P lists the key fuel rod design parameters for the ATRIUM 11 fuel.

Further descriptions of the fuel assembly components are provided in ANP-3745P.
3.2.2.5 Applicability of Methodologies for Analysis of ATRIUM 11 Fuel Rod Design

To perform specific evaluations for the ATRIUM 11 fuel rod design, the licensee utilized specific NRC-approved methodologies. NRC approval of these methodologies is conditional on adequately meeting the limitations and conditions listed in the NRC staff's SE for each of the topical reports. A discussion of how these limitations and conditions are met for Susquehanna is provided below for each of the topical reports directly supporting the ATRIUM 11 fuel rod design evaluations, as well as a discussion of the applicability of topical reports already in use at Susquehanna for analysis of the ATRIUM 10 fuel rod design that may not automatically apply to the ATRIUM 11 fuel rod design.

ANF-89-98(P)(A), "Generic Mechanical Design Criteria for BWR Fuel Designs," Revision 1, and Supplement 1, dated May 1995.

ANF-89-98(P)(A) provides some generic fuel rod design criteria that were approved by the NRC for use with evaluation of Framatome fuel designs. The ATRIUM 11 fuel rod design as reported in ANP-3745P, as discussed in Section 3.2.2.6 (ATRIUM 11 Fuel Rod Design Evaluation) of this SE) , describes how the design criteria presented in ANF-89-98(P)(A) apply to the ATRIUM 11 fuel rod design.

BAW-10247PA, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," Revision 0, dated February 2008 [13]

Section 3.2.2.6 (ATRIUM 11 Fuel Rod Design Evaluation) of this SE includes a discussion under the "Oxidation, Hydriding, and Crud Buildup" subsection that describes how the crud effects are addressed. ANP-10340P-A [14] contains a similar limitation and condition on the

[[   ]], which is addressed through an automated software check. The remaining limitations and conditions are addressed by only utilizing the methodology within the bounds defined by the limitations and conditions.

ANP-10340P-A, "Incorporation of Chromia-Doped Fuel Properties in AREVA-Approved Methods," Revision 0, dated May 2018 [14]

The chromia-doped fuel properties and models described in this topical report are directly applicable to the ATRIUM 11 fuel pellets. The limitations and conditions are met through a combination of automated software checks and administrative controls, as described in
Section 2-18 of the BWR compendium. The automated software checks are managed through the Framatome software quality assurance program, which is subject to normal NRC oversight activities as part of verifying compliance with Appendix B to 10 CFR Part 50. The NRC staff notes that the methodologies that will be used to evaluate the ATRIUM 11 fuel at Susquehanna are approved for maximum fuel rod burnups of up to 62 gigawatt days per metric ton of uranium (GWd/MTU).

3.2.2.6 ATRIUM 11 Fuel Rod Design Evaluation

The NRC staff's review of fuel rod thermal-mechanical analyses for the ATRIUM 11 fuel was performed using acceptance criteria from ANP-89-98(P)(A), Revision 1, and Supplement 1 and the RODEX4 analysis methodology described in BAW-10247PA [10] and [13]. The methodology described in ANP-10340P-A was used to address the impact of the chromia additive in the fuel pellets for ATRIUM 11 fuel assemblies. The RODEX4 fuel rod analysis code and methodology are used to analyze the fuel rod for fuel centerline temperature, cladding strain, rod internal pressure, cladding collapse, cladding fatigue, and external oxidation.

Fuel Rod Design Evaluation

The ATRIUM 11 fuel assembly design contains multiple changes in geometry to accommodate the change from a 10x10 rod array to an 11x11 rod array within the same basic channel dimensions. The part length rod specifications also differ from the ATRIUM 10 design. The ATRIUM 11 fuel also utilizes two relatively new materials in its overall composition—the chromia additive in the fuel pellets and the Z4B alloy used for some of the structural elements.

Additional details regarding the fuel rod design are provided in Section 3.1 of ANP-3745P (Enclosure 11a to the LAR). The fuel rod geometry and compositions generally fit within the applicability of the NRC-approved RODEX4 thermal-mechanical analysis methodology, with the addition of the chromia-doped fuel properties and models reviewed and approved by the NRC [14]. Therefore, the RODEX4 code was used to evaluate the fuel rod thermal-mechanical performance of the ATRIUM 11 fuel rod design, as appropriate.

Table 2-1 of ANP-3745P provides a summary of the findings from the fuel rod design evaluations that demonstrates that the acceptance criteria are met. The key fuel rod design parameters used in the fuel rod design evaluations are provided in Table 3-1. Table 3-2 provides the specific results based on the equilibrium cycle for MELLLA conditions. The fuel rod analyses, such as those for fuel centerline temperature and cladding strain, cover normal operating conditions and AOOs. More detail on the NRC staff considerations in reviewing each acceptance criterion is provided below.

Internal Hydriding

The absorption of hydrogen by the cladding can result in cladding failure due to reduced ductility and the formation of hydride platelets. As stated in Section 3.3.1 of ANP-3745P, a fabrication limit is imposed **[[ ]]** and enforced via moisture controls. The NRC staff finds this to be an acceptable approach to ensure that the potential sources for hydrogen absorption inside the cladding are minimized, since the fabrication limit is based on NRC-approved mechanical design criteria.

Cladding Collapse

Fuel pellets undergo a densification process during irradiation, which can result in pellet shrinkage and generate axial gaps along the fuel column. The coolant system pressure causes the cladding to slowly creep inward and close the radial gap between the fuel pellet and the cladding. Since large axial gaps may cause the cladding to collapse into the space between fuel pellets and fail, Framatome imposes an upper limit on the size of the axial gaps. RODEX4 is used to predict the size of the gaps that may form. Since RODEX4 is a best estimate code, a statistical method is applied to confirm that the maximum size of the axial gaps due to densification is not exceeded for **[[**

]] This approach is consistent with the use of the RODEX4 code and the acceptance criterion in the NRC-approved fuel rod evaluation methodology and, therefore, is acceptable.

Overheating of Fuel Pellets

One of the limitations on the use of the RODEX4 methodology is that it may not be used to model fuel above incipient fuel melting temperatures. In practice, this is avoided by ensuring that the fuel centerline temperatures remain below melting. As necessary, the licensee adjusted the melting point to account for **[[**

]]. RODEX4 is used to determine the fuel centerline temperature for normal operating conditions and AOOs to establish an upper limit on the linear heat generation rate

(LHGR) that ensures that no centerline melting will occur. This approach is consistent with the use of the RODEX4 methodology and, therefore, is acceptable.

Stress and Strain Limits

Under transient conditions, the inner diameter of the cladding may shrink more rapidly than the outer diameter of the fuel pellet due to differences in their rates of change in temperature. If the cladding surface presses on the outside of the fuel pellet, this results in the pellet-clad interaction phenomenon. The pressure of the fuel pellet resisting the shrinkage of the cladding can cause local deformation of the cladding or cladding strain. The RODEX4 methodology is used to calculate the predicted cladding strain **[[**

]] to confirm that the strain is no more than one percent. This is consistent with the RODEX4 methodology and the one percent strain limit is consistent

with the NRC-approved fuel rod evaluation methodology and, therefore, is acceptable. Cladding stresses are calculated using solid mechanics elasticity solutions and finite element

methods. Stresses are calculated for the primary and secondary loadings.

[[

]]. The results were determined for both beginning of life and end of life conditions to bound the spectrum of

possible stresses and were then compared against the design limits prescribed by Section III of the ASME BPV Code [11]. This is consistent with NRC-approved mechanical design criteria and, therefore, is acceptable.

Fuel Densification and Swelling

There are no specific acceptance criteria for fuel densification and swelling; however, these phenomena may affect other acceptance criteria. Consequently, their effects are explicitly included in the RODEX4 methodology. The NRC has reviewed and approved the models used in RODEX4 to address these phenomena; therefore, this is an acceptable disposition.

Fatigue

The fuel rod cladding experiences cyclic thermal loads due to power changes during normal operating maneuvers. The thermal cycling translates to cyclic stress, which can lead to fuel rod cladding fatigue. The stresses are calculated using the RODEX4 methodology and **[[**

]]. This information can be used to determine fatigue usage factors for each axial region of the fuel rod, which represents the ratio of the number of accumulated cycles to the maximum allowed number of cycles for a given set of loadings. The cumulative usage factor is determined for each fuel rod by combining the fatigue usage factors. The axial region with the highest cumulative usage factor is used in the subsequent **[[**

]] The results are confirmed to remain below the maximum cumulative usage factor specified as an acceptance criterion.

Since the acceptance criterion is consistent with the NRC-approved fuel rod evaluation methodology and the evaluation is performed with a combination of an NRC-approved fuel rod analysis methodology and appropriately applicable data, the NRC staff finds this to be acceptable.

Oxidation, Hydriding, and Crud Buildup

The RODEX4 code and methodology are used to determine cladding external oxidation and its effect on the heat transfer coefficient from the cladding to the coolant. The acceptance criterion for oxidation is discussed within the NRC-approved RODEX4 fuel rod evaluation methodology, along with a discussion of how the impact of hydriding and crud buildup are to be addressed. The RODEX4 calculational methodology is calibrated to obtain an appropriate fit to measured oxide thickness data along with relevant uncertainties. The result is used to perform a

[[

]]. A brief discussion of the applicability of hydriding and crud buildup to Susquehanna is provided below.

• [[

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- • BAW-10247PA [13] discusses what constitutes “abnormal crud” and how to capture the effect using the crud heat transfer coefficient. Since the corrosion model takes into consideration the effect of the thermal resistance of the crud on the corrosion rate, this is already incorporated into the RODEX4 code. A similar approach would be used to address abnormal corrosion. However, no such observations have been made at Susquehanna for ATRIUM 10. The cladding properties for the ATRIUM 11 fuel assembly design are not different from the ATRIUM 10 fuel assembly design, so no change is expected as a result of transitioning to ATRIUM 11 fuel.
- **• [[**

  ]]

  The effects of oxidation, crud buildup, and hydriding are addressed through the use of the NRC-approved RODEX4 fuel rod evaluation methodology and its acceptance criteria, as appropriately applied to Susquehanna and the ATRIUM

11 fuel assembly design; therefore, the NRC staff finds the disposition as discussed above to be acceptable.

Rod Internal Pressure
The fuel rod internal pressure is calculated using the RODEX4 code and methodology. The

maximum rod pressure is limited to **[[**

]] under both steady-state and transient conditions, consistent with the acceptance criterion defined in ANF-89-98(P)(A). The NRC staff finds this

approach to be acceptable since it is based on a methodology and acceptance criteria that the NRC has previously reviewed and approved.

Water Chemistry

GDC 10 requires that specified acceptable fuel design limits are not exceeded during normal operation, including the effects of AOOs. Oxidation and hydriding are two specified acceptable fuel design limits that ensure components maintain strength and ductility. Section 3.5.1 of ANP-3762P mentions that water chemistry is controlled to reduce oxidation in the fuel channel.

The licensee stated in its February 6, 2020, letter that the plant water chemistry will be operated in accordance with the Electric Power Research Institute (EPRI) BWR Water Chemistry Guidelines (BWRVIP-190). The key figures of merit for water chemistry are those defined as “needed” or “control” parameters in Chapter 2 of BWRVIP-190, Volume 1. The measurement frequencies and operating limits for these parameters are defined in the guidelines, as is the response timeline for any excursions. Any deviations from the guidelines requirements for “needed” or “control” parameters must be justified by the licensee and documented in the plant’s strategic water chemistry plan. The NRC staff reviewed this response and found it acceptable because the industrial guideline is followed to ensure the satisfactory performance of

ATRIUM 11 fuel and Z4B water channel, which complies with the GDC 10 requirement to maintain fuel integrity.

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Summary of Sections 3.2.2.4 to 3.2.2.6

The NRC staff reviewed the licensee’s application of the RODEX4 code, analysis methodologies, and acceptance criteria, as approved in ANF-89-98(P)(A) and BAW-10247PA, in the fuel rod thermal-mechanical analyses for the ATRIUM 11 fuel design that is planned to be loaded and used for operation at Susquehanna. The NRC staff determined that the fuel design criteria, as supported by the applicable regulations and sections of NUREG-0800, have been satisfied and provide reasonable assurance of safe operation at Susquehanna.

3.2.3 Conclusion of ATRIUM 11 Fuel Assembly/Rod Design

For evaluation of the ATRIUM 11 fuel assembly/rod design (Section 3.2 of this SE, (ATRIUM 11 Fuel Assembly/Rod Design), the NRC staff concludes that the application of ATRIUM 11 fuel (fuel assembly and fuel rod) to Susquehanna is acceptable because it complies with the requirements of GDC 10, 27, and 35. This conclusion is based on the following:

1. The application meets the requirements of GDC 10 with respect to the specified acceptable fuel design limits not being exceeded during any condition of normal operation, including the effects of AOOs by:
    1. Developing and complying with fuel system damage criteria for all known damage mechanisms and operating conditions as evaluated in Sections 3.2.2.3 (Fuel Assembly Mechanical Design Evaluation) and 3.2.2.6 (ATRIUM 11 Fuel Rod Design Evaluation) and
    2. Applying NRC-approved fuel system design methodologies and adequately meeting the limitations and conditions listed in the NRC staff's SE for each of the applied topical reports as evaluated in Sections 3.2.2.2 (Applicability of Methodologies for Analysis of ATRIUM 11 Fuel Assembly Mechanical) Design and 3.2.2.5 (Applicability of Methodologies for Analysis of ATRIUM 11 Fuel Rod Design)
2. The application meets the requirements of GDC 27 with respect to the reactivity control system being designed with margin to have capability of reliably controlling reactivity changes by ensuring that fuel system damage is never so severe as to prevent control rod insertion when it is required. For example, as evaluated in Section 3.2.2.3 (Fuel Assembly Mechanical Design Evaluation) for Susquehanna) of this SE, the fatigue and fretting wear of the fuel assembly components was tested to ensure that it does not interfere with control blade insertion. As demonstrated by analysis, the fuel will not lift under normal or AOO conditions. It will not become disengaged from the fuel support under faulted conditions or block insertion of the control blade in all operating conditions. The fuel channel was specifically evaluated for changes due to exposure to the reactor environment that may lead to loss of strength or deformation to affect the control rod insertability.
3. The application meets the requirements of GDC 35 with respect to the fuel system being able to transfer heat from the reactor core following any loss of reactor coolant at an acceptable rate by ensuring that the fuel rod damage does not interfere with effective emergency core cooling and that the cladding temperatures do not reach a temperature high enough to allow a significant metal-water reaction to occur. These assurances are achieved by developing and complying with the fuel coolability-related criteria for all

sections below for

3.3.2 Technical Evaluation

severe fuel rod damage mechanisms as addressed in Section 3.2.2.6 (ATRIUM 11 Fuel Rod Design Evaluation) (e.g., internal hydriding, cladding collapse, overheating of fuel pellets, cladding stress and strain limits, fuel densification and swelling, and clad oxidation, hydriding, and crud buildup). The application applied NRC-approved RODEX4 fuel rod evaluation methodology and adequately met the limitations and conditions listed in the NRC staff's SE for each of the applied topical reports.

3.3 Thermal-Hydraulic Design of ATRIUM 11 Fuel Assemblies 3.3.1 Regulatory Basis

The ATRIUM 11 fuel design was developed using the thermal-mechanical design bases and limits as outlined in ANF-89-98(P)(A), compliance with which ensures that the fuel design meets the criteria for fuel system damage, fuel failure, and fuel coolability identified in Section 4.2 of the SRP. The SRP is intended to provide comprehensive guidance for NRC staff review of whether LARs satisfy regulatory requirements, including the evaluation of the safety of light-water nuclear power plants and review of safety analysis reports.

SRP Section 4.2, "Fuel System Design"; Section 4.3, "Nuclear Design"; and Section 4.4, "Thermal and Hydraulic Design," provide regulatory guidance for the review of fuel rod cladding materials, the fuel system, the design of the fuel assemblies and control systems, and the thermal and hydraulic design of the core. In addition, the SRP provides guidance for compliance with the applicable GDC in Appendix A to 10 CFR Part 50.

In accordance with SRP Section 4.2, the fuel system safety review provides assurance that:

- • the fuel system is not damaged as a result of normal operation and AOOs;
- • fuel system damage is never so severe as to prevent control rod insertion when it is

  required;

- • the number of fuel rod failures is not underestimated for postulated accidents; and
- • coolability is always maintained.

The NRC staff reviewed the LAR to evaluate the applicability of Framatome BWR methodology

to the use of ATRIUM 11 fuel at Susquehanna to confirm that the use of the methodology is

within NRC-approved ranges of its applicability and to verify that the results of the analyses

comply with the requirements of GDC 10, 12, 15, 20, 25, 26, 27, 28, and 35 (

further discussion).

This section describes the NRC staff's evaluation of the licensee's thermal-hydraulic analyses to demonstrate the hydraulic compatibility of ATRIUM 11 fuel with the co-resident ATRIUM 10 fuel at Susquehanna. The licensee is proposing to transition from the current ATRIUM 10 fuel design to ATRIUM 11 fuel. Enclosure 10a to the LAR [1] provides the results of the thermal-hydraulic analyses to support a finding that ATRIUM 11 fuel is hydraulically compatible with the co-resident ATRIUM 10 fuel. The results from the thermal-hydraulic analyses are compared to acceptance criteria established in NRC-approved topical reports ANF-89-98(P)(A), Revision 1, Supplement 1, and XN-NF-80-19(P)(A), Volume 4, Revision 1 [15]. Susquehanna, Units 1 and 2, have the same core power, flow, geometries, and bundle geometries. Both units

see the following

operate on a 24-month fuel cycle resulting in minimal differences in fuel and core neutronic design. Based on the minimal differences between Units 1 and 2, the information that is included in the submittal is provided for Unit 2 Cycle 21 – limited information needs to be provided for Unit 1. Therefore, the licensee will include the Unit 1 Cycle 23 reload safety analysis report with transmittal of the combined operating limits report prior to startup from the Unit 1 Cycle 23 refueling outage (i.e., spring 2022), which will load the first reload batch of ATRIUM 11 fuel into the Unit 1 reactor core.

The licensee performed thermal-hydraulic analyses to verify that the design criteria were satisfied and to establish thermal operating limits with acceptable margins of safety during normal reactor operation and AOOs. Due to reactor and cycle operating differences, many of the analyses supporting these thermal-hydraulic operating limits were performed on a plant- and cycle-specific basis and are documented in plant- and cycle-specific reports. Table 3.1 of ANP-3761 lists the applicable thermal-hydraulic design criteria, analyses, and results for hydraulic compatibility, thermal margin performance, fuel centerline temperature, rod bow, bypass flow, stability, LOCA analysis, CRDA analysis, ASME over-pressurization analysis, and seismic/LOCA liftoff. The subsections below summarize the results from selected design criteria and analyses results.

Hydraulic Characterization

Basic dimension parameters for the ATRIUM 10 and ATRIUM 11 fuel assembly designs are summarized in Table 3.2 of ANP-3761. Table 3.3 provides a comparison of key hydraulic characteristics, including loss coefficients, flow resistances, and friction factors for the two fuel assembly designs. A summary of the testing and analysis performed to determine the hydraulic characteristics for the fuel assembly designs is included in Section 3.1 of ANP-3761.

The testing and analysis approaches used for the ATRIUM 11 fuel assembly design are similar to the approaches that have previously been used to characterize the ATRIUM 10 fuel assembly design, as reviewed by the NRC for applicability to other plants operating in the MELLLA flow regime. There are no attributes associated with the ATRIUM 11 fuel assembly design that would be expected to require special treatment relative to the ATRIUM 10 fuel assembly design. Therefore, the NRC staff finds the hydraulic characterization of the ATRIUM 11 fuel assembly design to be acceptable.

Thermal-Hydraulic Compatibility

The thermal-hydraulic compatibility analyses were performed in accordance with the Framatome thermal-hydraulic methodology for BWRs [15]. The XCOBRA code predicts the steady-state thermal-hydraulic performance of fuel assemblies in BWR cores at various operating conditions and power distributions. The thermal-hydraulic compatibility analysis evaluates the relative thermal performance of the ATRIUM 10 and ATRIUM 11 fuel assembly designs that are planned to be inserted in the Susquehanna core. The analyses were performed for full-core and mixed-core configurations.

In essence, the hydraulic compatibility analysis **[[**

]] This analysis is performed utilizing different typical axial power shapes and radial power factors for rated and off-rated

conditions. The input conditions used for the analysis are listed in Table 3.4 of ANP-3761, while representative results are given in Tables 3.5 through 3.8 and Figures 3.2 and 3.3. **[[**

]] The most important result from the perspective of thermal-hydraulic compatibility is that the following parameters do not change significantly

throughout the transition from a full complement of ATRIUM 10 fuel to a full complement of ATRIUM 11 fuel: **[[**

]] The performance characteristics important for safety analysis purposes are captured by the correlations and

specifications unique to each fuel assembly design.

Based on the changes in **[[** **]]** caused by the transition from ATRIUM 10 fuel to ATRIUM 11 fuel, the NRC staff finds that the hydraulic compatibility analyses for the transition cores at Susquehanna, Units 1 and 2, provide reasonable assurance that the resident and co-resident fuel designs will satisfy the thermal-hydraulic design criteria for mixed cores.

Thermal Margin Performance

The thermal margin analyses were performed using the NRC-approved thermal-hydraulic methodology for steady-state critical power ratio (CPR) evaluations with XCOBRA listed in the Susquehanna TSs. Empirical correlations for the ATRIUM 10 [16] and ATRIUM 11 [17] fuel assembly designs were used based on results of boiling transition test programs. These CPR correlations account for assembly design features through modification of the K-factor term in the CPR correlations.

The hydraulic compatibility analysis discussed in the previous subsection includes steady-state CPR values calculated for various radial peaking factors. As expected, **[[**

]]

Therefore, there is no significant impact on the thermal margin performance for either fuel assembly design as a result of mixed core operations. Since the fuel assembly design-specific considerations are addressed by use of fuel assembly design-specific CPR correlations, appropriate thermal margins will be maintained through use of appropriate constraints on design and operation of the cores throughout the transition.

Based on the above, the NRC staff finds that the introduction of ATRIUM 11 fuel will not cause an adverse impact on thermal margin for the co-resident ATRIUM 10 fuel.

Rod Bow
Rod bow is addressed as part of the mechanical design analyses see Section 3.2.2.3 (Fuel

Assembly Mechanical Design Evaluation) of this SE for further discussion). **[[**

Bypass Flow

[[

]]

The NRC staff finds this disposition to be acceptable based on the fact that this is consistent with Framatome methodologies and the impact is appropriately evaluated. Based on the above, the NRC staff finds that adequate bypass flow will be available with the introduction of the ATRIUM 11 fuel design and that applicable design criteria will be met.

Stability

The thermal-hydraulic design criteria approved by the NRC in ANF-89-98(P)(A) include a requirement to confirm that the stability characteristics for a new fuel design are equivalent to or better than that of prior approved fuel designs. This evaluation is performed using the STAIF code as prescribed in ANF-89-98(P)(A), and the results are documented in ANP-3761 for Susquehanna. This evaluation is adequate to meet the requirements within the NRC-approved generic fuel assembly mechanical design criteria used by Framatome to qualify new fuel designs. However, the NRC staff did not review the STAIF evaluation in detail because the confirmation density algorithm-based hardware trip is expected to detect and suppress any power oscillations resulting from stability issues, as confirmed through the use of the Option III analytical methodology. Additionally, the fact that the ATRIUM 11 fuel assembly design does not represent a significant departure from prior fuel assembly designs provides assurance that the assumptions made in the stability analyses have not been invalidated. This would ensure that the regulatory requirements associated with stability performance are met.

Void Fraction

Section 5.1 of ANP-3753P discusses the use of the **[[**

]] correlation for ATRIUM 11 fuel. The NRC staff questioned **[[**

]] Based on discussions during an audit during the review of the Brunswick Steam Electric Plant (Brunswick) fuel transition to ATRIUM 11, it was clarified that

]]

the **[[**

evidence, **[[**

]] This

]], and the approach is acceptable.

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- 21 - 3.3.3 Thermal-Hydraulic Design Conclusion

The NRC staff reviewed the thermal-hydraulic compatibility analytical approaches and results intended to demonstrate that the ATRIUM 11 fuel design is hydraulically compatible with the ATRIUM 10 fuel currently used at Susquehanna. The NRC staff determined that the generic thermal-hydraulic design criteria, as approved by the NRC in ANF-89-98(P)(A), have been used in the analyses. Based on the above, the NRC staff concludes that although the ATRIUM 10 and ATRIUM 11 fuel assemblies contain a number of differences in their geometric and hydraulic characteristics, they remain hydraulically compatible.

3.4 Stability

Stability methodology at Susquehanna is described in Section 7 of ANP-3753P. Stability analyses at Susquehanna are performed using the approved Option III stability methodology in the RAMONA5-FA [18], which was approved before the implementation of chromia-doped fuel. Methods in this stability solution were updated to account for the use of ATRIUM 11 fuel rod property models. Both Susquehanna units continue to implement stability Option III.

3.4.1 Regulatory Basis

The plant-specific Option III long-term stability solution and related licensing basis were developed to comply with the requirements of GDC 10 and 12.

GDC 10, "Reactor design," states that, "The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences."

GDC 12, "Suppression of reactor power oscillations," states that, "The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed."

Consistent with GDC 10 and 12, the NRC staff determines whether the licensee performs the plant-specific trip setpoint calculations for long-term stability using acceptable methodologies as prescribed in the SRP (NUREG-0800), Sections 4.4 and 15.9.

3.4.2 Technical Evaluation

The RAMONA5-FA [18] and STAIF [19] methods used in the Option III methodology have been updated to address advanced fuel design features of ATRIUM 11 using **[[**

]]. The fuel property models implemented are the same models used in the Framatome generic anticipated transient without scram (ATWS)-I methodology described in ANP-10346NP-A [20]. Susquehanna is only implementing the fuel rod property models from the Framatome generic ATWS-I methodology. While the licensee references the topical report for the Framatome generic ATWS-I methodology, it does not intend to adopt the methodology in its entirety, but only adopt the fuel rod property models for chromia-doped fuel. While the fuel rod property models are included within the description of the ATWS-I methodology, they are not a specific feature of the ATWS-I methodology itself. Rather, this topical report was the first opportunity for Framatome to document the implementation of chromia-doped fuel properties and models within the RAMONA5-FA code for NRC review and approval. The NRC staff's evaluation of the implementation of these models, as applicable to the intended application for Susquehanna, is provided in the following sections.

3.4.2.1 **[[ ]]** Fuel Rod Models

ANP-3753P describes that in the Option III methodology at Susquehanna, **[[**

]] The licensee accounted for the effects of chromia doping in fuel pellets by modifying the standard UO_2 thermal conductivity and **[[ ]]**

models. The material properties, pellet-clad gap heat transfer coefficient, and radial power distribution in fuel pellets used in the Option III methodology at Susquehanna are identical to that used in the generic ATWS-I methodology. Although **[[**

plant-specific evaluation of these areas is provided below.

3.4.2.1.1 Material Properties

The ANP-10346P-A methodology uses fuel pellet and cladding thermophysical properties based on **[[ ]].** The NRC staff finds this approach acceptable for use in the RAMONA5-FA ATWS-I calculations at Susquehanna because these models account for all important fuel characteristics relevant to ATWS-I, including the **[[**

]].

Appendix A to Duke Energy, ANP-3782P, Revision 1, "Brunswick ATRIUM 11 Advanced Methods Response to Request for Additional Information," dated May 29, 2019 (ADAMS Accession No. ML19149A320 (Non-Public)). Reference [22] includes an update to ANP-10346P-A that, among other changes, appends Appendix D, which presents modified fuel rod models that account for chromia doping of the UO_2 fuel pellets. The fuel thermal conductivity model was adapted from the approved RODEX4 model in Reference [14]. The

[[ ]] model was developed by benchmarking to the approved RODEX4 model. The NRC staff finds these models acceptable for use in characterizing chromia-doped fuel properties for ATWS-I analyses at Susquehanna because these models are based on previously reviewed and approved models for chromia-doped fuel using the methodology described in ANP-10346P-A.

3.4.2.1.2 Pellet Clad Gap Heat Transfer Coefficient

Based on the similarity **[[ ]]**, inclusion of the important physics relevant to ATWS-I, close agreement of the RAMONA5-FA ATWS-I results to measured BWR stability data, and **[[**

]] of the stability results under most scenarios to variations in gap conductance, the NRC staff concludes that the fuel rod heat transfer model, including the gap conductance model, is acceptable for use in the ATWS-I analyses.

3.4.2.1.3 Radial Power Deposition Distributions in Fuel Pellets

The NRC staff has reviewed the methodology and determined that it provides the needed accuracy for calculating the radial power distribution in fuel pellets, including

]], a

[[

[[

3.4.3 Stability Conclusion

]]

]] Therefore, the NRC staff finds the radial power distribution methodology to be acceptable.

3.4.2.2 STAIF Reactor Benchmarks Using New Fuel Rod Property Models

The licensee reanalyzed all reactor benchmarks in the STAIF benchmarking suite (Section 4.0 of Reference [19]) using the new fuel rod property models evaluated in Section 3.4.2.1

(**[[ ]]** Fuel Rod Models) of this SE. The NRC staff compared decay ratios calculated with the new fuel rod property models to the measured decay ratios from various stability tests. **[[**

]]

3.4.2.3 RAMONA5-FA Reactor Benchmarks Using New Fuel Rod Property Models

The licensee reanalyzed all reactor benchmarks in the RAMONA5-FA benchmarking suite (Section 5.0 of Reference [19]) using the new fuel rod property models evaluated in Section 3.4.2.1 (**[[** **]]** Fuel Rod Models) of this SE. The predicted growth ratios and frequencies using the RAMONA5-FA with RODEX4 based fuel property models were compared to the results using the original fuel rod property models for each benchmark.

Based upon its review, the NRC staff determined that the Option III calculation procedure provides an acceptable means of determining licensing basis safety limit for minimum critical power ratio (SLMCPR) protection during anticipated stability events at Susquehanna.

3.5 ATRIUM 11 Transient Demonstration 3.5.1 Regulatory Basis

In addition to the GDC described in Section 2.0 of this SE, the following regulatory

3.5.2.1

10 CFR 50.62, “Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants,” which requires licensees to provide the means to address an ATWS, which means an AOO as defined in Appendix A to 10 CFR Part 50 followed by the failure of the reactor trip portion of the protection system specified in GDC 20.3.5.2Technical Evaluation

MCPR Fuel Cladding Integrity Safety Limit

Section 5.4 of ANP-3753P describes the SLMCPR methodology at Susquehanna. The ANP-10307PA, Revision 0 methodology used at Susquehanna is [23] is used to determine that

99.9 percent of fuel rods are expected to avoid boiling transition during normal reactor operation and AOOs. Of note is a plant-specific extension to the approved methodology. After reviewing the licensee’s RAI response, the NRC staff concluded that the plant-specific extension is acceptable because the licensee has an appropriate process in place if the error bounds are exceeded.

3.5.2.2 AOOs

The licensee submitted information to demonstrate the applicability of the AURORA-B AOO methodology for Susquehanna, compliance with the NRC limitations and conditions imposed for application of the AURORA-B AOO topical report, and a demonstration analysis of select licensing basis events using the AURORA-B AOO methodology to demonstrate that the results of the analyses meet the applicable acceptance criteria. This information is found in the ANP-3753P and ANP-3783P attachments to the LAR [1] in conjunction with the licensee's responses to the NRC staff's RAIs [2], [3].

3.5.2.2.1 AURORA-B AOO Methodology Overview

The AURORA-B AOO methodology and the NRC staff's SE of the methodology is found in ANP-10300NP-A, Revision 1 [24]. The methodology is used to evaluation transients, postulated accidents, and beyond design-basis scenarios for BWRs. The methodology is built upon three computers codes:

- • S-RELAP5, which provides the thermal-hydraulic code to simulate BWR system response;
- • MB2-K, which provides the core neutronic response; and
- • RODEX4, which provides the thermal-mechanical response of the individual fuel rods.

  The methodology uses non-parametric order statistics to evaluate the impact of uncertainties in the methodology. This means that for each scenario analyzed, several runs are executed (e.g., 59 runs), varying certain parameters to achieve a result at a certain confidence level. In the case of the AURORA-B AOO methodology, the uncertainty analysis is used to bound the 95 percent worst case result at 95 percent confidence. Table 3.6 of the SE for the AURORA-B AOO methodology contains the uncertainty parameters used for the uncertainty analysis.

  The licensee provided a demonstration analysis in the ANP-3783P attachment to the LAR. The demonstration analysis provides analyses for the following transients, accidents, and beyond design-basis events:

  - • load rejection without bypass/turbine trip without bypass;
  - • feedwater controller failure;
  - • inadvertent startup of the high-pressure coolant injection (HPCI) pump;
  - • ASME over-pressurization analysis; and
  - • ATWS over-pressurization analysis.

3.5.2.2.2 Applicability of the AURORA-B AOO Methodology to Susquehanna

The NRC staff reviewed the LAR to ensure that the AURORA-B AOO methodology was applicable to Susquehanna. As described in Section 3.1 (Applicability of Framatome BWR

Methods to Susquehanna with ATRIUM 11 Fuel) of the SE for the AURORA-B AOO methodology [24], the methodology is applicable, in part, to BWR/3 through BWR/6 plants. Since Susquehanna is a BWR/4 plant, the methodology is applicable to Susquehanna in this respect. The NRC staff considered three additional major considerations to determine the applicability of the methodology to Susquehanna: (1) the applicability for use with ATRIUM 10 fuel; (2) the applicability for use with ATRIUM 11 fuel; and (3) the applicability for use in the MELLLA operating domain.

Upon initial implementation of the AURORA-B AOO methodology, the Susquehanna core will still contain ATRIUM 10 fuel. Therefore, the NRC staff considered the applicability of the AURORA-B AOO methodology to this fuel design. In general, the AURORA-B AOO methodology was developed around the ATRIUM 10 and ATRIUM 10XM fuel bundle design (see Section 3.3.1 (Regulatory Basis) of the SE for the AURORA-B AOO methodology). Also, as implied in Limitations 4 and 5 in Section 5.0 of the SE for the AURORA-B AOO methodology, ATRIUM 10 and ATRIUM 10XM are not new fuel designs relative to the AURORA-B AOO methodology and need not be explicitly justified for use with the methodology. Susquehanna is operating with ATRIUM 10 fuel within the fuel design limits. Since the AURORA-B AOO methodology was developed based on the ATRIUM 10 and ATRIUM 10XM fuel design, and Susquehanna is operating with ATRIUM 10 fuel within its approved design, the NRC staff determined that the AURORA-B AOO methodology is applicable to Susquehanna with

ATRIUM 10 fuel.

As described in Limitations 4 and 5 in Section 5.0 of the SE for the AURORA-B AOO methodology, a licensee is required to justify new fuel designs relative to those approved for use in the AURORA-B AOO methodology. ATRIUM 11 is a new fuel design for use with the AURORA-B AOO methodology. The licensee provided justification in the ANP-3753P and ANP-3783P attachments to the LAR. Specifically, the licensee provided justification for ATRIUM 11 with respect to transients and accidents in Section 4.0 of ANP-3783P and ATWS in Section 8.3 of ANP-3753P. The major concern for the transients and accidents is how the void-quality correlation uncertainties are incorporated into the analyses for transients and accidents. These uncertainties are important because they could impact the results of the analyses (e.g., MCPR). The NRC staff notes that it is also important for the licensee to use a void-quality correlation that is applicable to the fuel it is using. For Susquehanna, the licensee stated that it will be using the **[[** **]]** void correlation for the ATRIUM 11 fuel.

As described in the LAR, the licensee stated that these uncertainties were not explicitly included in the transient and accident analyses. Rather, they are implicitly included in the power prediction, and the uncertainties in the power prediction are included in the analysis to determine the SLMCPR. Susquehanna uses the SAFLIM3D methodology [25]. The NRC staff confirmed that the power prediction was incorporated into the SAFLIM3D methodology. Additionally, the NRC staff confirmed that the Susquehanna methodology used to calculate the power prediction, MICROBURN-B2 [26], incorporated the void-quality correlation. Since the licensee incorporates the void-quality uncertainty in the power prediction, and the power prediction uncertainty is included in the calculation of the SLMCPR, the NRC staff determined that the licensee appropriately addressed the ATRIUM 11 fuel for SLMCPR.

The LAR describes that the void-quality correlation uncertainty is incorporated into the delta critical power ratio (ΔCPR) as a result of a transient that is used to determine the operating limit minimum critical power ratio (OLMCPR).[1] The licensee also discussed how the void-quality correlation uncertainty is implicitly accounted for by conservatism in the computer code models and input parameters used for the analysis. The conservatism in the computer codes exist because they are tuned to bound the power increases relative to the benchmark tests. The uncertainty in the void-quality correlation uncertainty will impact the uncertainty in the power prediction (which has a direct influence on ΔCPR). Since the computer codes are tuned to bound the power predictions in the benchmark tests, they will inherently incorporate the void-quality correlation uncertainty. The licensee also stated that the input parameters for the transient analysis are biased to, in part, account for void-quality correlation uncertainty. Since the void-quality correlation is inherently accounted for in the transient analysis to determine ΔCPR, and the initial conditions are conservatively biased, the NRC staff determined that the licensee has adequately addressed the ATRIUM 11 fuel for ΔCPR.

The licensee intends to use the AURORA-B AOO methodology to analyze ATWS events except for ATWS-I. ATWS analysis is an approved analysis in the AURORA-B AOO methodology. In Section 8.1 of ANP-3753P, the licensee justified that the ATWS vessel over-pressurization event in the AURORA-B AOO code suite is not impacted by the ACE/ATRIUM 11 critical power correlation that was approved for ATRIUM 11 fuel. The justification provided is that the AURORA-B AOO methodology ignores dryout (and, therefore, does not need to use a critical power correlation) in the ATWS vessel over-pressurization event because it is more conservative to assume maximum heat transfer to the coolant for an overpressure event. The NRC staff determined that this justification is reasonable because maximizing heat transfer to the coolant will increase the pressure in the vessel, which is appropriate for analyzing an overpressure event. The NRC staff also determined that ignoring the dryout in the fuel is conservative because once the fuel is in dryout, heat transfer from the rod to the coolant is diminished, and heat transfer to the coolant would, therefore, be reduced.

The licensee also discussed the void-quality correlation's impact on the ATWS vessel overpressure analysis. Like the transient and accident discussion above, the licensee provided justification that the void-quality correlation uncertainties are inherently incorporated into the code, and that the input parameters are conservatively biased to account for uncertainties. Therefore, the NRC staff determined that the void-quality correlation uncertainties are appropriately accounted for in the ATWS methodology. The NRC staff notes that for ATWS analyses, the void-quality correlation is more important for predicting peak vessel pressure. For Susquehanna, the licensee stated that it will be using the void-quality correlation found in the ATRIUM 11 fuel.

Section 8.3 of ANP-3753P contains an evaluation of the ATWS containment heatup calculation. The licensee provided justification that **[[**

]]. The ATWS containment analysis is addressed in Section 3.5.2.2.5 (ATWS Containment Heatup) of this SE.

3.5.2.2.3 AURORA-B Methodology Limitations and Conditions

The AURORA-B AOO methodology contains 26 limitations and conditions in Section 5.0 of the NRC staff's SE (ANP-10300P-A, Revision 1). The licensee stated in the LAR that the limitations and conditions for the Framatome topical reports are included in ANP-2637P, "Boiling Water

[1] OLMCPR is calculated as the sum of the SLMCPR and the ΔCPR. Susquehanna operates above the OLMCPR to ensure that an AOO does not cause the plant to violate the SLMCPR.

Reactor Licensing Methodology Compendium," and compliance with the limitations and conditions is assured by implementing them within the engineering guidelines or by incorporating them into the computer codes. Discussion of the limitations and conditions for the AURORA-B AOO methodology is found starting on page 5-32 of ANP-2637P.

The NRC staff notes that Limitations and Conditions 20 through 26 in Section 5.2 of the SE for the AURORA-B AOO methodology are related to the change process of the methodology itself. The licensee requested AURORA-B AOO methodology as approved; therefore, these limitations and conditions are not applicable to the LAR.

Limitation and Condition 1 relates to using the method's coupled calculational devices within their approved range. The coupled calculational devices used for this analysis are RELAP5, MB2-K, MICROBURN-B2, and RODEX4. The NRC staff confirmed that these calculational devices are used within their approved ranges.

Limitation and Condition 2 relates to the cladding oxidation limit (i.e., 13 percent) when using the Cathcart-Pawal oxidation correlation. The NRC staff confirmed that the AURORA-B AOO results meet this limit.

Limitation and Condition 3 relates to using the approved uncertainty distributions in the analysis. The NRC staff confirmed that the generic uncertainty distributions presented in Table 3.2 of ANP-3783P are consistent with those in Table 3.6 of the SE for the AURORA-B methodology. For the **[[** **]]**, the licensee stated that the range was developed based on the approved process in Section 3.6.4.10 of the methodology. Therefore, the NRC staff finds that the licensee adequately addressed this limitation and condition.

Limitation and Condition 4 relates to the justification of void fraction prediction for new fuel designs. The licensee discussed the void fraction prediction in Section 6.1 of ANP-3753P. The NRC staff reviewed the void fraction prediction in Section 3.3.2 (Technical Evaluation) of this SE and found that it was acceptable. Therefore, the NRC staff finds that the licensee adequately addressed this limitation and condition.

Limitation and Condition 5 relates to the justification of the **[[** **]]** void-quality correlation for new fuel designs. The licensee discussed the void-quality correlation in Section 5.1 of ANP-3753P. The NRC staff reviewed this in Section 3.3.2 (Technical Evaluation) of this SE and found that it was acceptable. Therefore, the NRC staff finds that the licensee adequately addressed this limitation and condition.

Limitation and Condition 6 relates to the use of the **[[**

]] The licensee stated that it followed the approved process of Sections 3.6.4.10 and 3.6.4.13 for

[[**]]** of the methodology to determine the uncertainty range. Therefore, the NRC staff finds that the licensee adequately addressed this limitation and condition.

Limitation and Condition 7 relates to the licensee providing justification for the key plant parameters and initial conditions selected for performing sensitivity analyses on an event-specific basis. In RAI response 2.3 [3], the licensee described how compliance with this requirement will be completed in the reload safety analysis report (RSAR) when it is submitted

in November 2020. Therefore, the NRC staff finds that the licensee adequately addressed this limitation and condition.

Limitation and Condition 8 relates to the truncation of sampling ranges for uncertainty distributions used in the non-parametric order statistics analyses. The licensee discussed in Section 3.2.2 of ANP-3783P how the sampling performed complies with the limitations and conditions of the SE for the AURORA-B methodology. The NRC staff confirmed that the licensee adequately addressed this limitation and condition.

Limitation and Condition 9 relates to uncertainties of medium or highly ranked phenomena identification and ranking table (PIRT) phenomena that are not addressed in given non-parametric order statistics analysis via sampling. To meet this limitation, the licensee modeled the phenomena as described in Tables 3.2 and 3.4 of the SE for the AURORA-B methodology. The NRC staff confirmed that the licensee complied with the requirements of the tables and, therefore, has adequately addressed this limitation and condition.

Limitation and Condition 10 relates to the assumptions of **[[**

]]. The licensee stated that it complied with the requirements of Tables 3.2 and 3.4 of the SE for ANP-10300P-

A, Revision 1 [27], as they relate to this limitation. The NRC staff confirmed that the licensee complied with the requirements of the tables and, therefore, has adequately addressed this limitation and condition.

Limitation and Condition 11 relates to justification for uncertainties used for highly ranked plant-specific PIRT parameters. In RAI response 2.3 [3], the licensee described how compliance with this requirement will be completed in the RSAR when it is submitted in November 2020. Therefore, the NRC staff finds that the licensee adequately addressed this limitation and condition.

Limitation and Condition 12 relates to plant-specific changes to AURORA-B to enhance

[[
the AURORA-B EM to the **[[**

]] when applying

]]. For Susquehanna, the Inadvertent HPCI event is identified as potentially limiting (see response to RAI 2.1.a). A method to evaluate the mixing was proposed

in Section 6.3 of the Methods Applicability Document (ANP-3753P) to be evaluated using [[

]] Once the amount of mixing has been determined, the AURORA-B licensing model will be constructed. In order to ensure a conservative estimation of mixing is used, [[

]] The licensee described how compliance with the requirement will be completed in the Reload Safety Analysis Report, which will be submitted

following approval. Therefore, the NRC staff finds that the licensee adequately addressed this limitation and condition.

Limitation and Condition 13 relates to the use of nominal calculations with the AURORA-B evaluation model. The events in this category are generally expected to be benign and, hence,

non-limiting. The licensee dispositions events in this category as non-limiting in its UFSAR; therefore, no additional evaluation is required. The NRC staff finds that the licensee adequately addressed this limitation and condition.

Limitation and Condition 14 relates to the scope of the NRC's approval for AURORA-B. Specifically, the approval does not include the advanced BWR design. Since Susquehanna is not an advanced BWR, its use is within the scope. Therefore, the NRC staff finds that the licensee adequately addressed this limitation and condition.

Limitation and Condition 15 relates to the application of AURORA-B to BWR/2s at extended power uprate or extended flow window conditions. Susquehanna is not a BWR/2; therefore, this limitation and condition is not applicable.

Limitation and Condition 16 relates to the justification of a plant-specific conservative flow rate. In RAI response 2.3 [3], the licensee described how compliance with this requirement will be completed in the RSAR when it is submitted in November 2020. Therefore, the NRC staff finds that the licensee adequately addressed this limitation and condition.

Limitation and Condition 17 relates to the uncertainty associated with heat transfer predictions in the film boiling regime. The licensee stated that no film boiling was encountered in the AOO analyses. Therefore, the NRC staff finds that the licensee adequately addressed this limitation and condition.

Limitation and Condition 18 relates to using conservative measures with the justification for the method of determining and applying conservative measures in future deterministic analyses for each figure of merit and re-performance of full statistical analysis if a scenario exceeds a

1σ magnitude difference. In RAI response 2.3 [3], the licensee described how compliance with this requirement will be completed in the RSAR when it is submitted in November 2020. Therefore, the NRC staff finds that the licensee adequately addressed this limitation and condition.

Limitation and Condition 19 relates to stipulations that would satisfy the 95/95 criterion for figures of merit calculated by AREVA in accordance with ANP-10300P-A. The licensee stated that all calculations completed in its demonstration analysis comply with the restrictions of Limitation and Condition 19. Therefore, the NRC staff finds that the licensee adequately addressed this limitation and condition.

The NRC staff reviewed each limitation and condition and finds that each was adequately addressed by the licensee for the demonstration case and will be supported by the RSAR when it is submitted in November 2020.

3.5.2.2.4 AURORA-B Methodology Analysis Results

The plant-specific UFSAR for Susquehanna contains the design-basis analyses to evaluate the effects of a wide range of AOOs. Since these analyses are performed on a cycle- and core configuration-specific basis during the standard reload analyses, the licensee provided demonstration analyses of the potentially limiting events.

Since the licensee's analysis in the LAR is a demonstration analysis, the NRC staff's review is to ensure that the licensee can adequately evaluate AOOs with the AURORA-B AOO

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methodology and ATRIUM 11 fuel. The NRC staff reviewed this analysis to ensure that the potentially limiting events are identified and considered for explicit analysis, the AOO results are realistic, and the results meet specified acceptable fuel design limits.

In the LAR, the licensee provided demonstration analyses for the load rejection without bypass event/turbine trip without bypass event, feedwater controller failure event, and inadvertent startup of the HPCI pump event.

For each cycle, the minimum set of analyses required to license the cycle is determined based on the disposition of events and operational flexibility needed such as equipment out of service and exposure windows. **[[**

]]

To ensure that there is appropriate coverage of the parameters used in the uncertainty analysis and to ensure that there are no significant trends with respect to the uncertainty parameters in the results, the NRC staff requested additional information in RAI 2.2. Specifically, the NRC staff requested to review the following data sets for the load rejection no bypass/turbine trip without bypass event at 100 percent power/108 percent flow, main steam isolation valve closure ATWS event at 100 percent power and 99 percent flow, and high-pressure coolant injection event at 100 percent power/108 percent flow:

- • the sampled values of the uncertainty parameters for all cases executed and
- • the figure of merit results for all cases executed.

The licensee's RAI response showed that implementation of the AURORA-B AOO methodology is sufficient to meet GDC 10 and the ATWS acceptance criteria. The NRC staff reviewed the analysis approach for the transition to AURORA-B AOO methods and found that the approach covers the full range of operating conditions and is acceptable.

3.5.2.2.5 ATWS Containment Heatup

Section 8.3 in ANP-3753P provides the licensee's evaluation of ATWS containment heatup. The NRC staff's evaluation of this section follows.

Changes in fuel design can impact the power and pressure excursions during an ATWS event. The power and pressure excursion changes can impact the suppression pool and containment temperature and pressure responses.

[[

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Additionally, the NRC staff requested information in RAI 1.b regarding the analysis performed to confirm that the fuel transition is bounded by the current analysis of record and the quantitative results for containment pressure and suppression pool temperature response. In its response, the licensee states, "the current licensing basis for Susquehanna ATWS containment shows the peak suppression pool temperature for MELLLA was 206 °F [degrees Fahrenheit] and the peak containment pressure was 16.1 psig [pounds per square inch gauge]...." The analysis is based on **[[ ]]** After this was completed, the licensee determined that the **[[**

]]

Finally, because containment heatup is directly impacted by the stored energy in the fuel and decay heat, a quantitative comparison of the decay heat between Framatome fuel types was reviewed [28]. The study compared **[[**

]]

Based on the above, the NRC staff determined that the analysis of record remains bounding for ATWS containment heatup with the transition to ATRIUM 11 at Susquehanna. Therefore, the NRC staff concludes that the applicable regulatory requirements continue to be met.

3.5.3 Application of Framatome Methodologies for Mixed Cores

Appendix A of ANP-3753P discusses the application of Framatome methodologies to mixed cores.

3.5.4 Transient Demonstration Conclusion

Regarding AOO and ATWS, the NRC staff reviewed the information in the licensee's submittals pertaining to the analysis of AOO and ATWS events for Susquehanna, including the original submittal as well as relevant responses to RAIs [2], [3]. Based upon its review, as summarized above, the NRC staff concludes that:

- The licensee has proposed to implement the AURORA-B AOO evaluation model in an acceptable manner and

- Compliance with the applicable regulatory requirements has been demonstrated.

3.6 ATWS-1

3.6.1 Regulatory Basis

In addition to the GDC described in Section 2.0 (Regulatory Evaluation) of this SE, the following regulatory requirements apply to the ATWS-I evaluation.

- 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants," which requires that the licensee provide an acceptable reduction of risk from ATWS events by inclusion of prescribed design features and demonstrating their adequacy in mitigation of the consequences of an ATWS event. Within the context of the review of the submittal, the ATWS-I analyses are intended to demonstrate that the combination of automated plant functions and prescribed operator actions will be sufficient to preclude fuel failure.

- 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," which, although not directly applicable to the ATWS-I event because it is intended to address postulated LOCAs rather than ATWS events, this regulation does present a set of acceptance criteria for ensuring adequate cooling of fuel such that significant fuel failures do not occur.

The SRP (NUREG-0800) is the primary regulatory guidance document used by the NRC staff to support its review of this LAR. In particular, SRP Section 15.8, "Anticipated Transients Without Scram", establishes acceptance criteria for ATWS events. Although SRP Section 15.8 includes additional GDC beyond those listed above, they define vessel, ECCS, and containment performance requirements. These are not a significant concern for ATWS-I events; therefore, these GDC were not considered as part of this review.

3.6.2 Technical Evaluation

The NRC staff noted that a plant-specific ATWS-I analysis was not included in the submittal. As referenced in the licensee's UFSAR [29], the analysis used by the licensee is found in NEDO-32047-A [30]. NEDO-32047 has been reviewed and approved by the NRC staff for generic use when the assumptions within the ATWS-I analyses are bounding. Since Susquehanna has not yet elected to operate in an extended flow window, the assumptions of the generic analyses remain bounding.

The thermal-hydraulic fuel properties of ATRIUM 11 fuel do not affect the ATWS-I results since they are demonstrated to be more stable than the historical fuel product lines used in the generic analyses (see Section 3.6 of ANP-3761).

3.6.3 Conclusion

Based upon its review, the NRC staff determined that the generic ATWS-I analyses found in Susquehanna NEDO-32047-A are an acceptable means of determining protection during instability events at Susquehanna.

3.7 LOCA Analysis for ATRIUM 11 Fuel

NRC regulations require that licensees of operating light-water reactors analyze a spectrum of accidents involving the loss of reactor coolant to assure adequate core cooling under the most limiting set of postulated design-basis conditions. The postulated spectrum of LOCAs range from scenarios with leakage rates just exceeding the capacity of normal makeup systems up through those involving rapid coolant loss from the complete severance of the largest pipe in the RCS.

To support the planned transition to ATRIUM 11 fuel at Susquehanna, the licensee analyzed the spectrum of LOCA events for this fuel design using the AURORA-B LOCA evaluation model [31]. The AURORA-B LOCA evaluation model is an Appendix K, “ECCS Evaluation Models,” to 10 CFR Part 50 conformant analysis methodology that was approved by the NRC in

March 2019.

As described in the evaluation below, the NRC staff reviewed the licensee's implementation of the AURORA-B LOCA evaluation model for Susquehanna to ensure compliance with applicable regulatory requirements. The NRC staff's review activities associated with the LOCA analysis for Susquehanna focused upon the review of pertinent sections of the licensee's submittals (particularly ANP-3784P). The NRC staff further conducted a regulatory audit on November 15, 2019 [32], which supported its review of the information.

3.7.1 Applicable Regulatory Requirement

The following regulatory requirements described below are pertinent to the analysis of the spectrum of LOCA events postulated to occur:

- 10 CFR 50.46;
- Appendix K to 10 CFR Part 50; and
- Appendix A to 10 CFR Part 50, GDC 35.

3.7.1.1 10 CFR 50.46

Key regulatory requirements specified in 10 CFR 50.46 that are relevant to the proposed license amendments include the following:

- Each boiling or pressurized light-water reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding[2] must perform analysis of core cooling performance under postulated LOCA conditions using an acceptable evaluation model.

- An acceptable LOCA evaluation model must be used that either applies realistic methods with an explicit accounting for uncertainties or follows the prescriptive, conservative requirements of Appendix K to 10 CFR Part 50.

- Core cooling performance must be analyzed for several postulated LOCAs of different sizes, locations, and other characteristics to ensure that the most severe event is calculated.

Furthermore, 10 CFR 50.46(b) provides acceptance criteria for analyses of the spectrum of LOCA events, which are summarized below.

Table 1: 10 CFR 50.46 Acceptance Criteria

Subparagraph	Figure of Merit	Acceptance Criterion
(b)(1)	Peak Cladding Temperature	≤ 2,200 °F
(b)(2)	Maximum (Local) Cladding Oxidation	≤ 17% of Unoxidized Thickness
(b)(3)	Maximum (Core-Wide) Hydrogen Generation	≤ 1% of Hypothetical Amount
(b)(4)	Core Geometry	Amenable to Cooling
(b)(5)	Long-Term Cooling	Maintained

[2] Note that the applicability conditions stated in this requirement would be satisfied for both proposed ATRIUM 11 and co-resident ATRIUM 10 fuel designs loaded at Susquehanna.

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In accordance with Limitation and Condition 4 from the NRC staff's final SE on ANP-10332P [31], the AURORA-B LOCA evaluation model may not be referenced as a basis for demonstrating adequate long-term core cooling in satisfaction of 10 CFR 50.46(b)(5).

To demonstrate continued adherence to this requirement, the licensee cited existing licensing basis analysis performed on a generic basis by the nuclear reactor vendor (i.e., General Electric), which is documented in approved topical report NEDO-20566A [33]. Accordingly, the proposed license amendments would not modify the licensing basis method for demonstrating satisfaction of the requirement in 10 CFR 50.46(b)(5) for adequate long-term core cooling.

3.7.1.2 Appendix K to 10 CFR Part 50
Appendix K to 10 CFR Part 50 consists of two parts:

- • required and acceptable features of LOCA evaluation models and
- • documentation required for LOCA evaluation models.

    The first part specifies modeling requirements and acceptable methods for simulating significant physical phenomena throughout all phases of a design-basis LOCA event, including relevant heat sources, fuel rod performance, and thermal-hydraulic behavior.

    The second part specifies requirements for the documentation of LOCA evaluation models, including a complete description, a code listing, sensitivity studies, and comparisons against experimental data.

    The NRC staff's basis for concluding that the AURORA-B LOCA evaluation model used to perform the LOCA analysis for Susquehanna conforms to the requirements of Appendix K to 10 CFR Part 50 is discussed in Section 6.2.1 of the NRC staff's SE on ANP-10332P [31].

3.7.1.3 Appendix A to 10 CFR Part 50, GDC 35

The GDC of Appendix A to 10 CFR Part 50 outline criteria for the design of nuclear power plants, typically in broad, qualitative terms. In particular, GDC 35 requires abundant core cooling sufficient to (1) prevent fuel and cladding damage that could interfere with continued effective core cooling and (2) limit the metal-water reaction on the fuel cladding to negligible amounts. GDC 35 further requires suitable redundancy of the ECCS such that it can accomplish its design functions assuming a single failure, irrespective of whether its electrical power is supplied from offsite or onsite sources. Section 3.1 of the Susquehanna UFSAR describes how the plant was designed to ensure conformance to GDC 35 and other GDC from Appendix A to 10 CFR Part 50.

3.7.2 Acceptability of LOCA Evaluation Model

The licensee analyzed the spectrum of postulated LOCA events to verify the satisfaction of applicable regulatory requirements following the transition to ATRIUM 11 fuel. The licensee used the AURORA-B LOCA evaluation model [31] to demonstrate compliance with the four acceptance criteria from 10 CFR 50.46 that apply to the short-term LOCA analysis (i.e., subparagraphs (b)(1) through (b)(4) in Table 1).

The AURORA-B LOCA evaluation model is an S-RELAP5 based methodology that incorporates a kernel of transient fuel rod thermal-mechanical subroutines from the RODEX4 code. As documented in an SE dated March 26, 2019 [31], the NRC staff found the AURORA-B LOCA evaluation model acceptable for application to LOCA analysis for BWR/3-BWR/6 plants. Susquehanna, Units 1 and 2, are General Electric BWR/4 plants.

While the generic evaluation model proposed by the licensee to support its proposed fuel transition has been previously found to be acceptable], the NRC staff reviews licensee implementation of analytical evaluation models to ensure:

- • Confirmation of acceptable plant-specific inputs to the evaluation model (Section 3.7.3.1 of this SE);
- • Confirmation of adherence to the approved evaluation model (Sections 3.7.3.2 and 3.7.3.3);
- • Confirmation that results calculated using the evaluation model satisfy regulatory acceptance criteria and otherwise conform to expectations (Section 3.7.4); and
- • Verification of acceptable responses to limitations and conditions specified in the NRC staff's SE (Section 3.7.5).

The following sections of this SE describe the NRC staff's review of these areas.
3.7.3 Evaluation Model Implementation
3.7.3.1 Plant-Specific Inputs

Some design differences may exist between Susquehanna, Units 1 and 2, that will affect the LOCA analysis. During an audit conducted on November 15, 2019, the NRC staff confirmed that the principal plant parameter input to the LOCA analysis is not changed between both units.

The NRC staff also confirmed during the audit that the LOCA break spectrum analysis based on a future equilibrium cycle of ATRIUM 11 fuel would bound transition cycles containing some co-resident legacy fuel bundles of the ATRIUM 10 design. The licensee stated that the thermal-hydraulic compatibility analysis demonstrates that the thermal-hydraulic characteristics of the ATRIUM 11 and the coexistent ATRIUM 10 fuel are similar so that the core responses during LOCA will be insignificant for transition cores. The licensee further stated that the LOCA analysis **[[**

]].

The NRC staff found the licensee's response acceptable because the licensee provided adequate evidence that the impacts of transition cycles containing co-resident ATRIUM 10 fuel on the LOCA evaluation were small and within the conservative bounds established by the existing analysis so that the evaluation results meet the required design criteria.

3.7.3.2 Break Spectrum Implementation

The NRC staff's review found that the break spectrum analysis described in ANP-3784P generally conforms to the approved evaluation model documented in ANP-10332P-A [34].

The analysis for Susquehanna considered a spectrum of postulated double-ended guillotine and split breaks in the recirculation system (i.e., upper and lower suction piping, discharge piping).

Table 5.1 of ANP-3784P identifies the single failures considered in the Susquehanna LOCA analysis. The break spectrum analysis for Susquehanna focused upon four potentially limiting single failures among six potential limiting single failures identified in the UFSAR: (1) the failure of one train of direct current power (i.e., single failure (SF) of battery (direct current) power (SF-backup battery power (BATT)); (2) the failure of an automatic depressurization system valve (i.e., SF-automatic depressurization system valve (ADS)); (3) the failure of an opposite unit false LOCA signal (i.e., SF-LOCA); and (4) the failure of a low-pressure coolant injection system injection valve (i.e., SF-low-pressure coolant injection (LPCI)). The licensee determined that the ECCS resources for SF-diesel generator (DGEN) will be equal to or greater than that for SF-LOCA and that the ECCS resources for SF-HPCI (single failure of HPCI) will be equal to or greater than that for SF-BATT, such that the analyses for SF-DGEN and SF-HPCI are not considered because they can be bounded by SF-LOCA and SF-BATT, respectively. The NRC staff's review found that this determination was appropriate and that the licensee had considered the full set of postulated single failures defined in the Susquehanna UFSAR.

Consistent with ANP-10332P, break spectra were calculated for both mid- and top-peaked axial power profiles at the time of maximum fuel stored energy (i.e., near the beginning of the operating cycle). Furthermore, considering that Susquehanna is licensed to the MELLLA domain, sufficient initial statepoints were considered in the break spectrum analysis to provide confidence that the most limiting conditions have been analyzed. In particular, break spectra were performed for the statepoints shown below.

Table 2: LOCA Analysis Statepoints

Point	Operating Recirculation Loops	Reactor Power (% rated)	[[
1	2	102	
2	2	102	
3	1	[[ ]]	]]

The first two analyzed statepoints were selected to envelope the full range of permissible core flows at rated thermal power. The third statepoint represents limiting power[[ ]] conditions for single-loop operation. The NRC staff found the selected analysis statepoints acceptable because the licensee has taken appropriate regulatory guidance into account with respect to analyzing the MELLLA operating domain.

In RAI 4.1, the NRC staff requested additional information concerning how the LOCA analysis addresses the full suite of operating domains and equipment out-of-service conditions to which

Susquehanna has been licensed.[3] Table 3 below summarizes the licensee's response to RAI 4.1.

Table 3: Susquehanna Licensed Operating Domains

Licensed Domain	Disposition
Two-Loop (Normal) Operation	Explicitly analyzed two statepoints that correspond to the maximum licensed power level.
Single-Loop Operation	Explicitly analyzed statepoint corresponding to limiting power and flow conditions during single-loop operation.
MELLLA	Explicitly analyzed the **[[** **]]** at rated thermal power.
[[**]]**	Licensee qualitatively dispositioned this operating condition, stating that **[[** **]]**.

The NRC staff found the licensee's response to the RAI acceptable because it identified the existing set of licensed operating domains and provided an appropriate basis in each case for concluding that the limiting figures of merit calculated in its LOCA analysis bound all licensed operating conditions.

3.7.3.3 Exposure-Dependent LOCA Analysis Implementation

The NRC staff's review found the exposure study analysis described in ANP-3784P generally conforms to the approved evaluation model documented in ANP-10332P [34]. As shown in Table 9.1 of ANP-3784P, the exposure study considered **[[** **]]**. In particular, the exposure study analyzed **[[** **]]** accounting for exposure-dependent limiting values of the linear heat generation rate and maximum average planar linear heat generation rate.

The exposure study for Susquehanna described in ANP-3784P deviated from the methodology approved in the NRC staff's SE on ANP-10332P in that at **[[** **]]** approved by the NRC staff's SE. However, because these **[[** **]]** do not appear to produce limiting results in the analysis under review, the NRC staff found this deviation from the approved evaluation model acceptable for the Susquehanna LOCA analysis described in ANP-3784P.

3.7.4 <u>Calculated Results</u>

3.7.4.1 Break Spectrum Results

The break spectrum analysis results were reported in Tables 6.2 and 7.1 of ANP-3784P. The following tables summarize the peak cladding temperature (PCT). The limiting case (i.e., the case with the highest PCT **[[** **]]**) is the 0.07 ft^2 break in

[3] The provision of this information is necessary to satisfy Limitation and Condition 16 from the NRC staff's final SE on ANP-10332P.

the recirculation system discharge piping with a single failure of SF-BATT and a top-peaked axial power shape when operating at 102 percent rated power and [[ ]]. In contrast, the limiting case for single-loop operation is the 0.09 ft^2 break in the pump discharge piping with a single failure of SF-BATT and a top-peaked axial power shape when operating at [[ ]] Note that similar to the two-loop operation analysis, [[ ]]. The results presented for single-loop operation in Table 4 is the limiting case (i.e., highest PCT case for single-loop operation).

Table 4: ANP-3784 Tables 6.2 and 7.1
Summary of Break Spectrum Analysis Results for Two-Loop Operation and Single-Loop Operation Recirculation Line Breaks

[[

During the regulatory audit, the NRC staff reviewed the break spectrum analysis calculation book and confirmed that the above limiting cases for both two-loop operation and single-loop operation were identified correctly among the [[ ]]. The NRC staff also found that the [[ ]]. Based on the audit findings, the NRC staff found that the licensee's break spectrum evaluation is acceptable because the break spectrum results are consistent with both (1) the expected physical behavior for the LOCA event at a BWR and (2) the procedure for break spectrum analysis in the approved AURORA-B LOCA evaluation model described in ANP-10332P.

Ol=l=ICIAb USE! ONbY PROPRleTARY INl=ORMATION

3.7.4.2 Exposure-Dependent LOCA Analysis generation rate (MAPLHGR) limit and [[11 presented in ANP-3784 are applicable to Susquehanna, as described in ANl-'-::S/l::S4, the licensee performed an exposure-dependent LOCA analysis. The licensee's exposure study for this limiting scenario predicted the figures of merit as shown below.

Table5: PredictedFiguresofMeritforSusquehanna Exposure-Dependent LOCA Analysis

11

To ensure that the ATRIUM 11 exposure-dependent maximum average planar linear heat

Figure of Merit

Peak Cladding Temperature

Maximum (Local) Cladding

Oxidation

Maximum (Core-Wide) Hydrogen

Generation

[[

Limiting

Exposure

Predicted Value

1,784 °F

4.64%

< 0.30%

Acceptance

Criterion

□2,200 °F

□17%

□1%

The NRC staff identified that there is a difference for the limiting PCT results between [[

11. The NRC In the response to RAI 4.2, the licensee stated that the break spectrum calculations were

performed [[

)) I he NKC starr round tne response acceptable because tne licensee provided tne

requested information and explained the difference in the limiting PCT due to [[

]].

In RAI 4.3, the NRC staff requested that the licensee explain the "abrupt" change of local

starr issued KAI 4.2 to resolve this discrepancy.

cladding oxidation from assembly average planar exposure [[

Jin I able l:l.1 or ANl-'-::S/l::S41-'.

I ne licensee responded tnat the abrupt cnange in local oxidation is due to [[

]] The NKC starr round the response acceptable because the requested mrormat1on nad been provided

and confirmed.

The NRC staff identified the following additional information to be requested in RAI 4.4:

- The process for determining the LHGR used for both U02 and Gd203-U02 pellets during exposure-dependent analysis in the AURORA-8 LOCA analysis - specifically, the LHGR limit curves presented in Figures 2.2 and 2.3 as shown in ANP-3784P [[

- Demonstration of the analysis margin for the MAPLHGR limit in Figure 2.1 of ANP-3784P **[[** **]]**.

The licensee responded for RAI 4.4.a as:
The LHGR limit curves presented in Figures 2.2 and 2.3 from the exposure

]]

analysis **[[**

The licensee responded for RAI 4.4.b as: Figure 4-3 shows the **[[**

]

]]

The NRC staff found the responses acceptable because the requested information had been provided and confirmed as reasonable from the figures provided in the RAI response.

3.7.5 Conformance with Limitations and Conditions

The licensee provided information in Appendix A of ANP-3784 on how it satisfies all limitations and conditions from the NRC staff's SE on ANP-10332P. The licensee's proposed disposition of limitations and conditions is in conformance to the regulatory position imposed therein. However, in certain instances, as discussed below, the NRC staff found more detailed review necessary to confirm that the licensee had appropriately addressed the applicable limitations and conditions.

Regarding Limitation and Condition 14, the NRC staff confirmed from the licensee during an audit [33] that the figures of merit for Susquehanna in the **[[**

]] had been determined and provided in Section 2.0 and its footnote of ANP-3784. Therefore, the NRC staff finds that the licensee adequately addressed this limitation and condition.

Regarding Limitation and Condition 19, the licensee dispositioned that the **[[**
] for mixed cores. For the first cycle of applying ATRIUM 11 fuel, the core for

Susquehanna will be a mixed core of ATRIUM 10 and ATRIUM 11 fuels. The NRC staff notes

that the LOCA analysis results presented in the current UFSAR are based on an equilibrium core of ATRIUM 10 fuel. A comparison of current UFSAR LOCA analysis results (UFSAR Table 6.3-3C) with ANP-3784, Table 6.2 in **[[**

]] for the same 102 percent power**[[**
]] and axial power peaked. Although the licensee analyzed and demonstrated that the

legacy ATRIUM 10 fuel would be **[[ ]]** with ATRIUM 11 fuel, the cause of the **[[ ]]** must be identified. The NRC staff confirmed the LOCA results presented by the licensee and concluded that the AURORA-B LOCA evaluation model described in ANP-10332P applied to the LOCA analysis for the **[[**

]] Both evaluation models (i.e., the current UFSAR LOCA analysis model and the AURORA-B LOCA model) are

NRC-approved models and methodologies; the AURORA-B LOCA evaluation model described in ANP-10332P will be the analysis of record for ATRIUM 11 fuel and the

EXEM BWR-2000 [35] analysis of record will remain in place for ATRIUM 10 fuel after this LAR is approved. Based on the above, the NRC staff finds that the licensee adequately addressed this limitation and condition.

3.7.6 Conclusion for LOCA Analysis with ATRIUM 11 Fuel

The NRC staff reviewed the information in the licensee's submittals pertaining to the analysis of the spectrum of postulated LOCA events for Susquehanna, including the submittal as well as relevant responses to RAIs. The NRC staff's review was further supported by a regulatory audit [32], which was used to confirm information referred to in docketed submittals. The NRC staff concludes that the LOCA analysis with ATRIUM 11 fuel to be used in Susquehanna, Units 1 and 2, is acceptable because it complies with the relevant requirements of 10 CFR 50.46, Appendix K to 10 CFR Part 50, and GDC 35. This conclusion is based on the following:

1. The licensee performed analyses of the performance of the ECCS with ATRIUM 11 fuel in accordance with 10 CFR 50.46. The analyses considered a spectrum of postulated break sizes and locations and were performed with an evaluation model that follows Appendix K to 10 CFR Part 50 and meets the requirements of 10 CFR 50.46. The results of the analyses (Sections 3.7.1.1 and 3.7.4 of this SE) show that the ECCS with ATRIUM 11 fuel satisfies the 10 CFR 50.46 criteria.
2. The evaluation meets the requirements of GDC 35 with respect to abundant emergency core cooling being provided that will transfer heat from the reactor core filled with ATRIUM 11 fuel in the event of a LOCA, and the suitable redundancy of components and features being provided so that the safety function can be accomplished assuming a single failure by:

a. Demonstrating with the LOCA analysis performed for ATRIUM 11 to be used in Susquehanna that abundant emergency core cooling is provided to transfer heat from the reactor core filled with ATRIUM 11 fuel in the event of a LOCA and showing that suitable redundancy of components and features is provided so that the safety function can be accomplished assuming a single failure, irrespective of whether its electrical power is supplied from offsite or onsite sources (Sections 3.7.3 and 3.7.4 of this SE).

b. Applying the NRC-approved LOCA evaluation model and methodology for the LOCA analysis with ATRIUM 11 fuel and adequately meeting the limitations and conditions listed in the NRC staff's SE for the applied topical reports (Sections 3.7.2 and 3.7.5 of this SE).

3.8 Control Rod Drop Accident (CRDA) 3.8.1 Regulatory Basis

GDC 13 and 28 and 10 CFR 50.67 are pertinent to the analysis of CRDA events. GDC 13 primarily applies to the CRDA event by ensuring that the limiting system operating parameters and other controls in place (i.e., rod withdrawal limitations) are sufficient to ensure that the CRDA acceptance criteria are not exceeded. This is satisfied by ensuring that the initial conditions represented in the CRDA analyses are sufficiently representative of the most conservative condition allowed by the aforementioned controls. In addition, Susquehanna is licensed under 10 CFR 50.67 to establish radiation dose limits for individuals at the boundary of the exclusion area and at the outer boundary of the low population zone.

The acceptance criteria for CRDA events to satisfy GDC 28 and 10 CFR 50.67 are currently defined in Chapter 15 of the SRP. Along with Chapter 15, SRP Section 4.2 provides an extensive discussion of acceptance criteria related to high temperature cladding failure, pellet clad mechanical interaction induced cladding failure, core coolability, and fission product inventory determination for dose assessment purposes. Regulatory Guides 1.183 and 1.195 are also referenced for further guidance related to fission product inventories.

However, the NRC staff is currently developing new guidance for rod injection accident acceptance criteria that will supersede SRP Section 4.2. The draft guidance document – draft guide (DG)-1327, has not become a final regulatory guide. The licensee indicated that it intends to adopt the DG-1327 criteria for use in analysis of the CRDA event. The NRC staff does not expect the specified acceptance criteria to change significantly, and the technical basis for use of the DG-1327 criteria is more robustly supported by recent research than the CRDA acceptance criteria that Susquehanna is currently licensed under. Therefore, the NRC staff considered the DG-1327 criteria at Susquehanna in lieu of SRP Section 4.2.

3.8.2 Technical Evaluation

In ANP-3771P (Enclosure 16a to the LAR the licensee provided information and some sample calculations demonstrating how the CRDA analysis methodology described in ANP-10333P-A [36] will be applied at Susquehanna to evaluate each cycle. The sample calculations were based on the equilibrium core design, but cycle-specific calculations will be performed to support each reload. A comparison of the information provided by the licensee against ANP-10333P-A shows that the licensee demonstrated an acceptable application of the methodology to evaluate the CRDA event for the Susquehanna equilibrium core design, with a few plant-specific nuances as discussed below. The licensee also provided information that allowed the NRC staff to confirm that all the limitations and conditions for ANP-10333P-A were met for the Susquehanna application.

In addition to finding that the information provided by the licensee shows that it will correctly apply the CRDA analysis methodology at Susquehanna, the NRC staff makes the following additional findings and observations specific to Susquehanna:

- • In Section 6.4 of ANP-3753P, the CHF correlation used for the CRDA calculations is discussed. The range of applicability for the fuel-specific CHF correlations for ATRIUM 11 does not extend to the cold startup conditions that the CRDA analyses are performed at. Instead, the licensee used the **[[**

  ]] correlation to be acceptable for use for this purpose.

- • The CRDA demonstration calculations utilize the fuel rod failure criteria from DG-1327, which has not yet completed the process of becoming a final regulatory guide. However, the NRC review and approval of ANP-10333P-A indicates that the methodology is acceptable for use with either the current CRDA acceptance criteria in Appendix B to SRP Section 4.2 or the new proposed criteria in DG-1327. Furthermore, the NRC has published the technical and regulatory basis for the new acceptance criteria in the “Technical and

Regulatory Basis for the Reactivity-Initiated Accident Acceptance Criteria and Guidance, Revision 1," dated March 16, 2015 (ADAMS Accession

No. ML14188C423) [37]. A review of this information indicates that sufficient evidence exists to support the use of the fuel failure threshold curves from DG-1327; therefore, the NRC staff finds the proposal to use the DG-1327 guidance in the manner described in ANP-3771P to be acceptable. The NRC staff also notes that the ATRIUM 11 fuel to be loaded at Susquehanna utilizes stress relief annealed unlined cladding, similar to the current ATRIUM 10 fuel at Susquehanna.

- • Appendix A to ANP-3771P describes the process used to establish an evaluation boundary curve to simplify the calculations. This process was approved as part of the ANP-10333P-A methodology, with a limitation and condition requiring the licensee to confirm the applicability of the curve to several local characteristics that may be present in the core being analyzed. This information was presented for the equilibrium core, but the licensee will need to confirm that the evaluation boundary curve is also applicable to ATRIUM 10 fuel prior to use in analysis of the transition cores.

  The licensee applied NRC-approved analytical methods to perform a demonstration CRDA analysis. The acceptance criteria are derived from the topical report for the approved CRDA analysis method. The licensee showed how it would determine whether fuel failures would occur and considered an artificial scenario where fuel failures occur so it could show how the radiological consequences would be evaluated. All calculations and evaluations were performed in a manner consistent with the basis for the NRC approval of the methods and demonstrated how they would determine whether acceptance criteria are met.

3.9

1. 2.

The licensee has proposed to implement the AURORA-B CRDA evaluation model in an acceptable manner and

Compliance with the applicable regulatory requirements has been demonstrated.

Revision of Low-Pressure Safety Limit in TSs 2.1.1.1 and 2.1.1.2

Based on the above, the NRC staff concludes that the proposed adoption of the CRDA analysis methods as part of the planned transition to ATRIUM 11 fuel is acceptable.

3.8.3 Conclusion

Pertaining to CRDA, the NRC staff reviewed the information in the Susquehanna submittals pertaining to the analysis of Susquehanna events, including the original submittal as well as relevant responses to RAIs [2], [3]. Based upon its review as summarized above, the NRC staff has concluded that:

TSs 2.1.1.1 and 2.1.1.2 ensure that the critical power correlation is only evaluated within the approved range of applicability. The ACE/ATRIUM 11 correlation that will be used for the ATRIUM 11 fuel at Susquehanna [17] is valid at pressures of at least 575 psig to ensure that it results in valid calculated CPR values. Therefore, the licensee proposes to increase the low-pressure safety limit from 557 psig to 575 psig. The proposed new limit conservatively bounds existing application of the Siemens Power Corporation B (SPCB) correlation used for the ATRIUM 10 fuel. Accordingly, the proposed change to TSs 2.1.1.1 and 2.1.1.2 continues to ensure that a valid CPR calculation is performed for AOOs at Susquehanna and, therefore, the NRC staff finds it acceptable.

3.10 Removal of Neutronic Methods Penalties 3.10.1 OPRM Amplitude Setpoint

The current Susquehanna operating licenses include a license condition on the OPRM setpoint determination. The OPRM amplitude setpoint penalty is applied to account for a reduction in thermal neutrons around the low-power range monitor detectors caused by transients that increase voiding, ultimately reducing the OPRM scram setpoint. This license condition was created before an in-depth review of this issue was fully evaluated by the NRC staff in the RAMONA5-FA licensing topical report [38]. The NRC staff's review of the approved RAMONA5-FA methodology [39] concluded **[[**

]] Therefore, the NRC staff finds it acceptable to remove the OPRM amplitude setpoint penalty applied through

this license condition.

3.10.2 Pin Power Distribution Uncertainty and Bundle Power Correlation Coefficient

The current Susquehanna operating licenses include a license condition for a penalty on SLMCPR pin power distribution uncertainty and bundle power correlation coefficient. No significant change in the uncertainty of the predicted detector response relative to the

measurements is anticipated for the transition to ATRIUM 11 fuel. The NRC staff's review of the AURORA-B methodology concluded that since the analysis and core monitoring at Susquehanna is based upon the CASMO-4/MICROBURN-B2 methodology, there is no need for any uncertainty penalties when using AURORA-B methods, and the use of the **[[ ]]** correlation for ATRIUM 11 fuel is justified. In addition, since Susquehanna is currently operating within approved extended power uprate conditions (and not in extended flow windows), operating conditions are within previously approved power/flow ratios. Therefore, the NRC staff finds it acceptable to remove the pin power distribution uncertainty and bundle power correlation coefficient penalty applied through this license condition.

3.11 Technical Evaluation Conclusions

The NRC staff reviewed the licensee's analyses related to the effect of the proposed amendments for Susquehanna to allow application of the Framatome analysis methodologies necessary to support a planned transition to ATRIUM 11 fuel under the currently licensed MELLLA operating domain under extended power uprate conditions. The NRC staff further reviewed the licensee's proposed changes to TS 5.6.5.b that support adoption of the intended Framatome analysis methodologies, to TSs 2.1.1.1 and 2.1.1.2 to revise the low-pressure safety limit, and to license conditions to remove neutronic methods penalties on OPRM amplitude setpoint and the pin power distribution uncertainty and bundle power correlation coefficient. Based on its review, as summarized in this SE, the NRC staff concludes that the proposed amendments for Susquehanna are acceptable.

3.12 Vessels and Internals Branch Evaluation of Aging Degradation of Vessel Internals

The NRC staff determined that the assessment of aging degradation due to irradiation embrittlement in RPV base metal and welds is determined by the evaluation of pressure--temperature (P-T) limits, evaluation of upper shelf energy (USE) of the RPV beltline base metals and welds, and the evaluation of adjusted reference temperature (ART) for the beltline base metals and welds. Higher ART values and lower USE values indicate that RPV base metals and welds are embrittled. An increase in transition temperature of the RPV materials due to exposure to neutron fluence results in an increase in embrittlement, and this is reflected in higher operating temperature of the vessel for given operating pressure.

For the reactor vessel internals (RVI) components, boiling-water reactor units now examine the RPV interior surfaces, attachments, and core support structures in accordance with BWR vessel internals inspection program (BWRVIP) and evaluation guidelines. Operating experience to date indicates that aging degradation would be active when the accumulated neutron fluence exceeds threshold limits applicable to each of the following aging degradation mechanisms in the RVI components: (1) inspection and evaluation; (2) irradiation assisted stress corrosion cracking; (3) irradiation stress relaxation; and (4) irradiation embrittlement.

In its February 6, 2020, letter, the licensee submitted information on the aging degradation of the RPV and RVI components due to the implementation of ATRIUM 11 fuel at Susquehanna, Units 1 and 2. The licensee stated that one of the benefits of the ATRIUM 11 fuel design is that smaller reload batch sizes will be required. To successfully design a core with smaller reload batch size, a greater number of older bundles are moved to the periphery core locations. Due to their higher exposures, the older bundles will have lower fission power and, therefore, generate fewer fast neutrons at the core periphery when compared to an ATRIUM 10 core.

Furthermore, the licensee stated that based on this general understanding of the core physics, the expectation is that the neutron fluence at the RPV wall, and also for beltline components located within the RPV (e.g., core shroud, jet pump components), will decrease. The licensee stated that an analysis of the fast neutron fluence in the RPV plates, welds, and nozzles throughout the beltline region, determined at 60 years, has been completed using the

NRC -approved RAMA fluence methodology. The fast neutron fluence was determined in accordance with the guidelines and requirements presented in RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."

In addition, the licensee stated that a review of the analysis results determined that the fast neutron fluence levels at the reactor vessel wall throughout the beltline region with ATRIUM 11 fuel is lower than with ATRIUM 10 fuel when analyzed out to 60 years. Similarly, an analysis was performed for RPV beltline internal components (e.g., core shroud, jet pump components, top guide, core plate) using the same methods as described above. These results are consistent with the fundamental understanding of the core physics for ATRIUM 11 fuel. Based on the above discussion, the change from ATRIUM 10 to ATRIUM 11 fuel results in a lower fast neutron fluence for both the RPV and RVI components located within the RPV. Based on this result, there is no effect on the aging degradation due to the transition to ATRIUM 11 fuel at Susquehanna, Units 1 and 2, in the current licensing period.

The implementation of ATRIUM 11 fuel design requires smaller reload batch sizes. Designing a core with smaller reload batch sizes results in a greater number of older bundles being moved to the periphery core locations. Due to their higher exposures, the older bundles will have lower fission power and, therefore, generate fewer fast neutrons at the core periphery when compared to an ATRIUM 10 core. This fuel arrangement falls under the "low leakage" category, which indicates that outer periphery in the beltline region will be exposed to lower fast neutron fluence (Energy level > 1 MeV) in comparison to the locations near the core region. Therefore, the NRC staff concludes that RPV and RVI components would be exposed to lower neutron fluence values with high energy levels (i.e., > 1 MeV) than the previous period of operations with ATRIUM 10 fuel.

Some of the RPV and RVI components in the beltline region may not have been exposed to neutron fluence values exceeding threshold limits for the onset of aging degradation mechanisms. In this case, there would be a delay in the onset of any aging degradation due to exposure to lower neutron fluence radiation associated with "low leakage" fuel arrangement with older bundles. The affected components are (1) RPV beltline base metals and welds (irradiation embrittlement is only active degradation mechanism) and (2) RVI beltline base metals and welds in core shroud, top guide, core plate, and jet pump components. For components that were already exposed to neutron fluence threshold limits, the damage associated with the aging degradation would be reduced. This is due to exposure to lower fluence values related to a "low leakage" fuel arrangement with older fuel bundles to the core periphery locations.

Based on the above, the NRC staff determined that the evaluation of inspection and evaluation of the RPV base metals and welds, which includes development of P-T limits, the evaluation of USE of the RPV beltline welds, and the evaluation of ART for the beltline base metals and welds will remain valid for the current licensing period at Susquehanna. Based on the plant operations, the neutron fluence values would increase over time, which requires reevaluation of the P-T limits, USE, and ART values. Accordingly, the licensee is expected to update these values at that time. Current aging management programs for the RVI components include

implementation of NRC-staff approved BWRVIP inspection and evaluation guidelines at Susquehanna, Units 1 and 2. This program evaluates aging effects in components (i.e., core shroud, top guide, core plate, and jet pumps) and it remains valid for the current licensing period at Susquehanna. Therefore, the staff finds that the licensee's implementation of ATRIUM 11 fuel at Susquehanna is acceptable.

3.12.1 Conclusion Regarding Aging Degradation

The NRC staff determined that implementation of ATRIUM 11 fuel at Susquehanna, Units 1 and 2, results in exposure to lower neutron fluence values on the RPV and RVI components due to "low leakage" fuel arrangement. Accordingly, the staff determined that the current evaluation for the RPV base metals and welds and current aging monitoring program for the RVI components remain valid for Susquehanna during the current licensing period and, therefore, concludes that the licensee's implementation of ATRIUM 11 fuel at Susquehanna is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the NRC's proposed issuance of the amendments on June 16, 2020. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding
(84 FR 56482; October 22, 2019). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 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.

7.0 REFERENCES

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23. [23] *ANP-10307PA Revision 0, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," AREVA NP, June 2011.*
24. [24] Framatome, Inc., ANP-10300NP-A, Rev. 1, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios," dated January 31, 2018 (ADAMS Accession No. ML18186A181).
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26. [26] Siemens Power Corporation, Publication of EMF-2158(P)(A) Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2," dated March 27, 2000 (ADAMS Accession No. ML003698556).
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28. [28] *Duke Energy, Brunswick, Units 1 and 2, "Response to Request for Additional Information Regarding Advanced Framatome Methodologies License Amendment Request," dated June 18, 2019 (ADAMS Accession No. ML19169A032).*
29. [29] Talen Energy, "Susquehanna Steam Electric Station Units 1 and 2, Updated Final Safety Analysis Report, Revision 68, dated November 28, 2017, ADAMS Accession No. ML17331A584".
30. [30] *GE Nuclear Energy, NEDO-32047-A, "ATWS Rule Issues Relative to BWR Core Thermal- Hydraulic Stability," June 1995, ADAMS Accession No. ML102230093.*
31. [31] NRC, "Final Safety Evaluation by the Office of Nuclear Reactor Regulation, Topical Report ANP-10332P, Revision 0, "AURORA-B: An Evaluation Model for

Boiling Water Reactors: Application to Loss of Coolant Accident Scenarios," March 2019.

32. [32] "U.S. NRC, Audit Report, “Audit Report for Susquehanna Steam Electric Station, Units 1 and 2 to Support Review of the License Amendment Request Regarding Application of Framatome Methodologies for Transition to ATRIUM 11 Fuel (ML20115E396)".
33. [33] General Electric, "NEDO-20566A, General Electric Company Analytical Model for Loss of Coolant Analysis in Accordance with 10 CFR 50 Appendix K," September 1986.
34. [34] Framatome, "ANP-10332P-A, AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios," March 2019.
35. [35] F. A. Inc., "Framatome ANP Inc., "EMF-2361 (P)(A) Revision 0, EXEM BWR-2000 ECCS Evaluation Model," Dated May 31, 2001 (ADAMS Accession No. ML012050396)".
36. [36] *Framatome Inc. Report ANP-10333P-A, “AURORA-B: An Evaluation Model for Boiling Water Reactors, Application to Control Rod Drop Accident (CRDA),” March 2018. (ML18208A447 (Non-Proprietary) / ML18208A448 (Proprietary)).*
37. [37] *U.S. NRC, “Technical and Regulatory Basis for the Reactivity-Initiated Accident Acceptance Criteria and Guidance, Revision 1,” dated March 16, 2015. (ADAMS Accession No. ML14188C423).*
38. [38] *AREVA NP Inc. Licensing Topical Report EMF 3028P-A, Volume 2, Revision 4, “RAMONA5-FA: A Computer Program for BWR Transient Analysis in the Time Domain: Theory Manual” (ADAMS Accession No. ML131550602 (Proprietary)).*
39. [39] *Safety Evaluation of EMF-3028P-A Volume 2 Revision 4, “RAMONA5-FA: A Computer Program for BWR Transient Analysis in the Time Domain Volume 2: Theory Manual,” March 2013, ML12318A206.*
40. [40] *Talen Energy, Inc. "Susquehanna, Submittal of Revised Topical Report to Support License Amendment Request Application of Advanced Framatome Methodologies", March 9, 2020 ADAMS Accession No ML20069D135.*

Principal Contributors: Ashley Smith Shie-Jeng Peng

Ganesh Cheruvenki

Date: January 21, 2021

SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 278 AND 260 TO ALLOW APPLICATION OF ADVANCED FRAMATOME ATRIUM 11 FUEL METHODOLOGIES
(EPID L-2019-LLA-0153) DATED JANUARY 21, 2021

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DATE 05/08/2020 OFFICE OGC – NLO NAME JWachutka DATE 07/20/2020

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February 11, 2021 – Letter from Sujata Goetz, Project Manager Plant Licensing Branch I Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to Susquehanna Nuclear, LLC with the subject of SUMMARY OF THE JANUARY 26, 2021, MEETING WITH SUSQUEHANNA NUCLEAR REGARDING A FUTURE LICENSE AMENDMENT REQUEST RELATED TO ADOPTING TECHNICAL SPECIFICATIONS TASK FORCE TRAVELER TSTF-505, REVISION 2 (EPID L-2020-LRM-0118)

On January 26, 2021, a Category 1 public teleconference meeting was held between the U.S. Nuclear Regulatory Commission (NRC) staff and the representatives of Susquehanna Nuclear (the licensee) using video conferencing capabilities. The purpose of the meeting was to discuss a future license amendment request (LAR) for Susquehanna Steam Electric Station Units 1 and 2 (Susquehanna) to adopt Technical Specifications Task Force (TSTF) Traveler TSTF-505, “Provide Risk-Informed Extended Completion Times – RITSTF [Risk-Informed TSTF] Initiative 4b,” Revision 2 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML18183A493). The meeting notice dated January 5, 2021, is available in ADAMS at Accession No. ML21005A161, and the meeting slides provided by Susquehanna Nuclear are available at ADAMS Accession No. ML21025A175. A list of attendees is provided as an enclosure to this letter.

The licensee expects to submit the proposed LAR for Susquehanna, Units 1 and 2, by the end of March 2021.

The proposed amendment would revise technical specification (TS) requirements to permit the use of risk-informed completion times (RICTs) for 22 limiting condition of operations (LCOs) for both units. The RICTs would be applicable in Modes 1 and 2. The licensee is not proposing to apply RICTs to loss of function actions and is proposing the following variations from TSTF-505, Revision 2, for Susquehanna:

- • Propose to include some plant-specific Required Actions that are not in TSTF-505.
- • Propose to include TS 3.3.2.1 in the RICT program, which has been generically

  excluded from TSTF-505.

- • TSTF-505 has some Conditions and Required Actions that are in standard technical specifications, but are not applicable to Susquehanna.

  Regarding probabilistic risk assessment (PRA), the licensee stated that:

• Susquehanna plans to use internal events, internal fire, and internal flooding models.

- • Susquehanna does not have a seismic PRA model and a seismic penalty will be applied to all RICTs. Additional large early release frequency will be applied when containment is de-inerted.
- • Other external hazards have been screened out.
- • Software being used will be similar to the one use for Maintenance rule (a)(4)

  configuration risk management program.

  The NRC staff asked the licensee about the revision of Regulatory Guide (RG) 1.200, “An Approach for Determining The Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” that would be supporting the application. The licensee stated that Revision 2 of RG 1.200, dated March 2009 (ADAMS Accession No. ML090410014), will be followed.

  The licensee stated during the meeting that mitigating strategies (FLEX) were credited in the PRA. The NRC staff stated that the depth and level-of-detail of the staff’s review of FLEX modeling is commensurate with its impact on the RICT calculations. The NRC staff cited Question 05 (“Probabilistic Risk Assessment Modeling and Uncertainty of FLEX Equipment and Actions”) in the Peach Bottom Supplement to License Amendment Request, dated

  December 2, 2020 (ADAMS Accession No. ML20337A301), as a representative example of the type of information requested as part of the staff’s review of FLEX modeling.

  The licensee stated during the meeting that the PRA was peer reviewed in accordance with

RG 1.200, Revision 2, and that a number of facts and observations (F&Os) were identified. The licensee further stated that 13 F&Os remain open after an F&O closure review was performed in accordance with Appendix X to Nuclear Energy Institute (NEI) 05-04/07-12/12-[13], "Close-Out of Facts and Observations (F&Os)." It was not clear to the NRC staff when this F&O closure review was conducted and whether it used the final revision of Appendix X dated February 21, 2017 (ADAMS Accession No. ML17086A431), as accepted by NRC in its letter dated May 3, 2017 (ADAMS Accession No. ML17079A427), which specifies conditions for accepting Appendix X. The NRC staff noted that F&O closure reviews conducted by other licensees before May 2017 typically used draft Appendix X guidance that had gaps relative to the final guidance as accepted by NRC. The NRC staff provided feedback that the LAR should indicate whether the F&O closure review is consistent with Appendix X, as accepted (with conditions) by NRC's letter dated May 3, 2017. If not consistent, the LAR should include one of the following to address the inconsistency: (1) describe the results of a gap assessment between the F&O closure review and Appendix X, as accepted, and how any gaps were resolved (which may require reconvening the independent assessment team, as applicable), or (2) disposition the F&Os that were closed during the F&O closure review.

No one identified themselves as being members of the public during the meeting. Public meeting feedback forms were not received.

Please direct any inquiries to me at 301-415-8004 or Sujata.Goetz@nrc.gov.

February 18, 2021 – Letter from Sujata Goetz, Project Manager Plant Licensing Branch I Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to Kevin Cimorelli Site Vice President Susquehanna Nuclear, LLC with subject of SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 279 AND 261 RE: REVISE TECHNICAL SPECIFICATIONS TO ADOPT TSTF-582, "RPV WIC ENHANCEMENTS" (EPID L-2020-LLA-0197)

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 279 to Renewed Facility Operating License No. NPF-14 and Amendment No. 261 to Renewed Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Units 1 and 2, respectively. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated September 1, 2020.

Susquehanna Nuclear, LLC (the licensee) requested that the NRC process the proposed amendment under the Consolidated Line Item Improvement Process. The proposed changes revise the TSs related to reactor pressure vessel (RPV) water inventory control (WIC) based on Technical Specifications Task Force (TSTF) Traveler TSTF-582, Revision 0, "RPV WIC Enhancements," and the associated NRC staff safety evaluation of TSTF-582.

A copy of the related safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's monthly *Federal Register* notice.

SUSQUEHANNA NUCLEAR, LLC
ALLEGHENY ELECTRIC COOPERATIVE, INC. DOCKET NO. 50-387
SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1 AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 279 Renewed License No. NPF-14

1. The U.S. Nuclear Regulatory Commission (NRC or the Commission) has found that:

1. The application for the amendment filed by Susquehanna Nuclear, LLC, dated September 1, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 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.

Enclosure 1

Attachment:
Changes to the Renewed Facility

Operating License and Technical Specifications

Date of Issuance: February 18, 2021

-2-

2. 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. NPF-14 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 279, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. Susquehanna Nuclear, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation

ATTACHMENT TO LICENSE AMENDMENT NO. 279 SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1 RENEWED FACILITY OPERATING LICENSE NO. NPF-14 DOCKET NO. 50-387

Replace the following page of the Renewed Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

REMOVE INSERT Page 3 Page 3

Replace the following pages of the Appendix A, Technical Specifications, with the attached revised pages. The revised pages are identified by amendment number and contain a marginal line indicating the areas of change.

REMOVE INSERT 1.1-3 1.1-3 3.3-44 3.3-44

1. 3.3-47a 3.3-47a
2. 3.3-47b 3.3-47b
3. 3.3-47c 3.3-47c

71. 3.3-71 3.3-71
72. 3.3-72 3.3-72

3.5-8 3.5-8 3.5-8a 3.5-8a 3.5-9 3.5-9 3.5-9a --

10. 3.5-10 3.5-10
11. 3.5-11 3.5-11

3.8-19 3.8-19

3. (3) Susquehanna Nuclear, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, posses, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed neutron sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;
4. (4) Susquehanna Nuclear, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, posses, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and
5. (5) Susquehanna Nuclear, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission nor or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

1. (1) Maximum Power Level

   Susquehanna Nuclear, LLC is authorized to operate the facility at reactor core power levels not in excess of 3952 megawatts thermal in accordance with the conditions specified herein. The preoperational tests, startup tests and other items identified in License Conditions 2.C.(36), 2.C.(37), 2.C.(38), and 2.C.(39) to this license shall be completed as specified.

2. (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 279, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. Susquehanna Nuclear, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

   For Surveillance Requirements (SRs) that are new in Amendment 178 to Facility Operating License No. NPF-14, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 178. For SRs that existed prior to Amendment 178, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of

   Amendment 178.

Renewed Operating License No. NPF-14

Amendment No. 279

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)

Definitions 1.1

Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," EPA, 1988, as described in Regulatory Guide 1.183. The factors in the column headed "effective" yield doses corresponding to the CEDE. The conversion factors that are used for the calculation of EDE (or DDE) from external exposure (submersion) shall be those listed in Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil," EPA, 1993, as described in Regulatory Guide 1.183. The factors in the column headed "effective" yield doses corresponding to the EDE.

The DRAIN TIME is the time it would take for the water inventory in and above the Reactor Pressure Vessel (RPV) to drain to the top of the active fuel (TAF) seated in the RPV assuming:

DRAIN TIME

a) b)

The water inventory above the TAF is divided by the limiting drain rate;

The limiting drain rate is the larger of the drain rate through a single penetration flow path with the highest flow rate, or the sum of the drain rates through multiple penetration flow paths susceptible to a common mode failure for all penetration flow paths below the TAF except:

1. Penetration flow paths connected to an intact closed system, or isolated by manual or automatic valves that are closed and administratively controlled in the closed position, blank flanges, or other devices that prevent flow of reactor coolant through the penetration flow paths;
2. Penetration flow paths capable of being isolated by valves that will close automatically without offsite power prior to the RPV water level being equal to the TAF when actuated by RPV water level isolation instrumentation; or

3. Penetration flow paths with isolation devices that can be closed prior to the RPV water level being equal to the TAF by a dedicated operator trained in the task, who in continuous communication with the control room, is stationed at the controls, and is capable of closing the penetration flow path isolation device without offsite power.

FUNCTION

2. LPCI System (continued)

f. Manual Initiation

3. High Pressure Coolant Injection (HPCI)

System

1. Reactor Vessel Water Level- Low Low, Level 2
2. Drywell Pressure- High
3. Reactor Vessel Water Level- High, Level 8

APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS

1,2,3

1, 2(d), 3(d)

1, 2(d),3(d)

1, 2(d), 3(d)

REQUIRED CHANNELS PER FUNCTION

2
1 per subsystem

4

4

2

CONDITIONS REFERENCED FROM REQUIRED ACTION A.1

C

B

B C D

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1.5

SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5

SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5

SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5

SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5

ALLOWABLE VALUE

NA

≥ -45 inches

≤ 1.88 psig
≤ 55.5 inches

≥ 40.5 inches above tank bottom

d. Condensate 1, 2 Storage Tank 2(d), 3(d)
Level-Low

(d) With reactor steam dome pressure > 150 psig.

3.3 INSTRUMENTATION
3.3.5.2 Reactor Pressure Vessel (RPV) Water Inventory Control Instrumentation

LCO 3.3.5.2 The RPV Water Inventory Control instrumentation for each Function in Table 3.3.5.2-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.5.2-1.

ACTIONS --NOTE----------------------------------
----------------------- Separate Condition entry is allowed for each channel. ---------------------
-- --

CONDITION

A. Oneormorechannels inoperable.

REQUIRED ACTION

COMPLETION TIME Immediately

Immediately

Immediately

A.1 Initiate action to place channel in trip.

OR

1. A.2.1 Declare associated penetration flow path(s)

   incapable of automatic isolation.

   AND

2. A.2.2 Initiate action to calculate DRAIN TIME.

SURVEILLANCE REQUIREMENTS --
NOTE-- These SRs apply to each Function in Table 3.3.5.2-1. --
----

SR 3.3.5.2.1

SR 3.3.5.2.2

SURVEILLANCE Perform CHANNEL CHECK.

------------------------------NOTE------------------------------- A test of all required contacts does not have to be performed. --

Perform CHANNEL FUNCTIONAL TEST.

FREQUENCY

In accordance with the Surveillance Frequency Control Program

In accordance with the Surveillance Frequency Control Program

Table 3.3.5.2-1 (page 1 of 1)
RPV Water Inventory Control Instrumentation

FUNCTION

1. NotUsed
2. NotUsed
3. RHRSystemIsolation

a. Reactor Vessel Water Level – Low, Level 3

4. ReactorWaterCleanup(RWCU)System Isolation

(a)

APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS

(a)

REQUIRED CHANNELS PER FUNCTION

2 in one trip system

ALLOWABLE VALUE

≥ 11.5 inches

(a)
When automatic isolation of the associated penetration flow path(s) is credited in calculating DRAIN TIME.

a. Reactor Vessel Water Level – Low Low, Level 2

2 in one trip system

≥ -45 inches

Table 3.3.7.1-1 (page 1 of 1)
Control Room Emergency Outside Air Supply System Instrumentation

FUNCTION

1. Reactor Vessel Water Level – Low Low, Level 2
2. Drywell Pressure – High
3. Unit 1 Refuel Floor High Exhaust Duct Radiation – High
4. Unit 2 Refuel Floor High Exhaust Duct Radiation – High
5. Unit 1 Refuel Floor Wall Exhaust Duct Radiation – High
6. Unit 2 Refuel Floor Wall Exhaust Duct Radiation – High
7. Railroad Access Shaft Exhaust

   Duct Radiation – High

8. Main Control Room Outside Air Intake Radiation – High
9. Manual Initiation

APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS

REQUIRED CHANNELS PER TRIP SYSTEM

2

2 1

1

1

1

1

1

1

CONDITIONS REFERENCED FROM REQUIRED ACTION A.1

B

B B

B

B

B

B

C

B

SURVEILLANCE REQUIREMENTS

SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.4 SR 3.3.7.1.5

SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.5

SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.4 SR 3.3.7.1.5

SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.4 SR 3.3.7.1.5

SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.4 SR 3.3.7.1.5

SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.4 SR 3.3.7.1.5

SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.4 SR 3.3.7.1.5

SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.4 SR 3.3.7.1.5

SR 3.3.7.1.5

ALLOWABLE VALUE

≥ -45 inches

≤ 1.88 psig ≤ 25 mR/hr

≤ 25 mR/hr

≤ 28 mR/hr

≤ 28 mR/hr

≤ 7 mR/hr

≤ 5 mR/hr

n/a

1, 2,

1, 2, (a)

(a)

(a)

(a)

(b)

3

3

1, 2, 3, (a)

1, 2, 3 (a)

1. (a) During CORE ALTERATIONS and during movement of irradiated fuel assemblies in the secondary containment.
2. (b) During movement of irradiated fuel assemblies within the Railroad Access Shaft, and above the Railroad Access Shaft with the Railroad Access Shaft Equipment Hatch open.

3.3 INSTRUMENTATION
3.3.8.1 Loss of Power (LOP) Instrumentation

LCO 3.3.8.1 The LOP instrumentation for each Function in Table 3.3.8.1-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTIONS ---NOTE-------------------------------- -------------------------- Separate Condition entry is allowed for each channel. ------------------- ---

CONDITION

1. One or more required channels inoperable.
2. As required by Required Action A.1 and referenced in Table 3.3.8.1-1.
3. AsrequiredbyRequired Action A.1 and referenced in Table 3.3.8.1-1.

REQUIRED ACTION

COMPLETION TIME Immediately

1 hour

1hour

A.1 Enter the Condition referenced in Table 3.3.8.1-1 for the channel.
B.1 Place channel in trip.
C.1 Restore the inoperable channel.

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS), REACTOR PRESSURE VESSEL (RPV) WATER INVENTORY CONTROL, AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.2 RPV Water Inventory Control

LCO 3.5.2

DRAIN TIME of RPV water inventory to the top of active fuel (TAF) shall be ≥ 36 hours.

AND
One low pressure ECCS injection/spray subsystem shall be OPERABLE.

--NOTE--- A Low Pressure Coolant Injection (LPCI) subsystem may be considered OPERABLE during alignment and operation for decay heat removal if capable of being manually realigned and not otherwise inoperable. --

APPLICABILITY:

MODES 4 and 5

ACTIONS CONDITION

1. Required ECCS injection/spray subsystem inoperable.
2. RequiredActionand associated Completion Time of Condition A not met.

REQUIRED ACTION

COMPLETION TIME 4 hours

Immediately

A.1 Restore required ECCS injection/spray subsystem to OPERABLE status.	
B.1 Initiate action to establish a method of water injection capable of operating without offsite electrical power.	

ACTIONS (continued) CONDITION

C. DRAIN TIME < 36 hours and ≥ 8 hours.

REQUIRED ACTION

COMPLETION TIME 4 hours

4 hours

4 hours

C.1 Verify secondary containment boundary is

capable of being established in less than the DRAIN TIME.

AND

C.2 Verify each secondary containment penetration

flow path is capable of being isolated in less than the DRAIN TIME.

AND

C.3 Verify one standby gas treatment (SGT) subsystem

is capable of being placed in operation in less than the DRAIN TIME.

ACTIONS (continued) CONDITION

D. DRAINTIME<8hours.

REQUIRED ACTION

RPV Water Inventory Control 3.5.2

COMPLETION TIME

D.1 ----------- NOTE --------------- Required ECCS

injection/spray subsystem or additional method of water injection shall be capable of operating without offsite electrical power. ------------------------------------

Initiate action to establish an additional method of water injection with water sources capable of maintaining RPV water level > TAF for

≥ 36 hours. AND

D.2 Initiate action to establish secondary containment

boundary. AND

D.3 Initiate action to isolate each secondary containment

penetration flow path or verify it can be automatically or manually isolated from the control room.

AND

D.4 Initiate action to verify one SGT subsystem is capable of

being placed in operation.

Immediately

Immediately

Immediately

Immediately

ACTIONS (continued) CONDITION

E. RequiredActionand associated Completion Time of Condition C or D not met.

OR
DRAIN TIME < 1 hour.

REQUIRED ACTION

RPV Water Inventory Control 3.5.2

COMPLETION TIME Immediately

E.1 Initiate action to restore DRAIN TIME to ≥ 36 hours.

SURVEILLANCE REQUIREMENTS SURVEILLANCE

FREQUENCY

In accordance with the Surveillance Frequency Control Program

In accordance with the Surveillance Frequency Control Program

In accordance with the Surveillance Frequency Control Program.

SR 3.5.2.1

SR 3.5.2.2

SR 3.5.2.3

Verify DRAIN TIME ≥ 36 hours.

Verify, for a required LPCI subsystem, the suppression pool water level is ≥ 20 ft 0 inches.

Verify, for a required Core Spray (CS) subsystem, the:

1. Suppression pool water level is ≥ 20 ft 0 inches; or
2. Condensate storage tank water level is ≥ 49% of capacity.

SURVEILLANCE REQUIREMENTS (continued) SURVEILLANCE

RPV Water Inventory Control 3.5.2

FREQUENCY

In accordance with the Surveillance Frequency Control Program

SR 3.5.2.4

SR 3.5.2.5 SR 3.5.2.6

Verify, for the required ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.

Not Used

-----------------------------NOTES-------------------------------

1. Operation may be through the test return line.
2. Credit may be taken for normal system operation to satisfy this SR.

Operate the required ECCS injection/spray subsystem for ≥ 10 minutes.

Verify each valve credited for automatically isolating a penetration flow path actuates to the isolation position on an actual or simulated isolation signal.

-------------------------------NOTE------------------------------- Vessel injection/spray may be excluded. ---

Verify the required ECCS injection/spray subsystem can be manually operated.

SR 3.5.2.7

SR 3.5.2.8

In accordance with the Surveillance Frequency Control Program

In accordance with the Surveillance Frequency Control Program

In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS SURVEILLANCE

FREQUENCY

SR 3.8.21

----------------------------------NOTE-------------------------------- The following SRs must be met but are not required
to be performed:

SR 3.8.1.3; SR 3.8.1.14; and SR 3.8.1.9; SR 3.8.1.16.
SR 3.8.1.10;

For required Unit 1 AC sources, the following SRs of Unit 1 Specification 3.8.1 are applicable:

SR 3.8.1.1; SR 3.8.1.3; SR 3.8.1.4; SR 3.8.1.5; SR 3.8.1.6;

SR 3.8.1.9;
SR 3.8.1.10;
SR 3.8.1.14; and SR 3.8.1.16.

In accordance with applicable SRs

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SUSQUEHANNA NUCLEAR, LLC
ALLEGHENY ELECTRIC COOPERATIVE, INC. DOCKET NO. 50-388
SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2 AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 261 Renewed License No. NPF-22

1. The U.S. Nuclear Regulatory Commission (NRC or the Commission) has found that:

1. The application for the amendment filed by Susquehanna Nuclear, LLC, dated September 1, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 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.

Attachment:
Changes to the Renewed Facility

Operating License and Technical Specifications

Date of Issuance: February 18, 2021

2. 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. NPF-22 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 261, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. Susquehanna Nuclear, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation

ATTACHMENT TO LICENSE AMENDMENT NO. 261 SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2 RENEWED FACILITY OPERATING LICENSE NO. NPF-22 DOCKET NO. 50-388

Replace the following page of the Renewed Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

REMOVE INSERT Page 3 Page 3

Replace the following pages of the Appendix A, Technical Specifications, with the attached revised pages. The revised pages are identified by amendment number and contain a marginal line indicating the areas of change.

REMOVE INSERT 1.1-3 1.1-3 1.1-3a 1.1-3a* 1.1-4 1.1-4*

1. 3.3-47a 3.3-47a
2. 3.3-47b 3.3-47b
3. 3.3-47c 3.3-47c

3.3-72 3.3-72 3.5-8 3.5-8 3.5-8a 3.5-8a

9. 3.5-9 3.5-9
10. 3.5-10 3.5-10
11. 3.5-11 3.5-11

21. 3.8-21 3.8-21
22. 3.8-22 3.8-22

*no changes; content rolled across pages only

3. (3) Susquehanna Nuclear, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, posses, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed neutron sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;
4. (4) Susquehanna Nuclear, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, posses, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and
5. (5) Susquehanna Nuclear, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission nor or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

1. (1) Maximum Power Level

   Susquehanna Nuclear, LLC is authorized to operate the facility at reactor core power levels not in excess of 3952 megawatts thermal in accordance with the conditions specified herein. The preoperational tests, startup tests and other items identified in License Conditions 2.C.(20), 2.C.(21), 2.C.(22), and 2.C.(23) to this license shall be completed as specified.

2. (2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 261, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. Susquehanna Nuclear, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

For Surveillance Requirements (SRs) that are new in Amendment 151 to Facility Operating License No. NPF-22, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 151. For SRs that existed prior to Amendment 151, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of

Amendment 151.

DOSE EQUIVALENT I-131 (continued)

actually present. The conversion factors that are used for this calculation of committed effective dose equivalent (CEDE) from inhalation shall be those listed in Table 2.1 of Federal Guidelines Report 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” EPA, 1988, as described in Regulatory Guide 1.183. The factors in the column headed “effective” yield doses corresponding to the CEDE. The conversion factors that are used for the calculation of EDE (or DDE) from external exposure (submersion) shall be those listed in Table III.1 of Federal Guidance Report 12, “External Exposure to Radionuclides in Air, Water, and Soil,” EPA, 1993, as described in Regulatory Guide 1.183. The factors in the column headed “effective” yield doses corresponding to the EDE.

The DRAIN TIME is the time it would take for the water inventory in and above the Reactor Pressure Vessel (RPV) to drain to the top of the active fuel (TAF) seated in the RPV assuming:

DRAIN TIME

a) b)

The water inventory above the TAF is divided by the limiting drain rate;

The limiting drain rate is the larger of the drain rate through a single penetration flow path with the highest flow rate, or the sum of the drain rates through multiple penetration flow paths susceptible to a common mode failure for all penetration flow paths below the TAF except:

1.

2.

Penetration flow paths connected to an intact closed system, or isolated by manual or automatic valves that are closed and administratively controlled in the closed position, blank flanges, or other devices that prevent flow of reactor coolant through the penetration flow paths;

Penetration flow paths capable of being isolated by valves that will close automatically without offsite power prior to the RPV water level being equal to the TAF when actuated by RPV water level isolation instrumentation; or

DRAIN TIME (continued)

3. Penetration flow paths with isolation devices that can be closed prior to the RPV water level being equal to the TAF by a dedicated operator trained in the task, who in continuous communication with the control room, is stationed at the controls, and is capable of closing the penetration flow path isolation device without offsite power.

c) The penetration flow paths required to be evaluated per paragraph b) are assumed to open instantaneously and are not subsequently isolated, and no water is assumed to be subsequently added to the RPV water inventory;

4. d) No additional draining events occur; and
5. e) Realistic cross-sectional areas and drain rates are used.

A bounding DRAIN TIME may be used in lieu of a calculated value.

The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable.

The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

The EOC RPT SYSTEM RESPONSE TIME shall be that time interval from initial signal generation by the associated turbine stop valve limit switch or from when the turbine control valve hydraulic oil control oil pressure drops below the pressure switch setpoint to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

END OF CYCLE RECIRCULATION PUMP TRIP (EOC RPT) SYSTEM RESPONSE TIME

ISOLATION SYSTEM RESPONSE TIME

ISOLATION SYSTEM RESPONSE TIME

(continued) LEAKAGE

include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

LEAKAGE shall be:

1. Identified LEAKAGE
    1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a collecting tank; or
    2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE;
2. Unidentified LEAKAGE

    All LEAKAGE into the drywell that is not identified LEAKAGE;

3. Total LEAKAGE
   Sum of the identified and unidentified LEAKAGE;
4. Pressure Boundary LEAKAGE

   LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.

The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all required logic components (i.e., all required relays and contacts, trip units, solid state logic elements, etc.) of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.

LINEAR HEAT GENERATION RATE (LHGR)

LOGIC SYSTEM FUNCTIONAL TEST

3.3 INSTRUMENTATION
3.3.5.2 Reactor Pressure Vessel (RPV) Water Inventory Control Instrumentation

LCO 3.3.5.2 The RPV Water Inventory Control instrumentation for each Function in Table 3.3.5.2-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.5.2-1.

ACTIONS --NOTE-----------------------------------
----------------------- Separate Condition entry is allowed for each channel. ---------------------
--- --

CONDITION

A. Oneormorechannels inoperable.

REQUIRED ACTION

COMPLETION TIME Immediately

Immediately

Immediately

A.1 Initiate action to place channel in trip.

OR

1. A.2.1 Declare associated penetration flow path(s)

   incapable of automatic isolation.

   AND

2. A.2.2 Initiate action to calculate DRAIN TIME.

SURVEILLANCE REQUIREMENTS -- NOTE--- These SRs apply to each Function in Table 3.3.5.2-1. -- ----------------------------- --

SR 3.3.5.2.1

SR 3.3.5.2.2

SURVEILLANCE Perform CHANNEL CHECK.

-------------------------------NOTE-------------------------------- A test of all required contacts does not have to be performed. --

Perform CHANNEL FUNCTIONAL TEST.

FREQUENCY

In accordance with the Surveillance Frequency Control Program

In accordance with the Surveillance Frequency Control Program

Table 3.3.5.2-1 (page 1 of 1)
RPV Water Inventory Control Instrumentation

FUNCTION

1. NotUsed
2. NotUsed
3. RHRSystemIsolation

a. Reactor Vessel Water Level – Low, Level 3

4. ReactorWaterCleanup(RWCU)System Isolation

APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS

(a)

REQUIRED CHANNELS PER FUNCTION

2 in one trip system

ALLOWABLE VALUE

≥ 11.5 inches

(a)
(a) When automatic isolation of the associated penetration flow path(s) is credited in calculating DRAIN TIME.

a. Reactor Vessel Water Level – Low Low, Level 2

2 in one trip system

≥ -45 inches

3.3 INSTRUMENTATION
3.3.8.1 Loss of Power (LOP) Instrumentation

LCO 3.3.8.1 The LOP instrumentation for each Function in Table 3.3.8.1-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTIONS ---NOTE--- Separate Condition entry is allowed for each channel. ---

CONDITION

1. Oneormorerequired channels inoperable for reasons other than Condition B.
2. Oneormorerequired channels associated with Unit 1 4.16 kV ESS Buses in one Division inoperable for the performance of Unit 1 SR 3.8.1.19.
3. As required by Required Action A.1 and

   referenced in Table 3.3.8.1-1.

REQUIRED ACTION

COMPLETION TIME Immediately

8hours

1 hour

A.1 Enter the Condition referenced in Table 3.3.8.1-1 for the channel.
B.1 Restore the inoperable channels.
C.1 Place channel in trip.

EMERGENCY CORE COOLING SYSTEMS (ECCS), REACTOR PRESSURE VESSEL (RPV) WATER INVENTORY CONTROL, AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.2
LCO 3.5.2

RPV Water Inventory Control

APPLICABILITY:

MODES 4 and 5

ACTIONS CONDITION

1. Required ECCS injection/spray subsystem inoperable.
2. RequiredActionand associated Completion Time of Condition A not met.

REQUIRED ACTION

COMPLETION TIME 4 hours

Immediately

DRAIN TIME of RPV water inventory to the top of active fuel (TAF) shall be ≥ 36 hours.

AND
One low pressure ECCS injection/spray subsystem shall be OPERABLE.

--NOTE--- A Low Pressure Coolant Injection (LPCI) subsystem may be considered OPERABLE during alignment and operation for decay heat removal if capable of being manually realigned and not otherwise inoperable. --

A.1 Restore required ECCS injection/spray subsystem to OPERABLE status.
B.1 Initiate action to establish a method of water injection capable of operating without offsite electrical power.

ACTIONS (continued) CONDITION

C. DRAIN TIME < 36 hours and ≥ 8 hours.

REQUIRED ACTION

COMPLETION TIME 4 hours

4 hours

4 hours

C.1 Verify secondary containment boundary is capable of being established in less than the DRAIN TIME.

AND

C.2 Verify each secondary containment penetration

flow path is capable of being isolated in less than the DRAIN TIME.

AND

C.3 Verify one standby gas treatment (SGT) subsystem

is capable of being placed in operation in less than the DRAIN TIME.

ACTIONS (continued) CONDITION

D. DRAINTIME<8hours.

REQUIRED ACTION

COMPLETION TIME

D.1 -------------NOTE------------------ Required ECCS

injection/spray subsystem or additional method of water injection shall be capable of operating without offsite electrical power. ---------------------------------------

Initiate action to establish an additional method of water injection with water sources capable of maintaining RPV water level > TAF for

≥ 36 hours. AND

D.2 Initiate action to establish secondary containment

boundary. AND

D.3 Initiate action to isolate each secondary containment

penetration flow path or verify it can be automatically or manually isolated from the control room.

AND

D.4 Initiate action to verify one SGT subsystem is capable of

being placed in operation.

Immediately

Immediately

Immediately

Immediately

ACTIONS (continued) CONDITION

E. RequiredActionand associated Completion Time of Condition C or D not met.

OR
DRAIN TIME < 1 hour.

REQUIRED ACTION

COMPLETION TIME Immediately

E.1 Initiate action to restore DRAIN TIME to ≥ 36 hours.

SURVEILLANCE REQUIREMENTS SURVEILLANCE

FREQUENCY

In accordance with the Surveillance Frequency Control Program

In accordance with the Surveillance Frequency Control Program

In accordance with the Surveillance Frequency Control Program

SR 3.5.2.1

SR 3.5.2.2

SR 3.5.2.3

Verify DRAIN TIME ≥ 36 hours.

Verify, for a required LPCI subsystem, the suppression pool water level is ≥ 20 ft 0 inches.

Verify, for a required Core Spray (CS) subsystem, the:

1. Suppression pool water level is ≥ 20 ft 0 inches; or
2. Condensate storage tank water level is ≥ 49% of capacity.

FREQUENCY

In accordance with the Surveillance Frequency Control Program

SR 3.5.2.4

SR 3.5.2.5 SR 3.5.2.6

Verify, for the required ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.

Not Used

------------------------------NOTES------------------------------

1. Operation may be through the test return line.
2. Credit may be taken for normal system operation to satisfy this SR.

Operate the required ECCS injection/spray subsystem for ≥ 10 minutes.

Verify each valve credited for automatically isolating a penetration flow path actuates to the isolation position on an actual or simulated isolation signal.

------------------------------NOTE------------------------------ Vessel injection/spray may be excluded. ---

Verify the required ECCS injection/spray subsystem can be manually operated.

SR 3.5.2.7

SR 3.5.2.8

In accordance with the Surveillance Frequency Control Program

In accordance with the Surveillance Frequency Control Program

In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS SURVEILLANCE

FREQUENCY

SR 3.8.2.1

------------------------------NOTE-------------------------------- The following SRs must be met but are not required to be performed:

SR 3.8.1.3; SR 3.8.1.14; and SR 3.8.1.9; SR 3.8.1.16.
SR 3.8.1.10;

For required Unit 2 AC sources, the following SRs of Unit 2 Specification 3.8.1 are applicable:

SR 3.8.1.1; SR 3.8.1.3; SR 3.8.1.4; SR 3.8.1.5; SR 3.8.1.6;

SR 3.8.1.9;
SR 3.8.1.10;
SR 3.8.1.14; and SR 3.8.1.16.

In accordance with applicable SRs

SURVEILLANCE REQUIREMENTS (continued) SURVEILLANCE

FREQUENCY

In accordance with applicable SRs

SR 3.8.2.2

-------------------------------NOTE------------------------------- When Unit 1 is in MODE 4 or 5, the Note to Unit 1
SR 3.8.2.1 is applicable for the performance of required Unit 1 SRs. ---------------------------
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For required Unit 1 AC sources, the following SRs of Unit 1 Specification 3.8.1 are applicable:

SR 3.8.1.1; SR 3.8.1.3; SR 3.8.1.4; SR 3.8.1.5; SR 3.8.1.6;

SR 3.8.1.9; SR 3.8.1.10; SR 3.8.1.14; SR 3.8.1.16; and

SR 3.8.1.8 (when more than one Unit 1 offsite circuit is required).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

1.0

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 279 TO
RENEWED FACILITY OPERATING LICENSE NO. NPF-14
AND AMENDMENT NO. 261 TO
RENEWED FACILITY OPERATING LICENSE NO. NPF-22 SUSQUEHANNA NUCLEAR, LLC
ALLEGHENY ELECTRIC COOPERATIVE, INC. SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 DOCKET NOS. 50-387 AND 50-388

INTRODUCTION

Susquehanna Nuclear, LLC (the licensee) requested changes to the technical specifications (TSs) for Susquehanna Steam Electric Station (Susquehanna), Units 1 and 2 by license amendment request (LAR, application), dated September 1, 2020

(Agencywide Documents Access and Management System (ADAMS) Accession No. ML20245E192). In its application, the licensee requested that the U.S. Nuclear Regulatory Commission (NRC, the Commission) process the proposed amendment under the Consolidated Line Item Improvement Process (CLIIP). The proposed changes would revise the TSs related to reactor pressure vessel (RPV) water inventory control (WIC) based on Technical Specifications Task Force (TSTF) Traveler TSTF-582, Revision 0, “RPV WIC Enhancements” (ADAMS Accession No. ML19240A260), and the associated NRC staff safety evaluation (SE) of TSTF-582 (ADAMS Accession

No. ML20219A333).

The boiling-water reactor (BWR) RPV design includes multiple penetrations located below the top of active fuel (TAF). These penetrations provide entry for control rods, recirculation flow, reactor water cleanup, and shutdown cooling. Since these penetrations are below the TAF, this creates a potential to drain the reactor vessel water inventory and lose effective core cooling. The loss of water inventory and effective core cooling can potentially lead to fuel cladding failure and radioactive release. Drain time is the time it would take for the water inventory in and above the RPV to drain to the TAF.

1.1 Proposed TS Changes to Adopt TSTF-582

In accordance with NRC staff-approved TSTF-582, the licensee proposed changes that would revise the TSs related to RPV WIC to incorporate operating experience and to correct errors and omissions that the licensee incorporated into the Susquehanna, Units 1 and 2, TSs when adopting TSTF-542, Revision 2, “Reactor Pressure Vessel Water Inventory Control” (ADAMS Accession No. ML16074A448). Specifically, the licensee proposed the following changes to adopt TSTF-582:

- • In the TS 1.1, “Definitions,” “Drain Time” would be revised to move the examples of common mode failure mechanisms to the Bases and delete seismic events.
- • In the TS 1.1, “Definitions,” “Drain Time,” the exception from considering the drain time for penetration flow paths isolated with manual or automatic valves that are that are “locked, sealed, or otherwise secured” would be revised to apply the exception for manual or automatic valves that are “closed and administratively controlled.”
- • The Actions of TS 3.3.5.2 would be revised to permit placing an inoperable isolation channel in trip as an alternative to declaring the associated penetration flow path incapable of automatic isolation.
- • TS 3.3.5.2 Required Action B.2 requires calculating drain time with a completion time (CT) of “immediately.” The Required Action would be renumbered as A.2.2 and revised to state, “Initiate action to calculate Drain Time.”
- • In TS 3.5.2, the first use of the acronym “SGT” would be defined in Required Action C.3, and the acronym “SGT” would be used in Required Action D.4.
- • TS 3.5.2 and TS 3.3.5.2 would be revised to eliminate the requirement for a manual emergency core cooling system (ECCS) initiation signal to start the required ECCS injection/spray subsystem and to instead rely on manual valve alignment and pump start. TS 3.5.2 surveillance requirements (SRs) related to manual initiation using the ECCS signal (such as verifying automatic alignment of

valves on an initiation signal) would be eliminated. Related to this change, the TS 3.3.5.2 functions, SRs, and Actions that only support manual initiation using an ECCS signal (including interlocks and minimum flow instruments) would be eliminated.

- • Susquehanna Units 1 and 2 share secondary containment structures between units. The TS 3.5.2 Actions would be revised to recognize that an operable secondary containment and operable secondary containment isolation valves satisfy the Required Actions.
- • A redundant definition of “LPCI” in SR 3.5.2.2 would be eliminated.
- • SR 3.5.2.6, that requires operating the required ECCS injection/spray subsystem for at least 10 minutes through the recirculation line, would be modified by the addition of two notes. The first Note would replace the existing SR that the ECCS subsystem be run through the recirculation line with a Note that states that operation may be through the test return line. The second Note would permit crediting normal operation of the low pressure ECCS subsystem for performance of the SR.
- • TS 3.8.2, “AC [Alternating Current] Sources - Shutdown,” SR 3.8.2.1, would be revised to not require SRs that test the ability of the automatic diesel generator (DG) to start in Modes 4 and 5. TSTF-542 eliminated the automatic ECCS initiation in Modes 4 and 5.

1.2 Additional Proposed TS Changes
The licensee proposed to make the following additional changes:

1.2.1

TS 3.3.8.1, “Loss of Power (LOP) Instrumentation,” would be revised to delete “When the associated diesel generator is required to be OPERABLE by LCO 3.8.2, ‘AC Sources – Shutdown’” from the applicability.

SR 3.8.2.1 would be revised to remove SRs 3.8.1.7, 3.8.1.11, 3.8.1.12, 3.8.1.13, 3.8.1.15, 3.8.1.18, and 3.8.1.19 from the list of SRs that are applicable.

Editorial Variations

In the Susquehanna, Unit 1 TSs, the title of Table 3.3.5.1-1, “Emergency Core Cooling System Instrumentation,” Function 3.a, would be revised to add one instance of the word “Low.” The proposed Function 3.a title would be “Reactor Vessel Water Level – Low Low, Level 2.”

Table 3.3.5.2-1 – All the Functions

The licensee proposed to retain the numbering for Function 3.a, “RHR [Residual Heat Removal] System Isolation, Reactor Vessel Water Level – Low, Level 3,” and Function 4.a, “Reactor Water Cleanup (RWCU) System Isolation Reactor Vessel Water Level – Low Low, Level 2.” Specifically, rather than deleting Functions 1 and 2 in their entirety and renumbering Functions 3 and 4, the license proposed revising Functions 1 and 2 to state, “Not Used.”

In the Susquehanna, Unit 1 TSs, the title of Table 3.3.7.1-1, Function 3, would be revised to add “High Exhaust Duct.” The proposed title of Function 3 would be “Unit 1 Refuel Floor High Exhaust Duct Radiation – High.”

The licensee proposed to modify the title of TS 3.5.2 by replacing the words “Reactor Pressure Vessel” from the title with the acronym “RPV.”

The licensee proposed to retain the numbering for the existing SRs in TS 3.5.2. Specifically, rather than deleting SR 3.5.2.5 in its entirety and renumbering SR 3.5.2.6, SR 3.5.2.7, and SR 3.5.2.8, the licensee proposed revising SR 3.5.2.5 to state, “Not Used” and leaving the remaining SRs as their current numbers.

REGULATORY EVALUATION

The regulation in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36(c)(2) requires that TSs include limiting conditions for operation (LCOs). Per 50.36(c)(2)(i), LCOs “are the lowest functional capability or performance levels of equipment required for safe operation of the facility.” The regulation also requires that 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 TS until the condition can be met.

The regulation at 10 CFR 50.36(c)(3) requires that TSs include items in the category of SRs, which are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

The NRC staff’s guidance for the review of TSs is in Chapter 16.0, “Technical Specifications,” of NUREG-0800, Revision 3, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition” (SRP), dated March 2010 (ADAMS Accession No. ML100351425). As described therein, as part of the regulatory standardization effort, the NRC staff has prepared standard technical specifications (STS) for each of the LWR nuclear designs. Accordingly, the NRC staff’s review includes consideration of whether the proposed changes are consistent with the “Standard Technical Specifications, General Electric, BWR/4 Plants,” NUREG 1433, Volume 1, “Specifications,” and Volume 2, “Bases,” Revision 4.0, dated April 2012 (ADAMS Accession Nos. ML12104A192 and ML12104A193, respectively), as modified by NRC-approved travelers.

Traveler TSTF-582 revised the STS related to RPV WIC to incorporate operating experience and to correct editorial errors in TSTF-542, Revision 2, “Reactor Pressure Vessel Water Inventory Control.” The NRC approved TSTF-542, Revision 2, on December 20, 2016 (ADAMS Package Accession No. ML16343B066). The NRC staff approved TSTF-582 under the consolidated line item improvement process (CLIIP) in letter dated August 13, 2020 (ADAMS Accession No. ML20219A333). The TSTF-582 safety evaluation (SE) states that a licensee may adopt the STS changes approved in TSTF-582, if the licensee has already adopted the STS changes approved in TSTF-542.

1. 3.0 TECHNICAL EVALUATION
2. 3.1 Proposed TS Changes to Adopt TSTF-582

The NRC staff compared the licensee's proposed TS changes in Section 1.1 of this SE against the changes approved in TSTF-582. In accordance with SRP Chapter 16.0, the NRC staff determined that the STS changes approved in TSTF-582 are applicable to the Susquehanna TSs because Susquehanna, Units 1 and 2, are a BWR/4 design, and the NRC staff approved the TSTF-582 changes for BWR/4 designs. The licensee meets the TSTF-582 SE provision for adoption of TSTF-582 since the licensee adopted TSTF-542 on September 26, 2018 (ADAMS Accession No. ML18222A203). Therefore, the NRC staff concluded that the licensee's proposed changes to the Susquehanna TSs in Section 1.1 of this SE are acceptable in that they are consistent with TSTF-582 and the terms for use stated in the NRC SE of TSTF-582.

The NRC staff finds that proposed changes to TS 1.1, "Definition," and LCOs 3.3.5.2 and 3.5.2, correctly specify the lowest functional capability or performance levels of equipment required for safe operation of the facility in accordance with 10 CFR 50.36(c)(2)(i). In addition, the NRC staff finds that proposed changes to the actions of LCOs 3.3.5.2 and 3.5.2 are adequate remedial actions to be taken until each LCO can be met and provide protection to the health and safety of the public, thereby satisfying 10 CFR 50.36(c)(2)(i).

The NRC staff finds that the proposed revisions to the SRs in TS 3.3.5.2, 3.5.2, and 3.8.2 continue to provide requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met in accordance with 10 CFR 50.36(c)(3).

Thus, the proposed changes continue to meet the requirements of 10 CFR 50.36(c)(2)(i) and 50.36(c)(3) as discussed in Section 3.0 of the NRC SE of TSTF-582.

3.2 Additional Proposed TS Changes
The licensee proposed to make the following additional changes:

- • TS 3.3.8.1, "Loss of Power (LOP) Instrumentation," APPLICABILITY would be revised to delete "When the associated diesel generator is required to be OPERABLE by
  LCO 3.8.2, 'AC Sources – Shutdown'" from the applicability.
- • SR 3.8.2.1 would be revised to remove SRs 3.8.1.7, 3.8.1.11, 3.8.1.12, 3.8.1.13, 3.8.1.15, 3.8.1.18, and 3.8.1.19 from the list of SRs that are applicable.

  The NRC staff notes that the above proposed changes are consistent with TSTF-583-T, Revision 0, "TSTF-582 Diesel Generator Variation" (ADAMS Accession No. ML20248H330). The NRC staff's evaluation of these additional changes is provided below.

3.2.1.1 TS 3.3.8.1 – Applicability

The licensee stated that TS 3.8.2 does not require automatic start and loading of a DG within 10 seconds on an ECCS initiation signal or a loss-of-offsite-power (LOOP) signal. Currently, TS 3.3.8.1, "Loss of Power (LOP) Instrumentation," is applicable in Modes 1, 2, and 3, and when the associated DG is required to be operable by TS 3.8.2. The NRC staff confirmed that TS 3.8.2 no longer requires automatic start and loading of a DG on a

LOP signal. The NRC staff finds it acceptable to revise the Applicability of LCO 3.3.8.1 by deleting “When the associated diesel generator is required to be OPERABLE by LCO 3.8.2, ‘AC Sources – Shutdown’,” because the LOP instrumentation that generates the LOP signal does not need to be operable when the DG is required to be operable by TS 3.8.2. Therefore, the NRC staff concludes that the LCO applicability changes will continue to provide for the lowest functional capability or performance levels of equipment required for safe operation of the facility and, therefore, meet the LCO requirements of 10 CFR 50.36(c)(2).

3.2.1.2 SR 3.8.2.1

LCO 3.8.2, “AC Sources - Shutdown,” requires one offsite circuit and two DGs capable of supplying one division of the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.8, “Distribution Systems - Shutdown,” to be operable in shutdown conditions. The existing SR 3.8.2.1 lists the TS 3.8.1 SRs that are applicable in shutdown conditions with some exceptions.

TS SR 3.8.1.7 and SR 3.8.1.15 require that the DG starts from standby or hot conditions, respectively, and achieves required voltage and frequency within 10 seconds within required steady state voltage and frequency ranges. The 10-second start requirement associated with the DG automatic start supports assumptions in the design-basis loss-of-coolant accident analysis. The NRC staff confirmed that 10-second timing is not required during a manual DG start to respond to a draining event, which has a minimum drain time of 1 hour. In addition, SR 3.8.1.2, which requires the DG to start from standby conditions and achieve the required steady state voltage and frequency ranges, is applicable under SR 3.8.2. The NRC staff finds that the SR 3.8.1.7 and SR 3.8.1.15 testing for the DG’s capability to achieve required steady state voltage and frequency ranges will be performed in SR 3.8.1.2, since SR 3.8.1.2 provides the test for this DG capability. Therefore, the NRC staff finds it acceptable to add SR 3.8.1.7 and SR 3.8.1.15 to the list of TS 3.8.1 SRs that are not applicable under SR 3.8.2.1.

TS SR 3.8.1.18 states, “Verify each sequenced load is within required limits of the design interval.” This SR verifies the 10 percent load sequence time interval tolerance between each sequenced load block when loads are sequentially connected to the engineered safety features (ESF) bus by an automatic sequencer while the DG is tied to the ESF bus. TS 3.5.2 requires manual starting of the equipment for water injection to respond to a draining event so that the DG will be manually loaded during a draining event. No other postulated events require automatic loading of the DG during shutdown conditions. The NRC staff confirmed that with respect to SR 3.8.18, the load sequencer is used for the automatic loading of the DG and is not used during a manual loading of the DG. Therefore, the NRC staff finds it acceptable to add SR 3.8.1.18 to the list of TS 3.8.1 SRs that are not applicable under SR 3.8.2.1.

The Susquehanna Unit 2 TSs contain an additional SR 3.8.2.2, which requires that certain SRs from Unit 1 LCO 3.8.1 are met for Unit 2. The licensee proposed modifying Unit 2 SR 3.8.2.2 by eliminating SR 3.8.1.7, SR 3.8.1.11, SR 3.8.1.15, SR 3.8.1.18, and SR 3.8.1.19 from the list of required Unit 1 LCO 3.8.1 SRs in Susquehanna, Unit 2 SR 3.8.2.2. The NRC staff finds the proposed change acceptable because the list of SRs to be removed is equivalent to the changes made to SR 3.8.2.1 for each unit’s TSs and is needed to align the operability requirements for Unit 1 AC sources powering Unit 2 equipment with the operability requirements for Unit 1 AC sources powering only Unit 1 equipment.

The NRC staff finds that the proposed changes to revise SR 3.8.2.1 are acceptable because the remaining applicable SRs will continue to demonstrate the operability of the required AC power sources and, as such, ensure the availability of the AC power required to operate the plant in a safe manner and mitigate postulated events during shutdown conditions. Therefore, the NRC staff finds the proposed changes to SR 3.8.2.1 are acceptable because the changes continue to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the associated LCO will continue to be met in accordance with 10 CFR 50.36(c)(3).

3.2.2 Editorial Changes

The NRC staff reviewed the editorial variations proposed by the licensee described in Section 1.2.2 of this SE.

The licensee proposed to correct an editorial error in Susquehanna Unit 1 TS Table 3.3.5.1-1, “Emergency Core Cooling System Instrumentation,” Function 3.a. The licensee stated that during adoption of TSTF-542, one instance of the word “Low” was inadvertently deleted. The function should be titled “Reactor Vessel Water Level – Low Low, Level 2.” However, it is currently titled, “Reactor Vessel Water Level – Low Level 2.” The NRC staff reviewed the correction and found it acceptable because the correct title used in the Susquehanna Updated Safety Analysis Review (UFSAR) and TSs is “Reactor Vessel Water Level – Low Low Level 2.” The NRC staff finds this change acceptable because it is editorial and does not substantively change the TS requirements.

The licensee proposed to retain the numbering for Table 3.3.5.2-1, Functions 3.a and 4.a. Rather than deleting Functions 1 and 2 in their entirety and re-numbering Functions 3 and 4, Functions 1 and 2 will be revised to state, “Not Used.” The licensee also proposed to retain the numbering for the existing Surveillance Requirements (SRs) in TS 3.5.2. Specifically, rather

than deleting SR 3.5.2.5 in its entirety and re-numbering SR 3.5.2.6, SR 3.5.2.7, and SR 3.5.2.8, the licensee proposes revising SR 3.5.2.5 to state, “Not Used” and leave the remaining SRs as their current number. The NRC staff finds this variation acceptable because it is editorial and does not substantively change the TS requirements. The proposed changes eliminate the need to revise existing Susquehanna Surveillance Procedures for the sole purpose of a changed SR number within TS 3.5.2 and Table 3.3.5.2-1.

The licensee proposed to correct an editorial error that was introduced during the adoption of TSTF-542. The licensee stated that during adoption of TSTF-542, an editorial error was introduced into the Susquehanna Unit 1 TSs. In Table 3.3.7.1-1, “Control Room Emergency Outside Air Supply System Instrumentation,” Function 3, the words “High Exhaust Duct” were inadvertently deleted. Function 3 should be titled, “Unit

1 Refuel Floor High Exhaust Duct Radiation – High." However, it is currently titled, "Unit 1 Refuel Floor Radiation – High." The NRC staff reviewed the correction and found it acceptable because the correct title used in the Susquehanna UFSAR and TSs is "Unit 1 Refuel Floor High Exhaust Duct Radiation – High." The NRC staff finds this change acceptable because it is editorial and does not substantively change the TS requirements.

The licensee proposed to modify the title of TS 3.5.2, to replace the words "Reactor Pressure Vessel" from the title with the acronym "RPV." The licensee stated that RPV is defined in the title of Chapter 3.5 of the TS; it is redundant to redefine the acronym in the title of TS 3.5.2. This change aligns TS 3.5.2 with TS 3.5.1 and TS 3.5.3. The NRC staff finds this change acceptable because it is editorial and does not substantively change the TS requirements.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendments on December 14, 2020. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change SRs. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on October 6, 2020 (85 FR 63149). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

Finally, the NRC staff reviewed the proposed TS changes for technical clarity and consistency with the existing requirements for customary terminology and formatting. The NRC staff finds that the proposed changes are consistent with Chapter 16 of the SRP and are therefore acceptable.

6.0 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.

Principal Contributor: T. Sweat Date: February 18, 2021

March 10, 2021 – Letter from Sujata Goetz, Project Manager Plant Licensing Branch I Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to Kevin Cimorelli Site Vice President Susquehanna Nuclear, LLC with subject of SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 280 AND 262 RE: REVISE TECHNICAL SPECIFICATION 3.8.1, "AC SOURCES – OPERATING," TO CREATE A NEW CONDITION FOR AN INOPERABLE MANUAL SYNCHRONIZATION CIRCUIT (EPID L-2019-LLA-0118)

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 280 to Renewed Facility Operating License No. NPF-14 and Amendment No. 262 to Renewed Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Units 1 and 2, respectively. These amendments consist of changes to the technical specifications in response to your application dated May 26, 2020.

The amendments revise Technical Specification 3.8.1, "AC [Alternating Current] Sources – Operating." Specifically, the amendments create a new technical specification action for an inoperable manual synchronization circuit requiring restoration within 14 days.

A copy of the related safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's monthly *Federal Register* notice.

SUSQUEHANNA NUCLEAR, LLC
ALLEGHENY ELECTRIC COOPERATIVE, INC. DOCKET NO. 50-387
SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1 AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 280 Renewed License No. NPF-14

1. The U.S. Nuclear Regulatory Commission (NRC or the Commission) has found that:

1. The application for the amendment filed by Susquehanna Nuclear, LLC, dated May 26, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 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, 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. NPF-14 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 280, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. Susquehanna Nuclear, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days.

James G. Danna, Chief Plant Licensing Branch I Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation

ATTACHMENT TO LICENSE AMENDMENT NO. 280 SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1 RENEWED FACILITY OPERATING LICENSE NO. NPF-14 DOCKET NO. 50-387

Replace the following page of the Renewed Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

REMOVE INSERT Page 3 Page 3

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain a marginal line indicating the area of change.

REMOVE	INSERT
4. 3.8-4	3.8-4
5. 3.8-5	3.8-5
6. 3.8-6	3.8-6
7. 3.8-7	3.8-7
8. 3.8-8	3.8-8
9. 3.8-9	3.8-9
10. 3.8-10	3.8-10
11. 3.8-11	3.8-11

3. (3) Susquehanna Nuclear, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, posses, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed neutron sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;
4. (4) Susquehanna Nuclear, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, posses, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

5. (5) Susquehanna Nuclear, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission nor or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

1. (1) Maximum Power Level

   Susquehanna Nuclear, LLC is authorized to operate the facility at reactor core power levels not in excess of 3952 megawatts thermal in accordance with the conditions specified herein. The preoperational tests, startup tests and other items identified in License Conditions 2.C.(36), 2.C.(37), 2.C.(38), and 2.C.(39) to this license shall be completed as specified.

2. (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 280, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. Susquehanna Nuclear, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

   For Surveillance Requirements (SRs) that are new in Amendment 178 to Facility Operating License No. NPF-14, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 178. For SRs that existed prior to Amendment 178, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 178.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. Two or more required DGs inoperable.	E.1	Restore at least three required DGs to OPERABLE status.	2 hours
F. Required Action and Associated Completion Time of Condition A, B, C, D, or E not met.	F.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	F.2	Be in MODE 4.	36 hours
G. One or more offsite circuits and two or more required DGs inoperable. <u>OR</u> One required DG and two offsite circuits inoperable.	G.1	Enter LCO 3.0.3.	Immediately
H. Manual synchronization circuit inoperable.	H.1	Restore manual synchronization circuit to OPERABLE status.	14 days

SURVEILLANCE REQUIREMENTS

--NOTE--
---------- Four DGs are required and a DG is only considered OPERABLE when the DG is aligned to the Class 1E distribution system. DG Surveillance Requirements have been modified to integrate the necessary testing to demonstrate the availability of DG E and ensure its OPERABILITY when substituted for any other DG. If the DG Surveillance Requirements, as modified by the associated Notes, are met and performed, DG E can be considered available and OPERABLE when substituted for any other DG after performance of SR 3.8.1.3 and SR 3.8.1.7. --
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SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.1.6	Verify the fuel oil transfer system operates to automatically transfer fuel oil from the storage tanks to each engine mounted tank.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.7	------------------------------NOTES------------------------------ 1. All DG starts may be preceded by an engine prelube period. 2. A single test at the specified Frequency will satisfy this Surveillance for both units. -- Verify each DG starts from standby condition and achieves, in ≤ 10 seconds, voltage ≥ 3793 V and frequency ≥ 58.8, and after steady state conditions are reached, maintains voltage ≥ 4000 V and ≤ 4400 V and frequency ≥ 59.3 Hz and ≤ 60.5 Hz.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.8	------------------------------NOTE------------------------------ The automatic transfer of the unit power supply shall not be performed in MODE 1 or 2. -- Verify automatic and manual transfer of unit power supply from the normal offsite circuit to the alternate offsite circuit.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.9	------------------------------NOTE------------------------------ A single test at the specified Frequency will satisfy this Surveillance for both units. -- Verify each DG rejects a load greater than or equal to its associated single largest post-accident load, and: a. Following load rejection, the frequency is ≤ 64.5 Hz; b. Within 4.5 seconds following load rejection, the voltage is ≥ 3760 V and ≤ 4560 V, and after steady state conditions are reached, maintains voltage ≥ 4000 V and ≤ 4400 V; and c. Within 6 seconds following load rejection, the frequency is ≥ 59.3 Hz and ≤ 60.5 Hz.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.10	------------------------------NOTES------------------------------ A single test at the specified Frequency will satisfy this Surveillance for both units. -- Verify each DG does not trip and voltage is maintained ≤ 4560 V during and following a load rejection of ≥ 4000 kW.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.11	------------------------------NOTES------------------------------ 1. All DG starts may be preceded by an engine prelube period. 2. This SR shall be performed for each DG on a rotational test basis and for each 4.16 kV ESS bus at the specified FREQUENCY. 3. This Surveillance shall not be performed in MODE 1, 2, or 3. --	
	Verify on an actual or simulated loss of offsite power signal: a. De-energization of 4.16 kV ESS buses; b. Load shedding from 4.16 kV ESS buses; and c. DG auto-starts from standby condition and: 1. energizes permanently connected loads in ≤ 10 seconds, 2. energizes auto-connected shutdown loads through individual load timers, 3. maintains steady state voltage ≥ 4000 V and ≤ 4400 V, 4. maintains steady state frequency ≥ 59.3 Hz and ≤ 60.5 Hz, and 5. supplies permanently connected loads for ≥ 5 minutes.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.8.1.12	-------------------------------NOTES------------------------------- 1. All DG starts may be preceded by an engine prelube period. 2. DG E, when not aligned to the Class 1E distribution system, may satisfy this SR for both units by performance of SR 3.8.1.12.a, b and c using the test facility to simulate a 4.16 kV ESS bus. SR 3.8.1.12.d and e may be satisfied with either the normally aligned DG or DG E aligned to the Class 1E distribution system. ---	
	Verify on an actual or simulated Emergency Core Cooling System (ECCS) initiation signal, each DG auto-starts from standby condition and: a. In ≤ 10 seconds after auto-start achieves voltage ≥ 3793 V, and after steady state conditions are reached, maintains voltage ≥ 4000 V and ≤ 4400 V; b. In ≤ 10 seconds after auto-start achieves frequency ≥ 58.8 Hz, and after steady state conditions are reached, maintains frequency ≥ 59.3 Hz and ≤ 60.5 Hz; c. Operates for ≥ 5 minutes; d. Permanently connected loads remain energized from the offsite power system; and e. Emergency loads are energized or auto-connected through the individual load timers from the offsite power system.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.13	-------------------------------NOTES------------------------------- 1. A single test at the specified Frequency will satisfy this Surveillance for both units. 2. DG E, when not aligned to the Class 1 E distribution system, may satisfy this SR for both units by using a simulated ECCS initiation signal. --- Verify each DG's automatic trips are bypassed on actual or simulated loss of voltage signal on the 4.16 kV ESS bus concurrent with an actual or simulated ECCS initiation signal except: a. Engine overspeed; and b. Generator differential current; and c. Low lube oil pressure.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.14	-------------------------------NOTES------------------------------- 1. Momentary transients outside the load ranges do not invalidate this test. 2. A single test at the specified Frequency will satisfy this Surveillance for both units. 3. DG E, when not aligned to the Class 1E distribution system may satisfy this SR by using the test facility. --- Verify each DG operates for ≥ 24 hours: a. For ≥ 2 hours loaded ≥ 4400 kW and ≤ 4700 kW for DGs A through D and ≥ 5000 kW and ≤ 5500 kW for DG E; and b. For the remaining hours of the test loaded ≥ 3600 kW and ≤ 4000 kW for DGs A through D and ≥ 4500 kW and ≤ 5000 kW for DG E.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.3	-------------------------------NOTES------------------------------- 1. DG loading may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one DG at a time. 4. This SR shall be preceded by and immediately follow, without shutdown, a successful performance of SR 3.8.1.7. 5. DG E, when not aligned to the Class 1 E distribution system, may satisfy this SR using the test facility. 6. A single test will satisfy this Surveillance for both units if synchronization is to the 4.16 kV ESS bus for Unit 1 for one periodic test and synchronization is to the 4.16 kV ESS bus for Unit 2 for the next periodic test. However, if it is not possible to perform the test on Unit 2 or test performance is not required per SR 3.8.2.1, then the test shall be performed synchronized to the 4.16 kV ESS bus for Unit 1. --	
	Verify each DG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ 3600 kW and ≤ 4000 kW.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.4	Verify each engine mounted day tank fuel oil level is ≥ 420 gallons for DG A-D and ≥ 425 gallons for DG E.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.5	Check for and remove accumulated water from each engine mounted day tank.	In accordance with the Surveillance Frequency Control Program

Amendment No. 262 Renewed License No. NPF-22

1. The U.S. Nuclear Regulatory Commission (NRC or the Commission) has found that:

1. The application for the amendment filed by Susquehanna Nuclear, LLC, dated May 26, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 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, 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. NPF-22 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 262, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. Susquehanna Nuclear, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days.

James G. Danna, Chief Plant Licensing Branch I Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation

ATTACHMENT TO LICENSE AMENDMENT NO. 262 SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2 RENEWED FACILITY OPERATING LICENSE NO. NPF-22 DOCKET NO. 50-388

Replace the following page of the Renewed Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

REMOVE Page 3 INSERT Page 3

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

REMOVE 3.8-5 INSERT 3.8-5

3. 3) Susquehanna Nuclear, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, posses, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed neutron sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;
4. (4) Susquehanna Nuclear, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, posses, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and
5. (5) Susquehanna Nuclear, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission nor or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

1. (1) Maximum Power Level

   Susquehanna Nuclear, LLC is authorized to operate the facility at reactor core power levels not in excess of 3952 megawatts thermal in accordance with the conditions specified herein. The preoperational tests, startup tests and other items identified in License Conditions 2.C.(20), 2.C.(21), 2.C.(22), and 2.C.(23) to this license shall be completed as specified.

2. (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 262, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. Susquehanna Nuclear, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

   For Surveillance Requirements (SRs) that are new in Amendment 151 to Facility Operating License No. NPF-22, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 151. For SRs that existed prior to Amendment 151, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of

   Amendment 151.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
G. One or more offsite circuits and two or more required DGs inoperable. <u>OR</u> One required DG and two offsite circuits inoperable.	G.1	Enter LCO 3.0.3.	Immediately
H. Manual synchronization circuit inoperable.	H.1	Restore manual synchronization circuit to OPERABLE status.	14 days

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 280 TO
RENEWED FACILITY OPERATING LICENSE NO. NPF-14
AND AMENDMENT NO. 262 TO
RENEWED FACILITY OPERATING LICENSE NO. NPF-22 SUSQUEHANNA NUCLEAR, LLC
ALLEGHENY ELECTRIC COOPERATIVE, INC. SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 DOCKET NOS. 50-387 AND 50-388

INTRODUCTION

By letter dated May 26, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20148L497), Susquehanna Nuclear, LLC (the licensee) requested changes to the technical specifications (TSs) for Susquehanna Steam Electric Station (Susquehanna), Units 1 and 2. The proposed amendments would revise TS 3.8.1, "AC [Alternating Current] Sources – Operating." Specifically, the proposed amendments would create a new TS action for an inoperable manual synchronization circuit requiring restoration within 14 days.

The licensee stated that the proposed amendments are necessary to reduce the potential for an unnecessary dual-unit shutdown. Based on the configuration of the AC power sources at Susquehanna, an inoperable manual synchronization circuit currently results in entry into Limiting Condition for Operation (LCO) 3.0.3 related to a condition when an LCO is not met and the associated actions are not met, or an associated action is not provided. For Susquehanna, Units 1 and 2, failure to comply with TS 3.8.1 requirements would result in dual-unit shutdown, which is not commensurate with the risk associated with having an inoperable manual synchronization circuit.

2.0 REGULATORY EVALUATION

The NRC staff applied the following U.S. Nuclear Regulatory Commission (NRC, the Commission) requirements to evaluate the license amendment request (LAR).

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, “Technical specifications,” requires, in part, that the operating license of a nuclear production facility include TSs.

Section 50.36(c)(2) of 10 CFR states that LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility, and when LCOs are not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the LCO can be met.

Section 50.36(c)(3) of 10 CFR requires that the TSs include surveillance requirements (SRs), which are requirements “relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting condition for operation will be met.”

Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50 (GDC), Criterion 17, “Electric power systems,” requires, in part, that an onsite electric power system and an offsite electric power system be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that
(1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences, and (2) the core is cooled, and containment integrity and other vital functions are maintained in the event of postulated accidents. The onsite electric power supplies, including the batteries and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions, assuming a single failure. GDC 17 provides, in part, the basis for the TS LCOs for the plant offsite and onsite electrical power systems.

GDC 18, “Inspection and testing of electric power systems,” requires, in part, that electric power systems important to safety be designed to permit appropriate periodic inspection and testing of important areas and features to assess the continuity of the systems and the condition of their components. GDC 18 also requires, in part that the systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation.

1. 3.0 TECHNICAL EVALUATION
2. 3.1 System Design and Operation

According to the Susquehanna Updated Final Safety Analysis Report, Chapter 8, there are four 4.16 kilovolt (kV) safety-related buses for Units 1 and 2. The Susquehanna electrical design differs from the typical dual-unit site in that the five diesel generators (DGs) and the four safety-related buses are shared between the two units instead of two DGs and associated buses being assigned to each unit.

Electric power from the offsite power sources to the onsite distribution system is provided by two physically separated 230 kV transmission lines. Two startup transformers (T-10 and T-20) step down the voltage from 230 kV to 13.8 kV for onsite distribution for both units. The 13.8 kV distribution system provides a preferred and an alternate source of AC electric power via emergency safeguard system (ESS) transformers to all safety-related loads through the

Class IE 4.16 kV distribution system.

The Class 1E power system has four 13.8/4.16 kV ESS transformers, T-101, T-111, T-201, and T-211, that can power the respective safety-related buses. Each bus has the capability to be

supplied from its preferred source (transformer) or the alternate source. According to the Updated Final Safety Analysis Report, the Class 1E AC system is divided into four load group channels per unit (load group channels A, B, C, and D) such that any combination of three-out-of-four load groups has the capability of supplying the minimum required safety loads of that unit.

Four independent DGs designated A, B, C, and D shared between the two units provide emergency power for each of the four ESS 1E AC load groups in each unit in the event of total loss of the preferred and alternate supplies. A spare Class 1E DG (E-diesel) is provided, which can be manually aligned as a replacement for any one of the other four DGs without violating the independence of the redundant safety-related load groups. In the event of a loss-of-offsite-power (LOOP), the engineered safety feature (ESF) loads are automatically connected to the DGs in sufficient time to support safe reactor shutdown and to mitigate the consequences of a design-basis accident (DBA) such as loss-of-coolant accident. DGs A, B, C, and D are each rated for continuous operation at 4,000 kilowatts (kW) 0.8 power factor and 4,700 kW for 2,000-hour operation. DG E is rated for continuous operation at 5,000 kW
0.8 power factor and 5,500 kW for 2,000-hour operation. The capacity of any DG aligned to the specific safety buses, assuming one of the aligned diesels fails, is sufficient to operate the ESF loads of one unit and those systems required for concurrent safe shutdown of the second unit.

At the 4 kV ESF power distribution subsystem, a three-way transfer system is provided to enable the ESF loads to connect to either of the two offsite power sources or to the standby DGs. This switch provides the means for a manual live bus transfer through a synchronizing device or allows a DG to be tied with any one of the two offsite sources for an indefinite time under test condition.

LAR Section 2.1.4 describes the design features for the manual synchronization circuit, in part, as follows:

The manual synchronization circuit provides a means to switch the power supply to an energized electrical circuit from one source to another for the 13.8 kV buses and the 4.16 kV buses, as well as tie the DGs to the 4.16 kV buses. There is one manual synchronization circuit shared between the two units; it is comprised of a synchronization bus, a bus differential voltmeter, a synchroscope, two white lights, and 37 synchronizing selector switches (referred to as “sync selector switches” hereafter). Eight of the sync selector switches are for the DGs, 16 are for the primary and alternate offsite power supply to the 4.16 kV buses, and the remaining 13 are for the 13.8 kV buses. In order to manually synchronize one power supply to another, the desired hand switch is taken to the ON position. This provides power from the bus (i.e., the “Running Voltage”), which is compared to the source voltage (“Incoming Voltage”) with ground as a reference point. The synchroscope, two white lights, and the bus differential voltmeter provide indication to operators as to how well the two sources are matched in frequency, phase angle, and voltage. When the sources are synchronized, the operator manually closes the breaker for the new power source. Because the sync selector switches share the synchronization bus, only one sync selector switch can be turned on at a time.

The sync selector switches can be used to:

Transfer the 4.16 kV Emergency Safeguard System (ESS) bus power source from the preferred power supply to the alternate power supply, or from the alternate power supply to the preferred power supply.

Manually connect DGs A-D (or E, if substituted) to their corresponding 4.16 kV bus for DG testing purposes.

Restore offsite power source to an ESS bus (such as following a Loss of Offsite Power (LOOP)) if a DG was powering the bus. A de-energized ESS bus can be powered by offsite power without the use of the sync selector switches.

Transfer the 13.8 kV bus power source between startup transformers.

Transfer the 13.8 kV auxiliary buses between auxiliary transformers.

The sync selector switches are only utilized for manual transfers. The automatic transfer functions do not utilize any of the manual synchronization equipment. The ability to manually transfer the power source for a 4.16 kV or 13.8 kV bus is not assumed in any accident analysis. Restoration of the normal power source can be made without the manual synchronization circuit by de-energizing the bus. The ability to synchronize a DG to an energized bus is also not assumed in any accident analysis, but is needed to perform certain tests.

3.2 Current TS Requirements

LCO 3.8.1 requires, for each unit, that two offsite sources and four onsite DGs be operable in Modes 1, 2, and 3.

The following SRs are performed to demonstrate the operability requirements of the offsite power sources and the DGs. Although manual synchronization circuit is not described in the LCO, the manual synchronization circuit is functional to demonstrate completion of the SRs and operability of both onsite and offsite power systems.

SR 3.8.1.3 states, “Verify each DG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ 3600 kW and ≤ 4000 kW.”

SR 3.8.1.8 states, “Verify automatic and manual transfer of unit power supply from the normal offsite circuit to the alternate offsite circuit.”

SR 3.8.1.16 states: Verify each DG:

1. Synchronizes with offsite power source while loaded with emergency loads upon a simulated restoration of offsite power;
2. Transfers loads to offsite power sources, and
3. Returns to ready-to-load operation.

The licensee stated in LAR Section 2.2 that based on the configuration of the Class 1E alternating current (AC) power sources at Susquehanna, a failure of a single sync selector switch (and, therefore, the entirety of the manual synchronization circuit) disables the capability to synchronize all four DGs and both offsite sources for both units. The inability to synchronize the DGs to an offsite source means that restoration of offsite power source to the ESS buses after recovery from a LOOP event cannot be achieved by parallel operation of the power sources. This results in non-compliance with the intent of Susquehanna TS SRs 3.8.1.8 and 3.8.1.16. Since Susquehanna TS 3.8.1 does not have a condition for an inoperable manual synchronization circuit, Required Action G.1 associated with Condition G, “One or more offsite circuits and two or more required DGs inoperable, OR one required DG and two offsite circuits inoperable,” requires entry into LCO 3.0.3 immediately. As all sync selector switches share a common synchronization bus for both units, the required entry into LCO 3.0.3 is applicable to both units. The failure of a sync selector switch does not impact the automatic transfer capability between offsite and onsite power sources assumed in the accident analyses.

3.3 Proposed TS Changes

TS 3.8.1 is revised to add a new Action H. Condition H will state, “Manual synchronization circuit inoperable.” The associated required action is to restore the manual synchronization circuit to an operable status with a completion time (CT) of 14 days.

3.4 NRC Staff Evaluation

In LAR Section 2.3, the licensee stated the reason for the proposed TS change. When the licensee performs verification of the transfer capabilities of the offsite power sources in
SR 3.8.1.8 and the DGs in SR 3.8.1.16, the appropriate sync selector switch is placed in the ‘on’ position. This is performed by rotating a keyed switch in the control room. In LAR Section 2.2, the licensee stated that SR 3.8.1.3 also requires verification that “each DG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ 3600 kW and ≤ 4000 kW.” However, the DG is connected to the ESS bus to support performance of the test, and that connection cannot be performed without the manual synchronization circuit. If the manual synchronization circuit is not available, the SR cannot be performed, and the capability to restore offsite power following a LOOP event is not assured.

The licensee stated that Susquehanna has identified material degradation of the plastic within the key switch as a potential failure mechanism for the sync selector switch, which has resulted in two failures since 2013. The failures resulted in the circuit remaining energized with the inability to be deenergized. There are 37 sync selector switches that share a common synchronization bus. A failure of one switch renders the manual transfer capability of the remaining 36 sync selector switches unavailable. However, the safety function of automatic transfers between offsite and onsite power sources is not impacted.

Based on its review of the DG and the 4.16 kV and 13.8 kV manual synchronization circuitry and automatic transfer circuitry, the NRC staff determined that manual synchronization circuitry and automatic transfer circuitry are independent, and the automatic transfer circuitry does not utilize the sync selector switches. Therefore, the staff concluded that the existing automatic transfer circuitry is not affected by an inoperable sync selector switch or a manual synchronization component.

The AC sources are designed to permit inspection and testing of important areas and features, especially those that have a standby function, in accordance with 10 CFR Part 50, Appendix A,

GDC 18. Periodic component tests are supplemented by extensive functional tests during refueling outages under simulated accident conditions. The NRC staff noted that the manual synchronization circuit, including the sync selector switches, is needed to perform:

- • TS SR 3.8.1.8 regarding transfer capabilities between normal and alternate offsite power sources,
- • TS SR 3.8.1.3 to verify each DG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ 3,600 kW and ≤ 4,000 kW transfer capabilities between the DGs and offsite power sources, and
- • TS SR 3.8.1.16 to verify each DG synchronizes with offsite power source while loaded with emergency loads upon a simulated restoration of offsite power transfers loads to offsite power sources and returns each DG to ready-to-load operation.

  The NRC staff reviewed the licensee's proposed request for the addition of a Condition H required action and the CT while the manual synchronization circuit is inoperable. The required action is to restore the manual synchronization circuit to an operable status within a CT of
  14 days. The NRC staff noted that the proposed change provides additional time for the plant to operate with the manual synchronization circuit inoperable. The licensee has stated that during the time that the manual synchronization circuit is inoperable, the automatic transfer functions of all Class 1E AC sources remain unaffected, and the power sources will be available in a manner commensurate with assumptions in accident analyses. Therefore, the NRC staff finds that the Condition H required action is acceptable.

  The NRC staff reviewed the licensee's analysis for the proposed CT duration of 14 days for the Condition H required action. The licensee stated in Section 3.1 of the LAR that TS SR 3.8.1.3 is typically performed once per week on successive

DGs such that each of the required DGs is typically tested once every 28 days, although the current frequency of SR 3.8.1.3 is 31 days. In order to perform the monthly DG runs currently required by SR 3.8.1.3, operators manually synchronize the DG to the power grid by operating the sync selector switch. If the sync selector switch becomes inoperable during the test, the surveillances for the remaining DGs cannot be performed.

The licensee further states that based on Susquehanna scheduling surveillance practices with an allowable delay of 25 percent in accordance with SR 3.0.2, the subsequent performance of SR 3.8.1.3 would be required within 17.75 days after which the DG scheduled to be tested would be declared inoperable. Based on this analysis, Susquehanna is proposing a 14-day CT for an inoperable manual synchronization circuit. The NRC staff finds the 14-day CT duration acceptable because the current automatic transfer functions of both onsite and offsite power system design-basis safety functions are not adversely impacted by the inoperable manual synch switch. The 14-day period also provides the licensee with sufficient flexibility in scheduling and repairing a failed switch and maintaining conformance with surveillances associated with the SR 3.8.1.3 frequency for each DG. The relatively low risk associated with an inoperable manual synchronization circuit, compared to entry into LCO 3.0.3 for both units and the associated shutdown risk, is acceptable.

The NRC staff concludes that the LAR meets the regulatory requirements as discussed in Section 2.0 above. In the event of a loss of preferred power, the ESF electrical loads are automatically connected to the DGs in sufficient time to provide for safe reactor shutdown and to

mitigate the consequences of a DBA such as a loss-of-coolant accident. The 14-day CT considers the operability of the AC sources and reasonable time for repairs.

The NRC staff has determined that the licensee’s request to add Condition H to the TS 3.8.1 required action and CT to restore the manual synchronization circuit to an operable status is consistent with the requirements specified in 10 CFR 50.36(c)(2) and (c)(3). Therefore, the staff concludes that the licensee’s proposed change complies with existing regulations.

4.0 STATE CONSULTATION

In accordance with the Commission’s regulations, the Pennsylvania State official was notified of the proposed issuance of the amendments on February 26, 2021. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change SRs. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration (85 FR 45449; July 28, 2020). There was one public comment on such finding, but it was not relevant to the LAR. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 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.

Principal Contributor: R. Mathew Date: March 10, 2021

April 23, 2021 – Letter from James G. Danna, Chief Plant Licensing Branch I Division of Operator 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 UNIT NO. 2 – APPROVAL OF ONE-TIME ALTERNATIVE TO FLAW CHARACTERIZATION AND REMOVAL REQUIREMENTS FOR N-16A NOZZLE (EPID L-2020-LLR-0144)

By letter dated November 4, 2020, as supplemented by letter dated November 24, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML20309B020 and ML20329A345, respectively), Exelon Generation Company, LLC (Exelon, the licensee) submitted a proposed one-time alternative to certain requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI in Articles IWB-3000 and IWA-4000 Peach Bottom Atomic Power Station, Unit No. 2 (Peach Bottom 2).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) paragraph 50.55a(z)(2), the licensee requested to use proposed alternative I5R-14 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.

On November 6, 2020 (ADAMS Accession No. ML20314A028), the U.S. Nuclear Regulatory Commission (NRC) staff verbally authorized the use of alternative request I5R-14. In its verbal authorization, the NRC staff determined that the proposed alterative to repair the degraded reactor vessel instrument penetration nozzle N-16A by a half-nozzle method is technically justified and provides reasonable assurance of structural integrity and leak tightness for the duration of operating cycle 24, which is scheduled to end in the fall of 2022.

The NRC staff has reviewed the subject request and 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).

The NRC authorizes the use of the proposed alternative in I5R-14 for Peach Bottom 2, for the duration of one operating cycle. The alternative provides reasonable assurance of structural integrity and leak tightness of the reactor vessel instrument penetration nozzle N-16A. All other ASME Code, Section XI, requirements for which relief was not specifically requested and approved in this relief request remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

If you have any questions concerning this matter, 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
RELIEF REQUEST I5R-14
ALTERNATIVE TO FLAW CHARACTERIZATION AND REMOVAL REQUIREMENTS FOR N-16 NOZZLE
EXELON GENERATION COMPANY, LLC
PEACH BOTTOM ATOMIC POWER STATION UNIT NO. 2
DOCKET NO. 50-277

INTRODUCTION

By letter dated November 4, 2020 as supplemented by letter dated November 24, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML20309B020 and ML20329A345, respectively), Exelon Generation Company, LLC (Exelon, the licensee) submitted a proposed one-time alternative to the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI in Articles IWB-3000 and IWA-4000 Peach Bottom Atomic Power Station, Unit No. 2 (Peach Bottom 2).

Specifically, the licensee requested a one-time relaxation of certain flaw characterization and removal requirements in Articles IWB-3000 and IWA-4000 of ASME Code, Section XI. Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) paragraph 50.55a(z)(2), the licensee requested to use the proposed alternative in Relief Request I5R-14 on the basis that performing the repair in accordance with the ASME Code, Section XI would result in an increased radiological exposure, and there exists a potential risk of loose parts or foreign materials accidentally getting into the reactor vessel (RV) during the ASME Code repair.

On November 6, 2020 (ADAMS Accession No. ML20314A028), the NRC staff verbally authorized the use of alternative request I5R-14. In its verbal authorization, the NRC staff determined that the proposed alternative to repair the degraded RV instrument penetration nozzle N-16A by a half-nozzle method is technically justified and provides reasonable assurance of structural integrity and leak tightness for the duration of operating cycle 24, which is scheduled to end in the fall of 2022. The verbal authorization documentation provides a summary of the NRC staff evaluation for this proposed alternative. This safety evaluation provides the details of the NRC staff review of proposed alternative I5R-14.

2.0 REGULATORY EVALUATION

The regulation at 10 CFR 50.55a(z) states, in part, that alternatives to the requirements of paragraphs (b) through (h) of 10 CFR 50.55a or portions thereof may be used when authorized by the Director, Office of Nuclear Reactor Regulation. A proposed alternative must be submitted and authorized prior to implementation. The applicant or licensee must demonstrate that:

1. (1) *Acceptable level of quality and safety*. The proposed alternative would provide an acceptable level of quality and safety; or
2. (2) *Hardship without a compensating increase in quality and safety*. Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

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 use of the proposed alternative.

1. 3.0 TECHNICAL EVALUATION
2. 3.1 ASME Code Component(s) Affected

The affected component is the RV instrument penetration nozzle N-16A. Specifically, the applicable ASME Code, Section XI, 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.

3.2 Applicable Code Edition and Addenda

The ASME Code, Section XI, 2013 Edition with no Addenda is the Code of Record for the fifth 10-year inservice inspection (ISI) interval. The code of construction for RV is the 1965 Edition through Winter 1965 Addenda of ASME Code, Section III.

3.3

Applicable Code Requirement

Flaw characterization requirements of IWB-3420 and IWB-3620 Flaw removal requirements of IWA-4412 and IWA-4611 Analytical flaw evaluation requirements of IWB-3600

The ASME Code requirements applicable to this request originate in Articles IWB-3000 and

IWA-4000 of Section XI, which include:

• • • •

ASME Code Case N-749, “Alternative Acceptance Criteria for Flaws in Ferritic Steel Components Operating in the Upper Shelf Temperature Range Section XI, Division 1.” ASME Code Case N-749 has been incorporated by reference into 10 CFR 50.55a via inclusion in Regulatory Guide (RG) 1.147, Revision 19, “Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1,” dated October 2019 (ADAMS Accession

No. ML19128A244), with a condition (i.e., In lieu of the code case defined upper shelf transition temperature T_c, the NRC-defined $T_c = 154.8\ ^oF + 0.82\ RT_{NDT}$ shall be used. In addition, the NRC defines temperature $T_{c1} = 95.36\ ^oF + 0.703\ RT_{NDT}$ which the linear elastic fracture mechanics (LEFM) must be applied. Between the NRC-defined T_{c1} and T_c, although the fracture mode is in transition from LEFM to elastic plastic fracture mechanics (EPFM), users should consider whether it is appropriate to apply the EPFM. Alternatively, a different T_c value may be used if it can be justified by the plant-specific Charpy curves).

- ASME Code Case N-638-7, “Similar and Dissimilar Metal Welding Using Ambient Temperature Machine GTAW [Gas Tungsten Arc Welding] Temper Bead Technique, Section XI,” has been incorporated by reference into 10 CFR 50.55a via inclusion in RG 1.147, Revision 19, with a condition (i.e., Demonstration of ultrasonic examination of the repaired volume is required using representative samples that contain construction-type flaws).

3.4 Proposed Alternative

During performance of a routine system leakage test in the refueling outage P2R23, the licensee discovered a leak at the RV instrument penetration nozzle N-16A. The results of combined and spatially-correlated internal and external visual and ultrasonic examinations suggested that the most probable cause of the external leakage identified in nozzle N-16A is a radial-axial-oriented intergranular stress-corrosion cracking (IGSCC) flaw which initiated in the Alloy 182 J-groove weld and propagated through the J-groove weld until it reached a depth where a leak path in the annulus between the nozzle and RV penetration existed. The licensee proposed to repair the degraded 2-inch nozzle N-16A using a half-nozzle method to restore the pressure boundary.

3.5 Basis for Use

To support its repair option, the licensee proposed an alternative to 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. In addition, the licensee provided for the NRC review the following information to demonstrate that the structural integrity and leak tightness of repaired nozzle N-16A will be maintained for the duration of one operating cycle.

3.6

The licensee's request is applicable to Peach Bottom 2 for the duration of one operating cycle.

An evaluation of the repair design, welding, and nondestructive examination (NDE) 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 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.

Duration of Proposed Alternative

4.0 NRC STAFF EVALUATION

The NRC staff evaluated alternative request I5R-14 pursuant to 10 CFR 50.55a(z)(2). The NRC staff focused on whether compliance with the specified requirements of 10 CFR 50.55a(g), or portions thereof, would result in hardship or unusual difficulty, without a compensating increase in the level of quality and safety.

Structural Integrity

In its evaluation, the NRC staff focused on two aspects of the licensee's technical basis which include the half-nozzle repair and examination process to restore the pressure boundary, and the plant-specific analysis (i.e., evaluation of the flaws left in service in the original J-groove weld and corrosion assessment of RV low-alloy steel (LAS)) to demonstrate that the structural integrity of repaired RV instrument penetration nozzle N-16A will be maintained for the duration of one operating cycle. Details of the licensee's plant-specific analyses are documented in non-proprietary Attachments 5, 6, and 7 of I5R-14 dated November 24, 2020 (ADAMS Accession No. ML20329A345).

Half-Nozzle Repair and Examination Process

To restore the pressure boundary, the licensee utilized a "half-nozzle" repair method where a portion of the existing degraded Alloy 600 nozzle N-16A assembly at or near the outside diameter (OD) surface of the RV was replaced with an Alloy 690 nozzle. The repair entailed a temper bead weld buildup (i.e., weld pad) on the OD of the RV using Alloy 52M filler metal in accordance with ASME Code Case N-638-7, and a final partial penetration manual welding between Alloy 690 nozzle and Alloy 52M weld pad using Alloy 52M filler metal. Alloy 690/52M materials are known to be resistant to IGSCC. The remnant of the original Alloy 600 nozzle and Alloy 182 partial penetration attachment weld that contained the flaw were left in place. The sketch of weld pad and other details of repair are documented in the figure in Enclosure 2 of I5R-14. The NRC staff finds that the licensee's half-nozzle repair method is acceptable because it relocates the pressure boundary from inside diameter to OD surface of the RV shell which includes Alloy 52M partial penetration J-groove weld joining Alloy 690 nozzle to Alloy 52M weld pad and the welding and design analysis comply with the ASME Code requirements. In addition, the NRC staff verified that the licensee performed the NDE required as a part of the half-nozzle repair to ensure compliance with the ASME Code, Sections III and XI, and ASME Code Case N-638-7 with the condition in RG 1.147, Revision 19. A brief summary of the repair and NDE performed is as follows:

- • Install foreign material exclusion sealing plug, detach piping near reducing coupling, cut the existing Alloy 600 nozzle outboard of the RV, grind the nozzle flush with the RV shell OD surface, and attach the capacitor discharge studs (welding and boring tools) to RV. Then, perform surface and volumetric examinations of the RV shell OD surface in preparation for installing Alloy 52M weld pad.
- • Install a weld dam to accommodate for depositing weld pad, deposit the Alloy 52M weld pad in accordance with ASME Code Case N-638-7, perform post weld grinding of the weld pad, and conduct dimensional inspection of weld pad. Then, perform surface and ultrasonic examinations of the weld pad upon completion of 48-hour hold time.
- • Remove weld dam, perform final machining of the weld pad bore, perform a dimensional measurement of the final bore. Then, perform surface examination of the final bore.

- • Machine replacement Alloy 690 nozzle. Then, perform visual and surface examinations of the replacement Alloy 690 nozzle.
- • Weld new reducing coupling to nozzle. Then, perform visual and surface examinations of the reducing coupling-to-nozzle weld.
- • Machine J-groove bevel in the weld pad. Then, perform visual and surface examinations of the J-groove bevel.
- • Perform installation and welding of the replacement Alloy 690 nozzle. Then, perform a progressive surface examination of J-groove weld joining the replacement Alloy 690 nozzle to the weld pad.
- • Remove capacitor discharge studs attached to RV. Then, perform surface examination of RV at the capacitor discharge stud attachment locations.
- • Attach piping to new reducing coupling and remove foreign material exclusion sealing plug.

Based on the above, the NRC staff finds that the licensee met the NDE requirements of ASME Code, Sections XI and III, and ASME Code Case N-638-7 with the condition in RG 1.147, Revision 19, as applicable.

Plant Specific Analysis

To demonstrate reasonable assurance of RV structural integrity for one operating cycle following the nozzle repair, the licensee used a plant-specific analytical evaluation based on combination of LEFM and EPFM in accordance with the

ASME Code, Section XI requirements, with the assumption that the entire as-left Alloy 182 J-groove attachment weld of Alloy 600 nozzle N-16A is completely cracked and the crack will potentially propagate into the RV LAS base material. The NRC staff finds that the licensee's assumption is conservative on the basis that any "as-left" flaws in the Alloy 182 J-groove weld cannot be characterized with reasonable confidence by the currently available NDE techniques, and this postulated initial flaw bounds any actual indications that have existed in the attachment weld of nozzle N-16A. The licensee further postulated that the preferential direction for crack propagation is radial-axial relative to the nozzle and RV because the hoop stress is determined to be dominant at the J-groove weld location. The stress intensity factors along the postulated crack front were calculated for pressure, residual stress, steady stress thermal and transient conditions. In its evaluation of the licensee's plant-specific analyses, the NRC staff verified that:

- • The licensee used a bounding crack growth rate data in BWRVIP-60-A, "BWR Vessel and Internals Project, Evaluation of Stress Corrosion Crack Growth in Low Alloy Steel Vessel Materials in the BWR Environment," to determine cracking into the Peach Bottom's LAS RV material from the service-related degradation. The NRC staff finds that BWRVIP-60-A is an NRC acceptable methodology to use for determination of stress-corrosion cracking (SCC) in RV LAS in BWR environment, thus is adequate for this analysis.
- • The licensee utilized a finite element model to obtain the applied stresses in the RV shell at the nozzle J-groove weld location based on bounding design basis transient conditions of normal/upset condition (heat-up/cool-down, loss-of-pump, and single relief)

and emergency/faulted condition (overpressure), and to perform fracture mechanics analysis. The licensee's modeling included the RV LAS base material, remnant of original Alloy 600/182 nozzle and attachment weld, stainless steel cladding, Alloy 52M weld pad, Alloy 690/52M replacement nozzle and attachment weld. Therefore, the NRC staff finds the licensee's finite element model acceptable because appropriate materials, plant-specific configurations, and loading conditions were used.

- • In addition to the thermal and pressure stresses, the licensee's analysis included the welding residual stress (WRS) that contributes to the crack driving force. For this analysis, the licensee assumed the magnitude of WRS based on room temperature yield strength of the Alloy 182 J-groove weld material which is reduced by the compressive stress in the RV LAS shell. The reduction of WRS would minimize the potential for the crack, if it exists at the interface between weld and RV, to grow into the RV shell. The NRC staff finds that the licensee's WRS is adequate for this analysis because: (a) the RV was post weld heat treated following welding during fabrication which reduces the WRS, (b) the licensee's calculation has shown that magnitude of WRS is reduced by the compressive stress in the RV shell, and (c) similar WRS estimation was previously accepted by the NRC in Quad Cities Nuclear Power Station's request dated April 6, 2012 (ADAMS Accession No. ML12100A012), and Limerick Generating Station's request dated May 15, 2017 (ADAMS Accession No. ML17135A423).

- • In its fracture mechanics analysis, the licensee utilized the screening criteria in ASME Code Case N-749 with the condition in RG 1.147, Revision 19, to determine acceptability of flaw in the RV LAS when the metal temperature is in the upper shelf range. The screening criteria is: (a) use EPFM method of analysis if the metal temperature exceeds the NRC-defined temperature T_c, (b) use LEFM method of analysis if the metal temperature drops below the NRC-defined temperature T_{c1}, and (c) for metal temperature between T_c and T_{c1}, assess suitability of using EPFM since the fracture mode is in transition from LEFM to EPFM. The NRC staff verified that the ASME Code required acceptance criteria for LEFM and EPFM are satisfied; therefore, the repaired instrumentation nozzle N-16A is acceptable for one operating cycle.

  Corrosion Evaluation of RV LAS Base Material

  In its review, the NRC staff assessed the licensee's corrosion analysis of the portion of the Peach Bottom 2's RV LAS base metal exposed to the boiler-water reactor (BWR) water environment as a result of the half-nozzle repair of nozzle N-16A. The possible corrosion mechanisms for LAS in BWR environment are known to be general corrosion, galvanic corrosion, crevice corrosion, and SCC. The licensee calculated the general corrosion rate on a per year basis for LAS based on bounding laboratory testing data and showed that the total surface corrosion of LAS at the exposed location for one operating cycle following nozzle repair would be very low. The licensee also addressed the crevice corrosion, galvanic corrosion, and SCC susceptibly of LAS, and determined that their rates are not significant for one operating cycle following the nozzle repair. The NRC staff finds the licensee's assertion that the SCC is not a concern for one operating cycle acceptable because of low corrosion rate of the LAS as well as the implementation of industry standard corrosion mitigate program at Peach Bottom 2 (e.g., on-line noble metal chemical addition with hydrogen water chemistry).

Hardship Justification

The NRC staff 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, and there exists a potential risk of loose parts or foreign materials accidentally getting into the RV during the ASME Code repair. Therefore, the NRC staff determines that concerns from the foreign material falling into the RV and an as low as is reasonably achievable criteria for radiological exposure support the licensee's hardship justification.

In summary, the NRC staff finds the licensee's plant-specific analysis acceptable because a conservative initial flaw is assumed, and the flaw evaluation has demonstrated that the initial flaw will not grow to an unacceptable depth into the RV LAS base material over one operating cycle. Furthermore, the impact on RV LAS from exposure to BWR water environment is determined to be low. As a result, the staff finds that the licensee's proposed alternative provides reasonable assurance of the RV structural integrity for the duration of one operating cycle.

5.0 CONCLUSION

As set forth above, the NRC staff has determined that complying with the specified requirements described in the licensee's relief request I5R-14 would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The proposed alternative provides reasonable assurance of structural integrity and leak tightness of the RV instrument penetration nozzle N-16A. The NRC staff concludes that the licensee has adequately addressed the regulatory requirements set forth in 10 CFR 50.55a(z)(2). Therefore, the NRC staff authorizes use of proposed alternative I5R-14 at Peach Bottom 2, for the duration of one operating cycle.

All other ASME Code, Section XI, requirements for which an alternative was not specifically requested and authorized remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: A. Rezai, NRR Date: April 23, 2021

<u>May 4, 2021</u> – Email from Sujata Goetz Project Manager, Susquehanna Steam Electric Station Nuclear Regulatory Commission to Shane Jurek with subject of Acceptance Review for Susquehanna TSTF-505 (EPID: L-2021-LLA-0062)

By letter dated April 8,2021 (Agencywide Document and Access Management System (ADAMS) Accession No. ML21098A206), Susquehanna Nuclear, LLC submitted a license amendment request for Susquehanna Unit 1 and Unit 2. The license amendment would modify TS 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 Initiative 4b" (ADAMS Accession No. ML18183A493).

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 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.

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. 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 2100 hours to complete. The NRC staff expects to complete this by end of August 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, 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.

If you have any questions, please contact me.

May 5, 2021 – Letter from Jonathan E. Greives, Chief Reactor Projects Branch 4 Division of Operating Reactor Safety to Brad Berryman President and Chief Nuclear Officer Susquehanna Nuclear, LLC with subject of SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 – INTEGRATED INSPECTION REPORT 05000387/2021001 AND 05000388/2021001

On March 31, 2021, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Susquehanna Steam Electric Station, Units 1 and 2. On April 29, 2021, the NRC inspectors discussed the results of this inspection with Mr. Derek Jones, Plant Manager, 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 Susquehanna Steam Electric Station, Units 1 and 2.

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 Susquehanna Steam Electric Station, Units 1 and 2.

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: 05000387 and 05000388
License numbers: NPF-14 and NPF-22
Report numbers: 05000387/2021001 and 05000388/2021001

Enterprise Identifier: I-2021-001-0084
Licensee: Susquehanna Nuclear, LLC
Facility: Susquehanna Steam Electric Station, Units 1 and 2

Location: Berwick, PA 18603
Inspection Dates: January 1, 2021, to March 31, 2021

Inspectors: C. Highley, Senior Resident Inspector

M. Rossi, Resident Inspector
E. Eve, Senior Reactor Inspector

Approved by: Jonathan E. Greives, 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 Susquehanna Steam Electric Station, Units 1 and 2, 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

Criterion III, Design Control, for Failure to Install Quality Components in Battery Room Exhaust Fans			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000387, 05000388/2021001-01 Open/Closed	[H.5] - Work Management	71111.12
A Green, non-cited violation of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix B, Criterion III, Design Control, was identified by inspectors when inspection activities revealed that a non-dedicated, non-quality controlled belt was installed on the 'B' battery room exhaust fan.			

Additional Tracking Items

Type	Issue Number	Title	Report Section	Status
LER	05000387/2020-001-00	LER 2020-001-00 for Susquehanna Steam Electric Station, Unit 1, Automatic Reactor Scram Due to Main Turbine Trip	71153	Closed
LER	05000387/2020-001-01	LER 2020-001-01 for Susquehanna Steam Electric Station, Unit 1, Automatic Reactor Scram Due to Main Turbine Trip Caused by an Electrical Ground Path in the B Main Transformer	71153	Closed
LER	05000387/2020-003-00	LER 2020-003-00 for Susquehanna Steam Electric Station, Unit 1, Condition Prohibited by Technical Specifications Due to Inoperable Turbine Stop Valve and Turbine Control Valve Instrumentation	71153	Closed

PLANT STATUS

Unit 1 began the inspection period at or near rated thermal power. On January 8, 2021, the unit was down powered to 78.5 percent for rod sequence exchange/rod pattern adjustment. The unit was returned to 100 percent power on January 9, 2021. On February 5, 2021, the unit was down powered to 97 percent for rod pattern adjustment. The unit was returned to 100 percent power on February 5, 2021. On March 11, 2021, the unit was down powered to 78.8 percent for rod pattern adjustment supporting turbine valve testing. The unit was returned to 100 percent power on March 11, 2021. On March 18, 2021, the unit was down powered to 74.2 percent for scram testing/rod sequence exchange/rod pattern adjustment. The unit was returned to

100 percent power on March 21, 2021. On March 21, 2021, the unit was down powered to 88 percent for rod pattern adjustment. The unit was returned to 100 percent on March 21, 2021. On March 31, 2021, the unit was down powered to 97 percent for rod pattern adjustment. The unit was returned to 100 percent on March 31, 2021.

Unit 2 began the inspection period at or near rated thermal power. On January 3, 2021, the unit down powered to 82.5 percent for a rod pattern adjustment. The unit was returned to
97.5 percent power on January 3, 2021, and commenced a coast down to its next refueling outage U2RIO20. On March 21, 2021, the unit commenced a down power from

69 percent. The unit was at zero percent and shutdown on March 22, 2021, for refueling.

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 (IP Section 03.02) (1 Sample)

(1) The inspectors evaluated the adequacy of the overall preparations to protect risk-significant systems from impending severe weather of a Nor Easter snowstorm on January 31, 2021.

71111.04 - Equipment Alignment
Partial Walkdown (IP Section 03.01) (6 Samples)

The inspectors evaluated system configurations during partial walkdowns of the following systems/trains:

1. (1) Unit 2, emergency core cooling systems during a 'B' core spray system outage window on January 12, 2021
2. (2) Unit 1, high-pressure coolant injection and reactor core isolation cooling systems prior to bus 0A106 work on February 2, 2021
3. (3) Unit Common, 'B' and 'D' emergency diesel generators during bus 0A106 work on February 4, 2021
4. (4) Unit 1, high-pressure coolant injection during repair of reactor core isolation cooling valve F045 on February 10, 2021
5. (5) Unit 2, division 2 core spray on March 16, 2021
6. (6) Unit 2, division 1 residual heat removal system prior to establishment of shutdown cooling on March 21, 2021

71111.05
Fire Area Walkdown and Inspection (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) Unit 1, control structure upper relay room, 754-foot elevation, fire zone 0-27E, on February 3, 2021
2. (2) Unit 2, standby liquid control piping penetration room, 749-foot elevation, fire zone 2-5E, on March 9, 2021
3. (3) Unit 1, load center room, 779-foot to 799-foot elevations, fire zone 1-6B, on March 12, 2021
4. (4) Unit 2, core spray pump rooms for A and B, 645-foot elevation, fire zones 2-1A and 2-1B, on March 12, 2021
5. (5) Unit 1, 4kV load center rooms, 749-foot elevation, fire zones 1-5F and 1-5G, on March 17, 2021
6. (6) Unit 2, drywell, 704-foot to 779-foot elevations, fire zone 2-4F, on March 23, 2021

- Fire Protection

Fire Brigade Drill Performance (IP Section 03.02) (1 Sample)

(1) The inspectors evaluated the onsite fire brigade training and performance during an unannounced fire drill on January 30, 2021.

71111.08G - Inservice Inspection Activities (Boiling-Water Reactor)

Boiling-Water Reactor Inservice Inspection Activities - Nondestructive Examination (NDE) and Welding Activities (IP Section 03.01) (1 Sample)

(1) The inspectors verified that the reactor coolant system boundary, reactor vessel internals, risk-significant piping system boundaries, and containment boundary were appropriately monitored for degradation and that repairs and replacements were

appropriately fabricated, examined, and accepted by reviewing the following activities starting March 29, 2021:

03.01.a - NDE and Welding Activities

1. Visual Examination (VT-3) of Containment External Concrete Surfaces, NDE Report Nos. VT-21-006, VT-21-007, VT-21-008, and VT-21-009
2. Visual Examination (VT-3) of Reactor Core Isolation Cooling Turbine Internal Surface and Shaft Inspection for License Renewal, NDE Report No. BOP-VT-21-069
3. In-Vessel Visual Examination of the Steam Dryer and Top Guide Grid, NDE Report Nos. IVVI-21-03 and IVVI-21-04
4. Ultrasonic Examination of Reactor Recirculation Weld, VRRB313-10-C, NDE Report No. UT-21-004
5. Ultrasonic Examination of Reactor Water Cleanup Bottom Head Drain Expander, DBA-221-1-9652-X, NDE Report No. FAC-U2-21-058
6. Repair/Replacement Activity

a. Replace 250F047 Reactor Core Isolation Cooling Discharge Check

Valve, Work Order 2276083
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 refueling outage U2RIO20 plant shut down on March 21, 2021.

Licensed Operator Requalification Training/Examinations (IP Section 03.02) (1 Sample)

(1) The inspectors observed and evaluated a licensed operator requalification simulator exam on January 14, 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 remain capable of performing their intended function:

(1) Unit 1, fire protection pipe hanger corrosion on the 656-foot level sump room due to ground water intrusion on March 8, 2021

Quality Control (IP Section 03.02) (1 Sample)

The inspectors evaluated the effectiveness of maintenance and quality control activities to ensure the following structures, systems, and components remain capable of performing its intended function:

(1) Unit Common, battery room exhaust fans 0V116A and 0V116B, after failure of 0V116A and subsequent identification of non-quality components being used on December 29, 2020

71111.13 - Maintenance Risk Assessments and Emergent Work Control Risk Assessment and Management (IP Section 03.01) (10 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, yellow risk during automatic depressurization level testing and containment instrument gas bottle bank work on January 5, 2021
2. (2) Unit 1, yellow risk during bus 0A106 work commencing on February 3, 2021
3. (3) Unit Common, elevated risk during 'E' emergency diesel generator swap-in on

   February 12, 2021

4. (4) Unit 2, elevated risk during overlapping work and system outages on February 18,

   2021

5. (5) Units 1 and 2, yellow risk due to inoperability of 'A' emergency diesel generator while

   'E' emergency diesel generator unavailable on February 23, 2021

6. (6) Unit 1, moisture separator drain tank dump valve air line fretting failure temporary

   repair on March 3, 2021

7. (7) Unit 2, residual heat removal logic functional test elevated risk activity on March 16,

   2021

8. (8) Unit 2, yellow risk for de-inerting the drywell 24 hours prior to planned shutdown to

   refueling outage U2RIO20 on March 23, 2021

9. (9) Unit 2, yellow shutdown risk during design basis testing (loss of coolant and loss of

   offsite power) on March 24 to 25, 2021

10. (10) Unit 1, yellow risk window for the division 1 emergency service water pipe

    replacement on March 25, 2021

71111.15 - Operability Determinations and Functionality Assessments
Operability Determination or Functionality Assessment (IP Section 03.01) (5 Samples)

The inspectors evaluated the licensee's justifications and actions associated with the following operability determinations and functionality assessments:

1. (1) Unit Common, Blue Max station portable diesel generator engine block heater failed to maintain proper temperature on December 20, 2020
2. (2) Unit 2, reactor protection system test box did not respond as expected during turbine control valve testing on December 21, 2020
3. (3) Unit 1, reactor core isolation cooling steam supply valve leaking steam as referenced in CR-2021-01838 on February 3, 2021
4. (4) Unit 2, Turbine Building closed cooling water to emergency service water/service water cross over pipe pinhole leak on February 23, 2021
5. (5) Unit 1, high-pressure coolant injection turbine exhaust 1B vacuum breaker valve (HV-155-F079) failed stroke time on March 9, 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 modifications:

(1) Unit Common, temporary modification for battery room exhaust fans 0V116A and 0V116B using non-quality belts in lieu of quality belts

71111.19 - Post-Maintenance Testing Post-Maintenance Test (IP Section 03.01) (7 Samples)

The inspectors evaluated the following post-maintenance test activities to verify system operability and functionality:

1. (1) Unit Common, battery room exhaust fan 0V116A belt replacement on January 7, 2021
2. (2) Unit Common, 'B' control structure chiller repair on January 12, 2021
3. (3) Unit 2, core spray pump 2B system outage window flow verification test

(SO-251-B02) on January 13, 2021

4. (4) Unit 1, reactor core isolation cooling valve F045 repair on February 11, 2021
5. (5) Unit Common, 'E' emergency diesel generator mid-cycle overhaul commencing

   February 12, 2021

6. (6) Unit 0, 'A' emergency diesel generator forced outage due to auto shutdown when

   securing after performance of monthly run (SO-024-001A) on February 23, 2021

7. (7) Unit 1, 1A moisture separator drain tank emergency dump valve (LV-10231A) and 3C

   feedwater heater emergency dump valve (HV-10444C) air line temporary pipe patch installation on March 3, 2021

71111.20 - Refueling and Other Outage Activities Refueling/Other Outage (IP Section 03.01) (1 Partial)

(1) (Partial)
The inspectors evaluated refueling outage U2RIO20 activities from March 21, 2021, to March 31, 2021. The inspectors completed IP Section 03.01, Sections A and B, and completed some portions of Section 03.01, Section C.

71111.22 - Surveillance Testing
The inspectors evaluated the following surveillance tests: Surveillance Tests (other) (IP Section 03.01) (6 Samples)

1. (1) Unit Common, 'A' emergency diesel generator 24-hour run on February 22, 2021
2. (2) Unit Common, 'A' loop emergency service water quarterly flow surveillance on February 25, 2021
3. (3) Unit Common, 'E' emergency diesel generator monthly surveillance on March 10, 2021
4. (4) Unit 1, turbine valve cycling surveillance on March 11, 2021
5. (5) Unit 2, 2-year residual heat removal logic functional test on March 16, 2021
6. (6) Unit 2, 2-year high-pressure coolant injection logic functional test, SQ-252-102, on

   March 19, 2021

Inservice Testing (IP Section 03.01) (3 Samples)

(1) Unit 2, reactor core isolation cooling quarterly flow verification on January 28, 2021
(2) Unit 2, 'B' loop residual heat removal comprehensive flow surveillance on

February 10, 2021
(3) Unit 1, high-pressure coolant injection flow surveillance (SO-152-002, Revision 74) on

March 10, 2021 71114.06 - Drill Evaluation

Select Emergency Preparedness Drills and/or Training for Observation (IP Section 03.01) (1 Sample)

(1) The inspectors evaluated focus area drills for the technical support center/operations support center on March 2, 2021, and the emergency operations facility on March 9, 2021.

OTHER ACTIVITIES – BASELINE

71151 - Performance Indicator Verification
The inspectors verified licensee performance indicators submittals listed below: IE01: Unplanned Scrams per 7000 Critical Hours (IP Section 03.01) (2 Samples)

1. (1) Unit 1 (January 1, 2020, to December 31, 2020)
2. (2) Unit 2 (January 1, 2020, to December 31, 2020)

IE03: Unplanned Power Changes per 7000 Critical Hours (IP Section 03.02) (2 Samples)

1. (1) Unit 1 (January 1, 2020, to December 31, 2020)
2. (2) Unit 2 (January 1, 2020, to December 31, 2020)

IE04: Unplanned Scrams with Complications (IP Section 03.03) (2 Samples)

(1) Unit 1 (January 1, 2020, to December 31, 2020) (2) Unit 2 (January 1, 2020, to December 31, 2020)

71153 - Followup of Events and Notices of Enforcement Discretion Event Report (IP Section 03.02) (2 Samples)

The inspectors evaluated the following licensee event reports (LERs):

(1) LER 05000387/2020-001-00 and LER 05000387/2020-001-01, Automatic Reactor Scram Due to Main Turbine Trip Caused by an Electrical Ground Path in the B Main Transformer (ADAMS Accession No. ML20183A146 and ML20310A258): The inspectors determined that it was not reasonable to foresee or correct the cause discussed in the LER; therefore, no performance deficiency was identified. The inspectors did not identify a violation of NRC requirements.

LER 05000387/2020-003-00, Condition Prohibited by Technical Specifications Due to Inoperable Turbine Stop Valve and Turbine Control Valve Instrumentation (ADAMS Accession No. ML20356A217): The inspectors evaluated the LER and determined that there was a violation of Technical Specification 3.0.4 on May 9, 2020, and the violation was documented in Susquehanna Steam Electric Station Integrated Inspection Report 05000387 and 05000388/2020002 (ADAMS Accession No. ML20224A179) in the Inspection Results section, Non-Cited Violation 05000387/2020002-01. No further performance deficiencies or violations of NRC regulations were determined.

INSPECTION RESULTS

<table>
<tr><td colspan="4">Criterion III, Design Control, for Failure to Install Quality Components in Battery Room Exhaust Fans</td></tr>
<tr><td>Cornerstone</td><td>Significance</td><td>Cross-Cutting Aspect</td><td>Report Section</td></tr>
<tr><td>Mitigating Systems</td><td>Green
NCV 05000387,05000388/[illegible]21001-01
Open/Closed</td><td>[H.5] - Work Management</td><td>71111.12</td></tr>
<tr><td colspan="4">A Green, non-cited violation of Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Appendix B, Criterion III, Design Control, was identified by inspectors when inspection activities revealed that a non-dedicated, non-quality controlled belt was installed on the 'B' battery room exhaust fan.</td></tr>
<tr><td colspan="4"><u>Description</u>: The battery room exhaust system functions to maintain the battery room design temperature, design pressure, and hydrogen concentration within limits. The battery room exhaust system consists of two independent, redundant subsystems consisting of fans, ductwork, dampers, and instrumentation and controls. The two safety-related fans, 0V116A (A fan) and 0V116B (B fan), each can provide the required air flow and use internal quality controlled class drive belts which connect the motor to the fan. By original design, the battery room exhaust fans relied upon using AX53, the quality controlled belt designated for this application.

In 2015, as documented in CR-2015-24859, the licensee discovered the use of non-quality controlled B52 belts in the battery room exhaust fans. At the time, the station completed a screening pursuant to 10 CFR Part 50.59, "Changes, Tests, and Experiments," due to the use of the non-quality controlled component. However, the station did not complete an engineering evaluation or engineering change document under the basis that these belts would not be used for this purpose in the future and assigned corrective actions to replace the non-quality controlled B52 belt with the appropriate quality controlled AX53 belt. In 2016, the corrective action to replace the B52 belts with the AX53 belts was completed. As part of the belt replacement, re-tensioning is required after 24 hours of run time. As documented in CR-2017-00265, the station failed to re-tension the belt in the 'A' fan; and as a result, belt replacement was required. As described in AR-2017-00266, the station reverted to the non-quality controlled B52 belts and performed a commercial dedication of the non-quality controlled belt.

In December 2020, the 'B' fan exhibited equipment issues; and a work order (RTPM 2306756) was required for the replacement of the installed AX53 belt. During the maintenance of the 'B' fan, station planning and maintenance staff were uncertain if the non-quality controlled B52 belt was allowed to be installed. After performing a document review, they identified in 2017 that a non-quality controlled B52 belt was installed in the 'A' fan, but did not investigate further as to why, or recognize the presence of the 2016 corrective action to preclude usage of the non-quality controlled B52 belt in the battery room exhaust fans. The licensee reverted to using the non-quality controlled B52 belt.

Shortly thereafter, as documented in CR-2020-17291, the station identified that the 'A' fan had thrown one of its installed belts. During the condition report and work order review, the NRC inspectors could not determine if the station had dedicated the non-quality controlled B52 belt installed in the 'B' fan during the December 2020 maintenance. As documented in CR-2021-00407, it was revealed that the non-quality controlled B52 belts were incorrectly</td></tr>
</table>

installed in both fans contrary to the 2015 decision to only use the quality controlled AX53 belts. It was revealed through subsequent discussions with the licensee that the non-quality B52 belt installed in the 0V116B fan was not dedicated.

Corrective Actions: The station assigned corrective actions to return the battery exhaust fans to original design by reinstalling AX53 belts, is developing an engineering evaluation to identify and ensure the proper belts are installed in future application, and is evaluating the preventative maintenance strategy.

Corrective Action References: CR-2021-00407, AR-2020-17299

Performance Assessment:

Performance Deficiency: The failure to ensure quality controlled components were installed in the battery room exhaust fans was a performance deficiency because it was within the licensee's ability to foresee and correct and should have been prevented. Specifically, the licensee's procedure NEPM-QA-0300, Revision 4, "Dedication of Commercial Grade Items and Services," specifies requirements for commercial grade dedication of components used in safety-related applications. The substitution of the non-quality controlled B52 belt should have resulted in the completion of a commercial grade dedication and verification that the component could meet all specified design functions; however, the inspectors identified that the licensee failed to complete the evaluation and dedication processes to ensure the function of the battery room exhaust fans could be maintained.

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 and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. This determination was informed by IMC 0612, Appendix E, Example 5.c, because the non-conforming component was installed and the system was returned to service and the extent of condition revealed additional non-conformances. Specifically, the degraded condition of the 'A' fan resulted in discovery that both trains of the system had been operating with components outside the original design scope, and the maintenance shortly prior to this had resulted in installation of a non-quality controlled component that was not commercially dedicated.

Significance: The inspectors assessed the significance of the finding using Appendix A, "The Significance Determination Process for Findings At-Power." This finding was determined to be Green because it is a deficiency affecting the design or qualification of a mitigating structure, system, and component and the system maintained its operability.

Cross-Cutting Aspect: H.5 - Work Management: The organization implements a process of planning, controlling, and executing work activities such that nuclear safety is the overriding priority. The failure of maintenance to coordinate and communicate belt replacement with the engineering work group in 2020 allowed a non-dedicated, non-quality belt to be installed inappropriately.

Enforcement:

Violation: 10 CFR Part 50, Appendix B, Criterion III, Design Control, requires, in part, that design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design and be approved by the organization that performed the original design unless the applicant designates another responsible

organization. Contrary to the above, the licensee did not implement approved design control measures commensurate with those applied to the original design. Specifically, in December 2020, when the licensee performed maintenance on the 'B' fan and replaced the battery room exhaust fan belt, they did not use the approved quality controlled belt and did not perform commercial grade dedication. Enforcement Action: This violation is being treated as a non-cited violation, 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 8, 2021, the inspectors presented the inservice inspection results to Mr. Kevin Cimorelli, Site Vice President, and other members of the licensee staff.
- On April 29, 2021, the inspectors presented the integrated inspection results to Mr. Derek Jones, Plant Manager, and other members of the licensee staff.

DOCUMENTS REVIEWED

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
71111.04	Drawings	E105952	Unit 2 P&ID Core Spray	Revision 26
71111.05	Corrective Action Documents	AR-2021-01559 AR-2021-01565 CR-2021-01557 CR-2021-01562 CR-2021-01563		
	Corrective Action Documents Resulting from Inspection	AR-2021-01563 CR-2021-01566 CR-2021-01567 CR-2021-01568		
	Fire Plans	FP-013-164	Unit 1 Upper Relay Room, FZ 0-27E, Control Structure, El. 754	Revision 7
		FP-113-123	Load Center Room (I-507, I-510)	Revision 4
		FP-213-100	Fire Zone 2-4F, Unit 2 Drywell	Revision 3
		FP-213-236	Fire Zone 2-1A, Core Spray Pump Room 'B'	Revision 6
		FP-213-237	Fire Zone 2-1B, Core Spray Pump Room 'A'	Revision 5
		FP-213-257	Pipe Penetration Room (II-506)	Revision 5
71111.11Q	Corrective Action Documents Resulting from Inspection	CR-2021-00800		
71111.12	Corrective Action Documents	AR-2017-00266 CR-2015-20731 CR-2015-24859 CR-2017-00265 CR-2018-06662 CR-2020-00684 CR-2020-17291 CR-2021-00407 CR-2021-00425 CR-2021-03222		
	Work Orders	PCWO 1930681 PCWO 2052696		

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		PCWO 2141600 PCWO 2145344 PCWO 2185421 RTPM 2306756		
71111.13	Corrective Action Documents	2021-02911		
71111.15	Corrective Action Documents	CR-2020-17085 CR-2020-17117 CR-2021-02613 CR-2021-03362		
	Corrective Action Documents Resulting from Inspection	CR-2021-01838		
	Operability Evaluations	ACT-01-CR-2021-02613		
71111.19	Corrective Action Documents	CR-2021-01943 CR-2021-02275 CR-2021-02681		
	Work Orders	ERPM 2265930 PCWO 2408814 RLWO 2408825 RLWO 2410979 RTSV 2401970		
71111.22	Corrective Action Documents	2021-03737		
	Corrective Action Documents Resulting from Inspection	CR-2021-03952		
	Procedures	RTSV 2237263	SO-249-B06, RHR Comprehensive Flow Verification Loop B	Revision 17
		SO-024-A01	Diesel Generator A Integrated Surveillance Test	Revision 1
		SO-249-007	RHR Logic System Functional Test (DIV 1) Online (Partial)	Revision 6
	Work Orders	RTSV 2250012		

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		RTSV 2365853 RTSV 2387144 RTSV 2387872		

July 29, 2021 – Letter from Mel Gray, Chief Engineering Branch 1 Division of Operating Reactor Safety to Brad Berryman President and Chief Nuclear Officer Susquehanna Nuclear, LLC with subject of SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 – INFORMATION REQUEST TO SUPPORT TRIENNIAL BASELINE DESIGN-BASIS CAPABILITY OF POWER-OPERATED VALVES INSPECTION; INSPECTION REPORT 05000387/2021010 AND 05000388/2021010

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 Susquehanna Steam Electric Station, Units 1 and 2. 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. ML 20220A667).

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 July 30, 2020, with Mr. Shane Jurek, Senior Licensing 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 August 16
- Onsite weeks: Weeks of November 1 and November 15

  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. Frank Arner, a Region I Senior Risk Analyst, will support David Kern during the information-gathering visit to review probabilistic risk assessment data and identify the final POV samples to be examined during the inspection.

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 August 16, 2021. 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. 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."

DOCUMENT REQUEST FOR DESIGN BASES ASSURANCE INSPECTION

Inspection Report: Onsite Inspection Dates:

Inspection Procedure: Lead Inspector:

05000387/2021010 and 05000388/2021010

November 1 through November 5, 2021; and November 15 through November 19, 2021

Inspection Procedure 71111.21N.02, Design-Basis Capability of Power-Operated Valves Under 10 CFR 50.55a Requirements

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 August 16, 2021, 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, 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 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.***

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 November 1, 2016 (include document No., title/short description, date).
7. List of corrective action documents related to each of the 30 POVs listed below since November 1, 2016 (include document No., title/short description, date).
8. List of significant modifications, repairs, or replacement of safety-related POVs completed since November 1, 2016, 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 November 1, 2016).
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. For each of the following MOVs, provide the information listed in the table below.

- ● HV01222B
- ● HV112F073A
- ● HV149F007
- ● HV151F004A
- ● HV151F016B
- ● HV151F028A

Residual Heat Removal Service Water - UHS Spray Bypass Isolation Valve
Residual Heat Removal Service Water - RHR/RHRSW Loop A Crosstie Valve

RCIC Turbine Pump - RCIS Turbine Steam Supply Inboard Isolation Valve
Residual Heat Removal - RHR Pump A Suppression Pool Suction Valve

Residual Heat Removal - RHR Loop B Drywell Spray Outboard Isolation Valve
Residual Heat Removal - RHR Loop A Suppression Pool Spray Test Shutoff Valve

DOCUMENT REQUEST FOR DESIGN BASES ASSURANCE INSPECTION

- ● HV155F003
- ● HV155F003
- ● HV155F079
- ● HV21210A
- ● HV249F013
- ● HV250F045
- ● HV250F046
- ● HV251F015B
- ● HV251F021A
- ● HV251F024B
- ● HV251F048B
- ● HV255F001
- ● HV255F006
- ● HV255F042

High Pressure Core Spray - HPCI Turb Steam Supply Outboard Isolation Valve
High Pressure Core Spray - HPCI Turb Steam Supply Outboard Isolation Valve

High Pressure Core Spray - HPCI Turb Exhaust Inboard Vac Breaker Valve
Residual Heat Removal Service Water - RHR HX A SW Supply Isolation Valve

RCIC Turbine Pump - RCIC Injection Valve
RCIC Turbine Pump - RCIC Turbine Steam Supply Valve
RCIC Turbine Pump - RCIC Lube Oil Cooler Water Supply Valve Residual Heat Removal - RHR Loop B Injection Outboard Isolation Valve
Residual Heat Removal - RHR Loop A Drywell Spray Inboard Isolation Valve
Residual Heat Removal - RHR Loop B Supp Pool Cooling/Test Control Valve
Residual Heat Removal - RHR HX B Shell Side Bypass Valve Residual Heat Removal - HPCI Turb Steam Supply Valve
High Pressure Core Spray - HPCI Injection Valve
High Pressure Core Spray - HPCI Pump Suction Suppression Pool Supply 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)

DOCUMENT REQUEST FOR DESIGN BASES ASSURANCE INSPECTION

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*

16. For each of the following AOVs/SOVs/HOVs, provide the information listed in the table below.

- HV141F022A Main Steam - "A" Main Steam Line Inboard Isolation Valve
- HV141F028D Main Steam - "D" Main Steam Line Outboard Isolation Valve
- HV16108A1 Liquid Rad Waste - Drywell Floor Drain Sump Pump Discharge Isolation Valve
- HV16116A2 Liquid Rad Waste - Drywell Equipment Drain Tank Discharge Isolation Valve
- HV18792B2 Reactor Building Chill Water - RRP A Cooling Water Inboard Isolation Reactor Building Chilled Water Return
- HV25703 Liquid Rad Waste - Suppression Chamber Purge Exhaust Upstream Isolation Valve
- HV25723 Containment Atmosphere Control - Drywell Air Purge Isolation Valve
- SV12654A Containment Instrument Gas - I-G to Main Steam PSV1F013 GJM
- TV01124E Emergency Service Water - DG E ESW Loop B Return Temperature Control Valve
- XV247F010A Control Rod Drive Hydraulic - CRD SDV Vent Valve
- XV247F010B Control Rod Drive Hydraulic - CRD SDV Vent 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)

DOCUMENT REQUEST FOR DESIGN BASES ASSURANCE INSPECTION

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*

August 4, 2021 – Letter from Jonathan E. Greives, Chief Projects Branch 4 Division of Reactor Projects to Brad Berryman President and Chief Nuclear Officer Susquehanna Nuclear, LLC with subject of SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 – INTEGRATED INSPECTION REPORT 05000387/2021002 AND 05000388/2021002

On June 30, 2021, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Susquehanna Steam Electric Station, Units 1 and 2. On July 29, 2021, the NRC inspectors discussed the results of this inspection with Mr. Kevin Cimorelli, 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: 05000387 and 05000388
License numbers: NPF-14 and NPF-22
Report numbers: 05000387/2021002 and 05000388/2021002

Enterprise Identifier: I-2021-002-0025
Licensee: Susquehanna Nuclear, LLC
Facility: Susquehanna Steam Electric Station, Units 1 and 2

Location: Berwick, PA
Inspection dates: April 1, 2021, to June 30, 2021

Inspectors: C. Highley, Senior Resident Inspector
M. Rossi, Resident Inspector
H. Anagnostopoulos, Senior Health Physicist M. Henrion, Health Physicist
D. Kern, Senior Reactor Inspector
A. Turilin, Reactor Inspector

Approved by: Jonathan E. Greives, Chief 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 Susquehanna Steam Electric Station, Units 1 and 2, 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 1 began the inspection period at rated thermal power and remained at or near rated thermal power until June 16, 2021, when the unit was down powered to 78 percent for turbine valve testing. The unit was returned to 100 percent power on June 16, 2021. On June 25, 2021, the unit was down powered to 66 percent for a rod sequence exchange. The unit was returned to 100 percent on June 30, 2021.

Unit 2 was shut down at the beginning of the inspection period. The unit was started up on April 22, 2021, and achieved 100 percent power on April 30, 2021. On May 1, 2021, the unit was down powered to 60 percent for a rod pattern adjustment. The unit was returned to
100 percent on May 3, 2021. On May 4, 2021, the unit was down powered to 90 percent for a rod pattern adjustment. The unit was returned to 100 percent on May 4, 2021. On May 5, 2021, the unit was down powered to 90 percent for a rod pattern adjustment. The unit was returned to 100 percent on May 5, 2021. On May 8, 2021, the unit was down powered to

88 percent for a rod pattern adjustment. The unit was returned to 100 percent on May 8, 2021. On May 11, 2021, the unit was down powered to 90 percent for a rod pattern adjustment. The unit was returned to 100 percent on May 12, 2021. On May 15, 2021, the unit was down powered to 62 percent for a rod pattern adjustment. The unit was returned to 100 percent on May 17, 2021, and remained at or near rated thermal power 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 (IP Section 03.01) (1 Sample)

(1) The inspectors evaluated readiness for seasonal extreme weather conditions prior to the onset of summer grid readiness for the following systems:

- 230kV, 500kV, and T-10 switchyard walkdown on May 6, 2021
- 'B1' spray pond header inspection on May 12, 2021

71111.04 - Equipment Alignment
Partial Walkdown (IP Section 03.01) (3 Samples)

The inspectors evaluated system configurations during partial walkdowns of the following systems/trains:

(1) Unit 2, 'A' core spray prior to divisional swap on April 2, 2021

(2) Units 1 and 2, spent fuel pool cooling system heat exchangers prior to removing shutdown cooling from service for residual heat removal system maintenance on April 5 and 6, 2021.

(3) Unit 1, standby liquid control system on June 17, 2021 Complete Walkdown (IP Section 03.02) (1 Sample)

(1) The inspectors evaluated system configurations during a complete walkdown of the Unit 2 division 2 residual heat removal system on May 14, 2021.

71111.05 - Fire Protection
Fire Area Walkdown and Inspection (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) Unit 2, main steam tunnel (FZ 2-4G) on April 5, 2021
2. (2) Units 1 and 2, fuel pool cooling heat exchanger rooms (FZ 1-5D and 2-5D) on

   April 5 and 6, 2021

3. (3) Unit 1, sump room, 645-foot elevation (FZ 1-1G), on April 15, 2021
4. (4) Unit 2, drywell (FZ 2-4F) on April 19, 2021
5. (5) Unit 2, condenser bay (FZ 2-31D, 2-32D, and 2-33C) and main steam pipeway

   (FZ 2-4G and 2-34B) on April 20, 2019

6. (6) Unit 2, equipment access area (FZ 2-3C) on May 3 and May 12, 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 1, control rod drive pumps on June 28, 2021 Cable Degradation (IP Section 03.02) (1 Sample)

The inspectors evaluated cable submergence protection in:

(1) Manway hole MH014, MH015, and MH060 vault inspection during monthly water removal on June 10, 2021

71111.07A - Heat Sink Performance
Annual Review (IP Section 03.01) (1 Sample)

The inspectors evaluated readiness and performance of:

(1) Unit 2, 'A' residual heat removal heat exchanger eddy current inspection on April 27, 2021

71111.07T - Heat Sink Performance
Heat Exchanger (Service Water Cooled) (IP Section 03.02) (1 Sample)

(1) The inspectors evaluated heat exchanger/sink performance on the following:

- Unit 1, 'A' turbine building, closed cooling water heat exchanger, cooled by service water
- Unit 2, 'B' reactor building, closed cooling water heat exchanger, cooled by service water
- Common, ultimate heat sink, sections 03.04a, 03.04c, and 03.04d 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 Unit 2 startup following a refueling outage U2RIO20 on April 22, 2021.

Licensed Operator Requalification Training/Examinations (IP Section 03.02) (1 Sample)

(1) The inspectors observed and evaluated licensed operator performance in response to an inadvertent zone 3 isolation, lowering vacuum, stuck open safety relief valve, and hydraulic anticipated transient without a scram in the plant simulator on May 13, 2021.

71111.12 - Maintenance Effectiveness
Maintenance Effectiveness (IP Section 03.01) (4 Samples)

The inspectors evaluated the effectiveness of maintenance to ensure the following structures, systems, and components remain capable of performing their intended function:

(1) Unit 2, reactor water clean up demineralizer back flush pipe flange corrosion control on June 14, 2021

(2) Unit Common, 'E' emergency diesel generator availability following an unplanned extension of maintenance in March on June 16, 2021

(3) Unit 1, instrument air system based on repetitive compressor failures on June 24, 2021

(4) Unit Common, periodic evaluation of maintenance rule program on June 30, 2021
71111.13 - Maintenance Risk Assessments and Emergent Work Control
Risk Assessment and Management (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 2, yellow shutdown risk during the common shutdown cooling work window (ZWO 2404109) on April 5 to 9, 2021

(2) Unit 2, yellow shutdown risk during cavity letdown on April 13, 2021
(3) Unit 2, yellow risk during change to MODE 2 with drywell de-inerted on April 21, 2021
(4) Unit 2, yellow risk during tie bus 0A107 10-year maintenance and inspection on

June 2, 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. (1) Unit 2, prompt functionality assessment for reactor pressure vessel drain line wall thickness issue on April 20, 2021
2. (2) Unit 1, 1B turbine building chiller trip on May 14, 2021
3. (3) Unit 2, reactor core isolation cooling F088 valve appears to be mid-position,

   CR 2021-04202, on May 17, 2021

4. (4) Unit Common, 'A' emergency service water pump in alert range, CR-2021-08565, on

   June 23, 2021

71111.18 - Plant Modifications

Temporary Modifications and/or Permanent Modifications (IP Section 03.01 and/or 03.02) (3 Samples)

The inspectors evaluated the following temporary or permanent modifications:

(1) Unit 2, temporary modification for reactor pressure vessel drain line flow modification on April 20, 2021

(2) Unit 2, permanent modification for emergency service water buried piping replacement on May 15, 2021

(3) Unit Common, permanent modification for magnetic trip setting revision of breakers for the 'E' emergency diesel generator starting air compressors on May 18, 2021

71111.19 - Post-Maintenance Testing Post-Maintenance Test (IP Section 03.01) (11 Samples)

The inspectors evaluated the following post-maintenance test activities to verify system operability and functionality:

1. (1) Unit 2, 'A' loop residual heat removal outage maintenance and repair work on April 2, 2021
2. (2) Unit 2, main steam safety relief valves D, E, F, G, J, K, L, M, N, and R remote actuation following maintenance, OT-283-001, Revision 5, on April 8, 2021
3. (3) Unit 2, 'A' residual heat removal service water heat exchanger disassembly and maintenance on April 13, 2021
4. (4) Unit 2, outboard main steam isolation valve 28A repair on April 14, 2021
5. (5) Unit 2, replaced cells 120 and 108 in 2D660-250VDC battery bank, Work Orders

   2299854 and 2136204, on April 15, 2021

6. (6) Unit 2, 2D650 A 250 volts direct current battery replacement, Work Order

   2072469, on April 20, 2021

7. (7) Unit 2, reactor core isolation cooling F008 valve repair on April 21, 2021
8. (8) Unit 2, reactor core isolation cooling 10-year overhaul (license renewal commitment)

   on April 22, 2021

9. (9) Unit Common, 'A' emergency diesel generator control air pressure regulator failure

   and repairs, SO-024-001A, Revision 29, on May 3, 2021

10. (10) Unit 1, scram discharge volume valve and piping repairs on May 11, 2021
11. (11) Unit Common, 'A' emergency service water pump lift adjustment on June 3, 2021

71111.20
Refueling/Other Outage (IP Section 03.01) (1 Sample)

- Refueling and Other Outage Activities

(1) Unit 2, the inspectors evaluated the refueling outage activities from March 21 to April 21, 2021

71111.22 - Surveillance Testing
The inspectors evaluated the following surveillance tests: Surveillance Tests (other) (IP Section 03.01) (5 Samples)

1. (1) Unit 2, loss of coolant accident coincident with loss of offsite power testing on April 10, 2021
2. (2) Unit 2, 2-year manual actuation of automatic depressurization system valves, SO-283-002, Revision 18, on April 10, 2021
3. (3) Unit 2, high-pressure coolant injection and reactor core isolation cooling 150 psi test during startup on April 22, 2021
4. (4) Unit 2, rod worth minimizer surveillance and operability test during startup on April 22, 2021
5. (5) Unit 0, 'D' emergency diesel generator monthly operability run May 27, 2021

Reactor Coolant System (RCS) Leakage Detection Testing (IP Section 03.01) (2 Samples)

1. (1) Unit 1, radio Iodine specific activity dose equivalent I-131 on June 8, 2021
2. (2) Unit 2, RCS leakage shiftly calculation surveillance on June 15, 2021

Containment Isolation Valve Testing (IP Section 03.01) (2 Samples)

1. (1) Unit 2, reactor core isolation cooling local leak-rate testing on April 2, 2021
2. (2) Unit 2, main steam isolation valve as-found local leak-rate testing on April 22, 2021

71114.06 - Drill Evaluation
Drill/Training Evolution Observation (IP Section 03.02) (1 Sample)

The inspectors evaluated:

(1) Emergency operations facility and joint information center focus area drill involving loss of offsite power on May 25, 2021

RADIATION SAFETY

71124.01 - Radiological Hazard Assessment and Exposure Controls Radiological Hazard Assessment (IP Section 03.01) (1 Sample)

(1) The inspectors evaluated how the licensee identifies the magnitude and extent of radiation levels and the concentrations and quantities of radioactive materials and how the licensee assesses radiological hazards.

Instructions to Workers (IP Section 03.02) (1 Sample)

The inspectors evaluated instructions to workers including radiation work permits used to access high radiation areas and reviewed the following:

(1) Radiation Work Packages (RWPs) • RWP 202124000

• RWP 20212360 • RWP 20212126

Electronic Alarming Dosimeter Alarms • Alarm on 2/8/2021
• Alarm on 2/10/2021

Labeling of Containers

- Unit 2, Reactor Building, 719-foot elevation, drum of hoses for work on hydraulic control units
- Unit 2, bagged valve parts in the outboard main steam isolation valve room (wingwall area)
- Unit 2, Reactor Building, 719-foot elevation, gang box of tools of the drywell

  Contamination and Radioactive Material Control (IP Section 03.03) (2 Samples)

  The inspectors evaluated licensee processes for monitoring and controlling contamination and radioactive material as follows:

1. (1) Unit 2, Reactor Building, 719-foot elevation, decontamination of a discrete radioactive particle area for the control rod drive mechanism removal pathway
2. (2) Unit 2, Reactor Building, 719-foot elevation, routine monitoring of personnel

Radiological Hazards Control and Work Coverage (IP Section 03.04) (3 Samples)

The inspectors evaluated in-plant radiological conditions during facility walkdowns and observation of radiological work activities. The inspectors also reviewed the following RWPs for areas with airborne radioactivity:

1. (1) RWP 2012112
2. (2) RWP 20212213
3. (3) RWP 2021353

High Radiation Area and Very High Radiation Area Controls (IP Section 03.05) (2 Samples)

The inspectors evaluated licensee controls of the following high radiation areas and very high radiation areas:

1. (1) Unit 2, Reactor Building, reactor backwash receiving tank room
2. (2) Unit 2, Reactor Building, outboard main steam isolation valve room (wingwall)

Radiation Worker Performance and Radiation Protection Technician Proficiency (IP Section 03.06) (1 Sample)

(1) The inspectors evaluated radiation worker and radiation protection technician performance as it pertains to radiation protection requirements.

71124.02 – Occupational As Low As Reasonably Achievable (ALARA) Planning and Controls Radiological Work Planning (IP Section 03.01) (3 Samples)

The inspectors evaluated the licensee’s radiological work planning for the following activities:

1. (1) RWP 20212320, Revision 0
2. (2) RWP 20212001, Revision 0
3. (3) RWP 20212002, Revision 0

Verification of Dose Estimates and Exposure Tracking Systems (IP Section 03.02) (3 Samples)

The inspectors evaluated dose estimates and exposure tracking.

(1) The inspectors reviewed the following ALARA planning documents: • ALARA Pre-Job Review for RWP 20212320, Revision 0

(2) The inspectors reviewed the following radiological outcome evaluations: • ALARA Post-Job Review for RWP 20212320, Revision 0

(3) The inspectors reviewed the following radiological outcome evaluations: • ALARA Post-Job Review for RWP 20212017, Revision 0

Implementation of ALARA and Radiological Work Controls (IP Section 03.03) (2 Samples)

The inspectors reviewed ALARA practices and radiological work controls and reviewed the following activities:

1. (1) Unit 2, decontamination of the subpile room in the drywell
2. (2) Work on the 28A outboard main steam isolation valve

Radiation Worker Performance (IP Section 03.04) (2 Samples)

The inspectors evaluated radiation worker and radiation protection technician performance during the following activities:

1. (1) Unit 2, decontamination of the subpile room in the drywell
2. (2) Unit 2, Reactor Building, decontamination of a discrete radioactive particle area at the

   control rod drive mechanism removal pathway

71124.04 - Occupational Dose Assessment
Special Dosimetric Situations (IP Section 03.04) (2 Samples)

The inspectors evaluated the following special dosimetric situation:

1. (1) The licensee’s implementation of requirements to manage radiation protection of nine declared pregnant workers.
2. (2) The licensee's method of assigning dose exposure when large dose gradients exist. 10

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.

Groundwater Protection Initiative 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 licensee performance indicators submittals listed below: BI01: RCS Specific Activity (IP Section 02.10) (2 Samples)

1. (1) Unit 1 (January 1, 2020, through December 31, 2020)
2. (2) Unit 2 (January 1, 2020, through December 31, 2020)

BI02: RCS Leak Rate (IP Section 02.11) (2 Samples)

(1) Unit 1 (January 1, 2020, through December 31, 2020) (2) Unit 2 (January 1, 2020, through December 31, 2020)

71152 - Problem Identification and Resolution Semiannual Trend Review (IP Section 02.02) (1 Sample)

(1) The inspectors reviewed the licensee’s corrective action program for potential adverse trends in non-condition adverse to quality (NAQ) documents that might be indicative of a more significant safety issue.

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) Control structure chilled water system reliability

INSPECTION RESULTS

Observation: Corrective Actions to Address Control Structure Chilled Water System Reliability Challenges	71152
The inspectors selected control structure chilled water (CSCW) system performance/reliability as an annual inspection sample due to the system experiencing 12 unplanned chiller trips or train outages over the last 3 years. The safety-related CSCW system has two 100 percent capacity trains and is designed to remove heat from the control room, computer room, control structure ventilation, and the Unit 1 emergency switchgear room. If a control structure chiller trips, the opposite train chiller is designed to automatically start and assume the system heat removal function. Inspection scope included plant walkdowns of the CSCW system; review of corrective maintenance, preventive maintenance (PM), and design modification records for the last 5-year period; and review of issue documentation and prioritization, causal evaluation, and corrective actions to improve CSCW reliability. The inspectors determined the causes of the chiller trips during the last 3 years were typically unrelated. While reviewing recent CSCW chiller modifications, the inspectors identified two recent PM revisions which extended the replacement interval for Agastat 7000 series relays beyond manufacturer recommendations. Engineers verified the affected relays were installed in 2018, remained well within their vendor recommended service life, and therefore did not adversely impact CSCW chiller reliability. The issue was documented in CR-2021-08362 to review the relay replacement PM intervals and potential extent-of-condition. The inspectors concluded the licensee appropriately monitored CSCW system operation and documented associated equipment reliability issues in the corrective action program. Periodic testing verified the system was capable of performing its design functions. Additionally, engineers evaluated CSCW equipment issues and initiated corrective actions which addressed the cause of the condition adverse to quality (CAQ) in a timely manner, commensurate with its safety significance. Corrective actions completed and/or scheduled were appropriate to reduce likelihood of additional CSCW chiller trips.	

Observation: Semi Annual Trend Review of Non-Condition Adverse to Quality Documents Meeting Definition for a Condition Adverse to Quality or Condition Adverse to Regulatory Compliance	71152
The inspectors reviewed the licensee's corrective action program for potential adverse trends in NAQ documents that might be indicative of a more significant safety issue. The inspectors reviewed a sampling, 30 of 480, of NAQ documents that were generated between October 1, 2020, and March 31, 2021, that had a subject indicating a potential to be a CAQ or a condition adverse to regulatory compliance (CARC). Of the 30 reviewed, none seemed to meet the definition of a CAQ or a CARC, and the inspectors determined that Susquehanna was appropriately classifying conditions as NAQs.	

EXIT MEETINGS AND DEBRIEFS

The inspectors verified no proprietary information was retained or documented in this report.

- On May 20, 2021, the inspectors presented the triennial heat sink inspection results to Mr. Kevin Cimorelli, Site Vice President, and other members of the licensee staff.
- On July 29, 2021, the inspectors presented the integrated inspection results to Mr. Kevin Cimorelli, Site Vice President, and other members of the licensee staff.

DOCUMENTS REVIEWED

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
71111.01	Drawings	M-80-2 Sheet 2	ESSW Spray Pond Network schematic	Revision 2
71111.04	Drawings	E105951, SH3	Unit 2 Residual Heat Removal	Revision 29
		E106253, SH1	Unit 1 P&ID Standby Liquid Control	Revision 40
71111.05	Corrective Action Documents Resulting from Inspection		CR-2021-05302, CR-2021-05309, CR-2021-05311, CR-2021-05312, CR-2021-05314, CR-2021-05603, CR-2021-05610, CR-2021-05612, R-2021-05631, CR-2021-05633, CR-2021-05694, CR-2021-05726, CR-2021-06204, CR-2021-06212, CR-2021-06268, CR-2021-06288, CR-2021-06305, CR-2021-06402, CR-2021-06658, CR-2021-06721, CR-2021-06737, CR-2021-07468, CR-2021-08023	
	Fire Plans	FP-113-117	Sump Pump Room (I-15) Fire Zone 1-1G Elevation 646'-0"	Revision 4
		FP-113-119	Circulation Space (I-500) and Adjacent Rooms (I-511, 517, 514, 508, 513) Fire Zones 1-5A-N, S, W, 1-5H	Revision 6
		FP-213-100	Drywell (II-400, II-516, II-607) Fire Zone 2-4F El. 704' through 807'	Revision 3
		FP-213-247	Equipment Access Area (II-202, 204, 205) Fire Zones 2-3C-N, W, S	Revision 5
		FP-213-253	Main Steam Pipeway Area (II-411) Fire Zone 2-4G	Revision 6
		FP-213-254	Fuel Pool Heat Exchanger Room (II-514) Fire Zones 2-5A-N, S, W, 2-5H	Revision 8
71111.06	Engineering Evaluations	EC-FLOD-0001	Internal Flooding Evaluations for Moderate Energy Pipe Cracks and Sprinkler Actuations	Revision 10
		EC-FLOD-1004	SSES Flood Hazard Reevaluation Report	Revision 0
71111.07A	Work Orders		2275302	
71111.07T	Calculations	EC-054-0537	ESW System Heat Load and Flow Rate Requirements for Uprated Power Conditions	09/15/2016
		EC-STRU-2076	Structural Monitoring Program Inspections of Spray Pond	05/12/2021
	Corrective Action Documents		CR-2019-10103, CR-2021-01109	
	Miscellaneous	1E123A	2010 Tube Plug Map	03/17/2011
		2E201B	2010 Tube Plug Map	05/10/2011

71111.07T	NDE Reports	1E123A	1A Heat Exchanger Inspections Report	08/28/2017
		2E201B	2B Heat Exchanger Inspection Report	08/22/2019
	Procedures	OT-054-076	ESW Flow Balance	02/25/2021
	Work Orders		1933602, 2277478	
71111.12	Corrective Action Documents		CR-2021-02304, CR-2021-02791, CR-2021-09304, DI-2020-06035, DI-2021-02555, DI-2021-02972	
	Procedures	OP-261-002	Backwashing a Reactor Water Clean Up Filter Demineralizer in Auto	Revision 64
71111.13	Corrective Action Documents Resulting from Inspection		CR-2021-05245	
71111.15	Calculations	EC-054-0511	Single ESW Pump Operation Bounding Conditions	Revision 7
	Corrective Action Documents		CR-2021-04202, CR-2021-08103, CR-2021-08565	
	Operability Evaluations	ACT-01-CR-2021-04937		
	Procedures	EO-200-030	Station Blackout Procedure	Revision 36
71111.18	Corrective Action Documents		AR-2021-04348, CR-2020-13872, CR-2020-15794, CR-2021-02871, CR-2021-03351	
	Engineering Changes	EC-2052409	Replacement/Repair of the Unit 2A (Div 1) ESW Supply and Return Header	04/30/2021
		EC-2417619	Revise Magnetic Trip Setting of Breakers	Revision 1
71111.19	Corrective Action Documents		CR-2021-07410	
	Procedures	SO-155-002	Quarterly SDV Vent and Drain Valves Operability Check	Revision 26
	Work Orders		2426808-0, 2428405-0, PCWO 2277122,PCWO 2280814, PCWO 2285293, PCWO 2286497, PCWO 2426817-1, RLWO 2435462, RTSV 2[illegible]94380, RTSV 2404579	
71111.22	Corrective Action Documents Resulting from Inspection		CR-2021-08178, CR-2021-08180	

71111.22	Procedures	SC-176-102	Unit 1 Primary Coolant Specific Activity Dose Equivalent I-131	Revision 17
		SE-259-022	LLRT of Main Steam Line Isolation Valves	Revision 21
		SO-024-001D	Monthly Diesel Generator 'D' Operability Test	Revision 29
		SO-224-117	Unit 2 Division 1 Diesel Generator LOCA LOOP Test	Revision 5
	Work Orders		RACT 2102549, RTSV 2077950, RTSV 2391626	
71152	Corrective Action Documents		AR-2020-17319, AR-2021-02344, AR-2021-03068, CR-2017-03751, CR-2020-13756, CR-2020-14086, CR-2020-14693, CR-2020-15194, CR-2020-15507, CR-2020-15803, CR-2020-15851, CR-2020-16116, CR-2020-16360, CR-2020-17167, CR-2020-17171, CR-2021-00758, CR-2021-10407, CR-2021-01823, CR-2021-01825, CR-2021-02129, CR-2021-02137, CR-2021-02156, CR-2021-03314, CR-2021-03409, CR-2021-04177, CR-2021-04190, CR-2021-04925, DI-2021-00422, DI-2021-00426, DI-2021-01087, DI-2021-03782	
	Corrective Action Documents Resulting from Inspection		CR-2021-08255, CR-2021-08362	
	Drawings	E106291	Control Structure Chilled Water System "A" Common P&ID	Revision 43
	Engineering Changes	EC-1684214	Control Structure Chiller Time Delay Changes	Revision 1
		TDEC-2427915	Temporary Control Structure Chiller 0K112A Bearing High Temperature Trip Elimination	Revision 0
	Engineering Evaluations		System 30 Control Structure HVAC Maintenance Rule Basis Document	Revision 5
	Miscellaneous	IOM 168	Operating Instructions for Carrier Centrifugal Refrigeration Machines	09/01/1976
		TRM 3.7.9	Control Structure HVAC	Revision 3
		TS 3.7.4	Control Room Floor Cooling System	Revision 271
		UFSAR 9.2.12	Chilled Water Systems	Revision 70
	Work Orders		1739956	

September 1, 2021 – Letter from Jonathan E. Greives, Chief Projects Branch 4 Division of Operating Reactor Safety to Brad Berryman President and Chief Nuclear Officer Susquehanna Nuclear, LLC with subject of UPDATED INSPECTION PLAN FOR THE SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 (REPORT 05000387/2021005 AND 05000388/2021005)

The enclosed inspection plan lists the inspections scheduled through June 30, 2023, for Susquehanna Steam Electric Station, Units 1 and 2. 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.

In addition to baseline inspections, the NRC will conduct Inspection Procedure 71003, "Post Approval Site Inspection for License Renewal," for Units 1 and 2 in January 2022, for Unit 1 in April 2022, and for Unit 2 in March 2023.

As stated in NRC Investigation Report 1-2020-001 and NRC Inspection Report 05000388/2021012, the NRC will conduct Inspection Procedure 92702, "Follow-up on Traditional Enforcement Actions Including Violations, Deviations, Confirmatory Action Letters, Confirmatory Orders, and Alternative Dispute Resolution Confirmatory Orders." The NRC staff will work with your appropriate regulatory affairs staff to schedule this inspection.

Additionally, during this period the NRC has scheduled an additional inspection per a revised version of Temporary Instruction 2515/194, "Inspection of the Licensee's Implementation of Industry Initiative Associated with the Open Phase Condition Design Vulnerabilities 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.

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

Susquehanna

07/01/2021 - 06/30/2023

Unit	Start	End	Activity	CAC	Title	Staff Count
FORCE-ON-FORCE PLANNING AND EXERCISE WEEKS - SQ						6
1, 2	7/19/2021	7/23/2021	IP 71130.03	000735	Contingency Response - Force-On-Force Testing	
FY21 Susquehanna Initial Examination						6
1, 2	7/25/2021	8/6/2021	EXAD	000500	FB OR INITIAL LICENSE EXAM ADMINISTRATION (EXAD)	
Rad Safety - Shipping						1
1, 2	8/23/2021	8/27/2021	IP 71124.08	000732	Radioactive Solid Waste Processing & Radioactive Material Handling, Storage, & Transportation	
Protective Strategy, Drill, SUS						2
1, 2	9/7/2021	9/9/2021	IP 71130.05	000737	Protective Strategy Evaluation	
Evaluations of Changes, Tests and Experiments						3
1, 2	9/19/2021	9/25/2021	IP 71111.17T	000709	Evaluations of Changes, Tests, and Experiments	
Rad Safety - RETS & PIV						1
1, 2	10/11/2021	10/15/2021	IP 71124.06	000730	Radioactive Gaseous and Liquid Effluent Treatment	
1, 2	10/11/2021	10/15/2021	IP 71151	000746	Performance Indicator Verification	
Access Control, Equipment Testing and Maintenance, Training, SPR						3
1, 2	10/18/2021	10/22/2021	IP 71130.02	000734	Access Control	
1, 2	10/18/2021	10/22/2021	IP 71130.04	000736	Equipment Performance, Testing, and Maintenance	
1, 2	10/18/2021	10/22/2021	IP 71130.07	000739	Security Training	
1, 2	10/18/2021	10/22/2021	IP 71130.09	001656	Security Plan Changes	
1, 2	10/18/2021	10/22/2021	IP 71151	001338	Performance Indicator Verification	
EP PROGRAM INSPECTION - SUSQUEHANNA						1
1, 2	10/25/2021	10/29/2021	IP 71114.02	000717	Alert and Notification System Testing	
1, 2	10/25/2021	10/29/2021	IP 71114.03	000718	Emergency Response Organization Staffing and Augmentation System	
1, 2	10/25/2021	10/29/2021	IP 71114.04	000719	Emergency Action Level and Emergency Plan Changes	
1, 2	10/25/2021	10/29/2021	IP 71114.05	000720	Maintenance of Emergency Preparedness	
1, 2	10/25/2021	10/29/2021	IP 71151	001397	Performance Indicator Verification	
TI-194 Open Phase Condition Inspection - Susquehanna						2
1, 2	10/25/2021	10/29/2021	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)	
Design Basis Assurance Inspection - Programs - Power Operated Valves - Susquehanna Units 1 and 2						3
1, 2	11/1/2021	11/5/2021	IP 71111.21N.02	001645	Design-Basis Capability of Power-Operated Valves Under 10 CFR 50.55a Requirements	
1, 2	11/15/2021	11/19/2021	IP 71111.21N.02	001645	Design-Basis Capability of Power-Operated Valves Under 10 CFR 50.55a Requirements	

IP 22b Site Inspection Activity Plan

SQ Requal Inspection with P/F Results						2
1, 2	11/15/2021	11/19/2021	IP 71111.11A	000703	Licensed Operator Requalification Program and Licensed Operator Performance	
1, 2	11/15/2021	11/19/2021	IP 71111.11B	000704	Licensed Operator Requalification Program and Licensed Operator Performance	
License Renewal Phase 2						4
1, 2	1/10/2022	1/28/2022	IP 71003	000687	Post-Approval Site Inspection for License Renewal	
Rad Safety - Airborne						1
1, 2	1/10/2022	1/14/2022	IP 71124.03	000727	In-Plant Airborne Radioactivity Control and Mitigation	
Access Authorization, FFD- SUS						2
1, 2	1/24/2022	1/28/2022	IP 71130.01	000733	Access Authorization	
1, 2	1/24/2022	1/28/2022	IP 71130.08	000740	Fitness For Duty Program	
License Renewal - Phase 1 - Susquehanna Unit 1						1
1	4/3/2022	4/9/2022	IP 71003	000687	Post-Approval Site Inspection for License Renewal	
Inservice Inspection - Susquehanna Unit 1						1
1	4/3/2022	4/9/2022	IP 71111.08G	000701	Inservice Inspection Activities (BWR)	
Rad Safety - Rad Hazards						1
1, 2	4/11/2022	4/15/2022	IP 71124.01	000725	Radiological Hazard Assessment and Exposure Controls	
PI&R BIENNIAL TEAM INSPECTION - SQ						4
1, 2	6/6/2022	6/10/2022	IP 71152B	000747	Problem Identification and Resolution	
1, 2	6/20/2022	6/24/2022	IP 71152B	000747	Problem Identification and Resolution	
ISFSI Operational (60855) and Dry Run (60854) Inspections						3
1, 2	7/4/2022	7/8/2022	IP 60854	001000	Preoperational Testing Of An ISFSI	
1, 2	7/4/2022	7/8/2022	IP 60856	001003	Review of 10 CFR 72.212(b) Evaluations	
1, 2	7/25/2022	7/29/2022	IP 60855	001001	Operation Of An ISFSI	
Material Control & Accountability - Susquehanna Units 1 and 2						1
1, 2	7/25/2022	7/29/2022	IP 71130.11	000742	Material Control and Accounting (MC&A)	
Design Basis Assurance Inspection - Teams - Susquehanna Units 1 and 2						6
1, 2	8/7/2022	8/13/2022	IP 71111.21M	000713	Design Bases Assurance Inspection (Teams)	
1, 2	8/21/2022	8/27/2022	IP 71111.21M	000713	Design Bases Assurance Inspection (Teams)	
EP EXERCISE INSPECTION - SUSQUEHANNA						4
1, 2	10/17/2022	10/21/2022	IP 71114.01	000716	Exercise Evaluation	
1, 2	10/17/2022	10/21/2022	IP 71151	001397	Performance Indicator Verification	
Access Control, Equip Perf, Sec Trng, SPR, PI-SUS						3
1, 2	10/24/2022	10/28/2022	IP 71130.02	000734	Access Control	
1, 2	10/24/2022	10/28/2022	IP 71130.04	000736	Equipment Performance, Testing, and Maintenance	
1, 2	10/24/2022	10/28/2022	IP 71130.07	000739	Security Training	

IP 22b Site Inspection Activity Plan

1, 2	10/24/2022	10/28/2022	IP 71130.09	001656	Security Plan Changes	
1, 2	10/24/2022	10/28/2022	IP 71151	001338	Performance Indicator Verification	
Rad Safety - Dosimetry & PIV						1
1, 2	11/14/2022	11/18/2022	IP 71124.04	000728	Occupational Dose Assessment	
1, 2	11/14/2022	11/18/2022	IP 71151	000746	Performance Indicator Verification	
Rad Safety - Instruments						1
1, 2	2/6/2023	2/10/2023	IP 71124.05	000729	Radiation Monitoring Instrumentation	
License Renewal Phase 1 - Susquehanna Unit 2						1
2	3/27/2023	3/31/2023	IP 71003	000687	Post-Approval Site Inspection for License Renewal	
Inservice Inspection - Susquehanna Unit 2						1
2	3/27/2023	3/31/2023	IP 71111.08G	000701	Inservice Inspection Activities (BWR)	
Rad Safety - Rad Hazards						1
1, 2	3/27/2023	3/31/2023	IP 71124.01	000725	Radiological Hazard Assessment and Exposure Controls	
EP PROGRAM INSPECTION - SUSQUEHANNA						1
1, 2	5/15/2023	5/19/2023	IP 71114.02	000717	Alert and Notification System Testing	
1, 2	5/15/2023	5/19/2023	IP 71114.03	000718	Emergency Response Organization Staffing and Augmentation System	
1, 2	5/15/2023	5/19/2023	IP 71114.04	000719	Emergency Action Level and Emergency Plan Changes	
1, 2	5/15/2023	5/19/2023	IP 71114.05	000720	Maintenance of Emergency Preparedness	
1, 2	5/15/2023	5/19/2023	IP 71151	001397	Performance Indicator Verification	
Access Control, Protective Strat Review, TS Review, Security Plan Review, PI- SUS						4
1, 2	5/22/2023	5/26/2023	IP 71130.02	000734	Access Control	
1, 2	5/22/2023	5/26/2023	IP 71130.05	000737	Protective Strategy Evaluation	
1, 2	5/22/2023	5/26/2023	IP 71130.09	001656	Security Plan Changes	
1, 2	5/22/2023	5/26/2023	IP 71130.14	000743	Review of Power Reactor Target Sets	
1, 2	5/22/2023	5/26/2023	IP 71151	001338	Performance Indicator Verification	
FY23 Susquehanna Initial License Exam						4
1, 2	6/25/2023	6/30/2023	OV	000956	FB-OR-ONSITE VALIDATION OF INITIAL LICENSE EXAMINATION (OV)	

Prepared on 8/25/2021 6:26:55 PM

<u>September 23, 2021</u> – Letter from Fred L. Bower, III, Chief Security, Emergency Preparedness and Incident Response Branch Division of Radiological Safety and Security to Brad Berryman President and Chief Nuclear Officer Susquehanna Nuclear, LLC with subject of SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 –

SECURITY BASELINE INSPECTION REPORT 05000387/2021402 AND 05000388/2021402

On September 9, 2021, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Susquehanna Steam Electric Station, Units 1 and 2 and discussed the results of this inspection with Mr. Doug LaMarca, Manager Nuclear Operations, 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: 05000387 and 05000388

License numbers: NPF-14 and NPF-22

Report numbers: 05000387/2021402 and 05000388/2021402

Enterprise Identifier: I-2021-402-0074

Licensee: Susquehanna Nuclear, LLC

Facility: Susquehanna Steam Electric Station, Units 1 and 2

Location: Berwick, PA.

Inspection dates: September 7, 2021 to September 9, 2021

Inspectors: D. Caron, Senior Physical Security Inspector

S. McCarver, Physical Security Inspector

Approved by: Fred L. Bower, III, Chief Security, Emergency Preparedness and Incident Response Branch Division of Radiological Safety and Security

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting a security baseline inspection at Susquehanna Steam Electric Station, Units 1 and 2, 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

Unless otherwise noted, baseline security inspections were conducted in accordance with the appropriate portions of the inspection procedures (IPs) in effect at the beginning of the inspection. Samples were declared complete when the IP requirements most appropriate to the inspection activity were met consistent with Inspection Manual Chapter 2201, "Security Inspection Program for Commercial Nuclear Power Reactors." 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. Publicly available IPs are located at: http://www.nrc.gov/reading-rm/doc-collections/insp-manual/inspection-procedure/index.html. 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.

In response to the COVID-19 Public Health Emergency declared on January 31, 2020, by the

Secretary of the United States Department of Health and Human Services on the public health

risks of the coronavirus, Susquehanna Steam Electric Station pursued an exemption to security

officer training and requalification requirements (ML20329A335). In addition to these

exemptions, licensees implemented additional actions or alternatives allowed in response to health concerns within existing regulatory guidance available in Regulatory Guide 5.75 to

ensure annual Force on Force exercises were conducted as safely as possible while

maintaining the requirements to simulate as closely as practicable site specific conditions and

minimizing the number and effects of artificialities associated with these exercises (Part 73,

Appendix B, VI.C.3).

SAFEGUARDS

71130.05 - Protective Strategy Evaluation

The inspectors evaluated the licensee's protective strategy through completion of the following procedure elements:

Protective Strategy Evaluation (1 Sample)

(1) The inspectors observed an annual Force on Force exercise to verify implementation of the Performance Evaluation Program sample.

INSPECTION RESULTS

No findings were identified.

EXIT MEETINGS AND DEBRIEFS

The inspectors verified no proprietary information was retained or documented in this report.

- On September 9, 2021, the inspectors presented the security baseline inspection results to Mr. Doug LaMarca, Manager Nuclear Operations, and other members of the licensee staff.

DOCUMENTS REVIEWED

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
71130.05	Corrective Action Documents	CR-2021-07584 DPA-02-D1-2620-08315		
	Miscellaneous	EC-032-1015	Target Set Determination for SSES Unit 1 and Unit 2	Revision 14

<u>October 25, 2021</u> – Letter from Fred L. Bower, III, Chief Security, Emergency Preparedness and Incident Response Branch Division of Radiological Safety and

Security to Brad Berryman President and Chief Nuclear Officer Susquehanna Nuclear, LLC with subject of SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 – SECURITY BASELINE INSPECTION REPORT 05000387/2021401 AND 05000388/2021401

On October 21, 2021, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Susquehanna Steam Electric Station, Units 1 and 2 and discussed the results of this inspection with Mr. Kevin Cimorelli, 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: 05000387 and 05000388
License numbers: NPF-14 and NPF-22
Report numbers: 05000387/2021401 and 05000388/2021401

Enterprise Identifier: I-2021-401-0093
Licensee: Susquehanna Nuclear, LLC
Facility: Susquehanna Steam Electric Station, Units 1 and 2

Location: Berwick, PA
Inspection dates: October 18, 2021 to October 21, 2021

Inspectors: K. Hussar, Senior Physical Security Inspector

D. Caron, Senior Physical Security Inspector

S. McCarver, Physical Security Inspector

Approved by: Fred L. Bower, III, Chief Security, Emergency Preparedness and Incident Response Branch Division of Radiological Safety and Security

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting a security baseline inspection at Susquehanna Steam Electric Station, Units 1 and 2, 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

Unless otherwise noted, baseline security inspections were conducted in accordance with the appropriate portions of the inspection procedures (IPs) in effect at the beginning of the inspection. Samples were declared complete when the IP requirements most appropriate to the inspection activity were met consistent with Inspection Manual Chapter 2201, "Security Inspection Program for Commercial Nuclear Power Reactors." 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. Publicly available IPs are located at: http://www.nrc.gov/reading-rm/doc-collections/insp-manual/inspection-procedure/index.html. 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.

SAFEGUARDS

71130.02 - Access Control

The inspectors evaluated the access control program through completion of the following inspection elements:

Access Control (1 Sample)

(1) • • •

Tier I: All Requirements Tier II: All Requirements Tier III: All Requirements

71130.04 - Equipment Performance, Testing, and Maintenance

The inspectors evaluated the security equipment testing and maintenance program through completion of the following inspection elements:

Equipment Performance, Testing, and Maintenance (1 Sample)

(1) • • •

Tier I: All Requirements
Tier II: 8 Requirements (02.09 a., b., 02.10, 02.11 a. - e.) Tier III: 2 Requirements (02.14, 02.15)

71130.07 - Security Training

The inspectors evaluated the security training program through completion of the following inspection elements:

Security Training (1 Sample)

(1) • Tier I: All Requirements • Tier II: All Requirements • Tier III: All Requirements

71130.09 - Security Plan Changes

The inspectors evaluated the security plan changes through completion of the following inspection elements:

Review Security Plan Changes (IP Section 02.01) (1 Sample)

(1) The opportunity to apply this procedure was not available in accordance with Inspection Manual Chapter 0306. This sample was not available because the licensee did not conduct the activity covered by this IP. Specifically, the licensee has not initiated a physical security plan change in accordance with 10 CFR 50.54(p)(2) since the last performance of this IP.

OTHER ACTIVITIES – BASELINE

71151 - Performance Indicator Verification
The inspectors verified licensee performance indicators submittals listed below:

PP01: Protected Area Security Equipment Performance Index Sample (IP Section 02.17) (1 Sample)

(1) October 1, 2020 through September 30, 2021

INSPECTION RESULTS

No findings were identified.

EXIT MEETINGS AND DEBRIEFS

The inspectors verified no proprietary information was retained or documented in this report.

• On October 21, 2021, the inspectors presented the security baseline inspection results to Mr. Kevin Cimorelli, Site Vice President, and other members of the licensee staff.

DOCUMENTS REVIEWED

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
71130.02	Procedures	NDAP-QA-0902	Safeguards Information Program and the Protection of Sensitive Unclassified Non-Safeguards Information	Revision 23
		NS-SO-004	Reporting/Alerting and Response to Alarms	Revision 13
		NS-SSP-002	Control, Issuance and Assignment of Security Controlled Locks and Keys	Revision 21
		SI-SO-002	Duties and Responsibilities of the Alarm Station Operators	Revision 56
		SI-SO-003	Duties and Responsibilities of the Access Control Officer	Revision 48
		SI-SO-004	Duties and Responsibilities of the Mobile Patrol Officer	Revision 27
		SI-SO-005	Duties and Responsibilities of Interior Patrol	Revision 27
		SI-SO-006	Duties and Responsibilities of the South Gate House Search Officer	Revision 74
		SI-SO-008	Emergency Access and Egress	Revision 22
		SI-SO-013	Personnel Access Control to Manholes/Handholes, Cable Chase Hatches, Panels and Penetrations	Revision 11
		SI-SO-036	Searches Conducted Outside the OCA	Revision 10
		SI-SO-038	Vital Area Access During Periods of Site Specific Credible Threat to Include Insider Threat	Revision 2
		SI-SO-21	Compensatory Measures for Components of the Susquehanna Security Computer System	Revision 41
71130.04	Corrective Action Documents	CR-2020-11733 CR-2020-15327 CR-2020-15936 CR-2021-03697 CR-2021-09907		
	Engineering Changes	EC 2127066	Perform Site Acceptance Test for Fiber Optic Intrusion Detection System (IDS) Per TD-031-008	
	Procedures	MT-032-001	Security System Performance Testing	Revision 30
		MT-032-002	Security Control Center 0D575 UPS Calibration	Revision 5
		MT-032-003	Security Control Center 0D585 UPS Calibration	Revision 3
		NS-SO-002	Security Communications	Revision 18
		NS-SO-004	Reporting/Alerting and Response to Alarms	Revision 13
		NS-SSP-003	Tests, Checks, and Inspections of Security Systems	Revision 56

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		NS-SSP-004	Tests, Checks and Inspections of the Susquehanna Security Computer System	Revision 58
		SI-SO-004	Duties and Responsibilities of the Mobile Patrol Officer	Revision 27
		SI-SO-005	Duties and Responsibilities of Interior Patrol	Revision 27
		SI-SO-006	Duties and Responsibilities of the South Gate House Search Officer	Revision 74
71130.07	Corrective Action Documents	CR-2020-10459 CR-2020-13813		
	Corrective Action Documents Resulting from Inspection	DI-2021-13011 CR-2021-15075		
	Miscellaneous		Susquehanna Nuclear, LLC Training and Qualification Plan	Revision 16
		LP 22	Comply With Physical Fitness Performance Requirements	Revision 5
	Procedures	SI-ST-001	Operation of the Weapons Firing Range	Revision 13
		SI-ST-002	Security and Safe Handling of Firearms, Ammunition, and Associated Equipment	Revision 38
		SI-ST-003	Nuclear Security Training Program	Revision 13

<u>November 3, 2021</u> – Letter from Jonathan E. Greives, Chief Reactor Projects Branch 4 Division of Operating Reactor Safety to Brad Berryman Senior Vice President and Chief Nuclear Officer Susquehanna Nuclear, LLC with subject of SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 – INTEGRATED INSPECTION REPORT 05000387/2021003 AND 05000388/2021003

On September 30, 2021, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Susquehanna Steam Electric Station, Units 1 and 2. On October 28, 2021, the NRC inspectors discussed the results of this inspection with you and other members of your staff. The results of this inspection are documented in the enclosed report.

Two findings of very low safety significance (Green) are documented in this report. One of these findings 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 Susquehanna Steam Electric Station, Units 1 and 2.

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 Resident Inspector at Susquehanna Steam Electric Station, Units 1 and 2.

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: 05000387 and 05000388
Licensee numbers: NPF-14 and NPF-22
Report numbers: 05000387/2021003 and 05000388/2021003

Enterprise Identifier: I-2021-003-0017
Licensee: Susquehanna Nuclear, LLC
Facility: Susquehanna Steam Electric Station, Units 1 and 2

Location: 769 Salem Blvd., Berwick, PA
Inspection dates: July 1, 2021, to September 30, 2021

Inspectors: C. Highley, Senior Resident Inspector
M. Rossi, Resident Inspector
H. Anagnostopoulos, Senior Health Physicist J. DeBoer, Reactor Inspector
B. Edwards, Health Physicist
N. Floyd, Senior Reactor Inspector
M. Henrion, Health Physicist
A. Turilin, Reactor Inspector

Approved by: Jonathan E. Greives, 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 Susquehanna Steam Electric Station, Units 1 and 2, 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 Perform an Appropriate Critique to Ensure an Incorrect Emergency Action Level Classification Was Corrected			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Emergency Preparedness	Green NCV 05000387,05000388/2021003-01 Open/Closed	[H.9] - Training	71111.05
The inspectors identified a Green non-cited violation (NCV) of 10 CFR 50.54(q)(2) when the licensee failed to follow and maintain an emergency plan that meets the requirements of 10 CFR Part 50, Appendix E, and the planning standards of 10 CFR 50.47(b). Specifically, the licensee failed to identify and correct a weakness when performing emergency classifications during a fire drill as required by 10 CFR 50.47(b)(14).			

Electrohydraulic Control System Leak Due to Improper Maintenance			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green FIN 05000387/2021003-02 Open/Closed	[H.12] - Avoid Complacency	71111.12
The inspectors identified a finding (FIN) of very low significance (Green) when plant maintenance personnel failed to properly torque the tie rods that hold the bottom plate on the 'D' main turbine steam bypass valve which resulted in a leak on the electrohydraulic control (EHC) system.			

Additional Tracking Items

Type	Issue Number	Title	Report Section	Status
LER	05000388/2020-001-01	LER 2020-001-01 for Susquehanna, Unit 2, Manual Scram Due to Rising Main Condenser Backpressure Caused by Failure of an Offgas Recombiner Inlet Valve	71153	Closed

PLANT STATUS

Unit 1 began the inspection period at rated thermal power. On July 21, 2021, the unit was shut down due to a scram. The unit was returned to rated thermal power on August 2, 2021. On August 13, 2021, the unit was down powered to 87 percent for a rod pattern adjustment. The unit was returned to rated thermal power on August 14, 2021. On

September 1, 2021, the unit was down powered to 85 percent for turbine valve testing and rod pattern adjustment. The unit was restored to rated thermal power on September 2, 2021. On September 14, 2021, the unit was down powered to 85 percent for a rod pattern adjustment. The unit was returned to 100 percent on September 14, 2021. On September 21, 2021, the unit was down powered to 94 percent for a rod pattern adjustment. The unit was returned to rated thermal power on September 22, 2021. On September 24, 2021, the unit was down powered to 92 percent for control rod friction testing. The unit was returned to rated thermal power on September 25, 2021, and remained at or near rated thermal power for the remainder of the inspection period.

Unit 2 began the inspection period at rated thermal power. On July 9, 2021, the unit was down powered to 60 percent for a rod sequence exchange. The unit was returned to rated thermal power on July 12, 2021. On July 13, 2021, the unit was down powered to 93.8 percent. The unit was returned to rated thermal power on July 14, 2021, and remained at or near rated thermal power 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 on-site activities, resident inspectors conducted plant status activities as described in IMC 2515, Appendix D, “Plant Status”; conducted routine reviews using IP 71152, “Problem Identification and Resolution”; 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 External Flooding (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 1, 2021.

71111.04 - Equipment Alignment
Partial Walkdown (IP Section 03.01) (2 Samples)

The inspectors evaluated system configurations during partial walkdowns of the following systems/trains:

(1) Unit 1, division II emergency core cooling systems while 'B' loop residual heat removal in shutdown cooling mode on July 23, 2021

(2) Unit 1, 1B core spray during 1A core spray system outage window on September 27, 2021

71111.05 - Fire Protection
Fire Area Walkdown and Inspection (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) Unit Common, 'A' emergency diesel bay, 660-foot to 710-foot elevation (FZ 0-41A), on July 23, 2021
2. (2) Unit 1, core spray pump room and high-pressure coolant injection pump room, 645-foot elevation (FZ 1-1B and 1-1C), on August 4, 2021
3. (3) Unit 1, equipment access area, 683-foot elevation (FZ 1-3C, N, W, and S), on August 18, 2021
4. (4) Unit 2, divisions I and II equipment rooms, 771-foot elevation (FZ 0-28A-I and 0-28A-II), on August 27, 2021
5. (5) Unit 1, control structure battery rooms, 771-foot elevation (FZ 0-28B-I, M, N, and J), on September 15, 2021
6. (6) Unit 2, reactor coolant isolation cooling pump room, 645-foot to 670-foot elevation, residual heat removal 'B' pump room, 645-foot to 670-foot elevation, sump pump room, 645-foot elevation (FZ 2-1D, 2-1E, 2-1G), on September 21, 2021

Fire Brigade Drill Performance (IP Section 03.02) (2 Samples)

1. (1) The inspectors evaluated the onsite fire brigade training and performance during an unannounced fire drill on July 9, 2021.
2. (2) The inspectors evaluated the onsite fire brigade training and performance during an announced fire drill with off-site personnel participation on September 15, 2021.

71111.06 - Flood Protection Measures
Cable Degradation (IP Section 03.02) (1 Sample)

The inspectors evaluated cable submergence protection in:

(1) Manholes 31 and 32 on September 14, 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 Unit 2 rod sequence exchange on July 9, 2021.

Licensed Operator Requalification Training/Examinations (IP Section 03.02) (1 Sample)

(1) Unit Common, the inspectors observed and evaluated operator performance in the simulator during performance of licensed operator requalification exam that included reactor scram, seismic event, loss of emergency core cooling equipment, and various pump and valve failures to include emergency action level (EAL) classifications on September 14, 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) Unit Common, review of AZZ-NLI breakers impending change from (a)(1) status under the maintenance rule program on August 19, 2021

(2) Unit 1, bypass valve maintenance practices and work order discrepancies on September 8, 2021.

71111.13 - Maintenance Risk Assessments and Emergent Work Control Risk Assessment and Management (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 1, yellow risk due to automatic depressurization system permissive and timer sequence testing on July 6, 2021
2. (2) Unit 1, protected equipment scheme during forced outage with 'B' loop shutdown cooling in service on July 23, 2021

(3) Unit Common, yellow risk window for the relay replacement on division I residual heat removal service water and diagnostic testing of division I emergency service water HV10943A2 on August 25, 2021
(4) Unit 2, scram discharge and vent/drain valve (SV-247F009A) solenoid replacement on August 26, 2021
(5) Unit Common, elevated risk for spray pond header cleaning of A1 spray nozzles on September 29, 2021

71111.15 - Operability Determinations and Functionality Assessments

Operability Determination or Functionality Assessment (IP Section 03.01) (9 Samples)

The inspectors evaluated the licensee's justifications and actions associated with the following operability determinations and functionality assessments:

(1) Unit 2, emergency service water to turbine building closed loop cooling water crossover through-wall leak as documented in CR-2021-09529 on July 7, 2021
(2) Unit Common, degraded fire wrap on conduit in 'A' emergency diesel bay as documented in CR-2021-11107 on August 16, 2021
(3) Unit 1, reactor core isolation cooling pump high oil level as documented in CR-2021-09559 on August 17, 2021
(4) Unit Common, divisions I and II emergency service water and residual heat removal service water control power cable tray corrosion in manholes 27 and 28 (MH27 and MH28) as documented in CR-2021-12561 and CR-2021-12566 on August 24, 2021
(5) Units 1 and 2, operability of steam bypass valves with tie rods being under torqued as documented in CR-2021-12659 on August 26, 2021
(6) Unit Common, control building heating, ventilation, and air conditioning surveillance failure which Engineering reperformed the calculation to remove some margin to pass the surveillance as documented in CR-2021-12739 on August 26, 2021
(7) Unit 1, spring cans identified to be bottom limit position as documented in CR-2021-11076 on September 1, 2021
(8) Unit 2, residual heat removal service water pump discharge pressure in alert range as documented in CR-2021-09456 on September 2, 2021
(9) Unit 2, residual heat removal heat exchanger outlet valve (HV21215A) overload trips on restoration as documented in CR-2021-13814 on September 22, 2021

71111.17T - Evaluations of Changes, Tests, and Experiments

Sample Selection (IP Section 02.01) (38 Samples)

The inspectors reviewed the following evaluations, screenings, and/or applicability determinations for 10 CFR 50.59 from September 20 to 23, 2021:

(1) 50.59 SE 00031, Evaluation of Framatome Computer Code Version Changes on U2C21 Analyses, Revision 0
(2) 50.59 SE 00032, Decrease in Turbine Valve Testing Frequency, Revision 0
(3) 50.59 SD 01502, Install Higher Head Pumps in Place of 0P511, 0P512, and 0P592, Revision 4
(4) 50.59 SD 02184, Check Valve Hinge Pin Material Change, Revision 0
(5) 50.59 SD 02188, Alternate Proprietary Concrete, Revision 0

(6) 50.59 SD 02194, Removing Seismic Sensor VT05704 and Its Recorder VRS05704, Revision 0
(7) 50.59 SD 02198, Add a Hydraulic Booster to the EDG E Governor, Revision 0
(8) 50.59 SD 02205, Nuclear Engineering Specification for Replacement of HRC (ESW and RHRSW) and JRD (Service Water) Piping or Repair of HRC (ESW and RHRSW) Piping, Revision 1
(9) 50.59 SD 02206, A-D Diesel Generator Exhaust Outlet Heat Shield, Revision 0
(10) 50.59 SD 02208, Replace 480V LC Transformers Per EC 2206354, Revision 0
(11) 50.59 SD 02235, Clearance Order with Valves Out of Position Greater Than 90 Days, Revision 0
(12) 50.59 SD 02240, NDAP-QA-0742, Revision 6
(13) 50.59 SD 02242, Stainless Steel Piping for Mitigation of FAC and Mechanical Erosion, Revision 3
(14) 50.59 SD 02257, Temporary Leak Repairs, Revision 1
(15) 50.59 SD 02259, Determ Control Valve Test Switch (CVTS-3) to Allow for Routine Testing of Circuit for Failed Limit Switch, Revision 0
(16) 50.59 SD 02261, Local Leak Rate Testing Scope Reduction for SSES Units 1 and 2, Revision 1
(17) 50.59 SD 02266, Add a Hydraulic Booster to the EDG A, B, C, D Mechanical Governor, Revision 0
(18) 50.59 SD 02277, Security SIEM Replacement, Revision 0
(19) 50.59 SD 02278, Revise FSAR Section 9.5.2.2.2 per EC 2217021, Revision 0
(20) 50.59 SD 02289, Replace U1 Standby Liquid Control Tank Ultrasonic Level Monitoring Equipment, Revision 0
(21) 50.59 SD 02291, Maintaining Door 571 R in the Open Position, Revision 0
(22) 50.59 SD 02297, Fuel Pool Cooling FD Flow Controller FIC-25444 Caution Tagged in Manual Control, Revision 0
(23) 50.59 SD 02333, TDC 2308928 Temporary Bypass Battery Cell #120 on Battery 2D660, 250VDC Battery Bank B, Revision 0
(24) 50.59 SD 02340, Caution Tag CO 24-001-2301047-0 Applied to A Diesel Generator Aux Fuel Oil Booster Pump 0P538A Handswitch HS-03483A for Greater Than 60 Days, Revision 0
(25) 50.59 SD 02341, Leakage Rate Assignment to MSIV Penetrations No. X-7A, B, C, D, Revision 0
(26) 50.59 SD 02349, SCT Applied to Remove Power to Cathodic Protection Circuit R10A and R10B (Breaker 0LP4-22), Revision 0
(27) 50.59 SD 02352, Issuance, Revision, or Deletion of the License Renewal System Walkdown Program, Revision 0
(28) 50.59 SD 02361, ESW Flow Balance Throttle Valve Adjustments, Revision 0
(29) 50.59 SD 02376, Replacement Pressure Switch for Core Spray and LPCI Permissive Pressure Switches, Revision 0
(30) 50.59 SD 02392, Evaluate Compensatory Measure Support POD ACT-01-CR-2020-10890, Revision 0
(31) 50.59 SD 02405, Compensatory Measure for POD ACT-01-CR-2020-13332, Revision 0
(32) 50.59 SD 02420, FSAR Section 3.14.2.28, and FSAR Table 3.14-1, Item 48: Addition of Lubricating Oil Analysis Program Exception for the Diesel Engine Driven Fire Pumps, Revision 0
(33) 50.59 SD 02440, Pandemic Response, Revision 1

(34) 50.59 SD 02446, Unit 2 500KV Tie-In for Commercial Customer Load Center, Revision 1
(35) 50.59 SD 02460, DBD 024 DBDCN for Circulating Water Pump Suct/Disch Valve Replacement, Revision 0
(36) 50.59 SD 02461, Generic 50.59 Screen for Polyethylene Foam from TN-B1 Shipping Container Lost in Reactor Vessel, Revision 0
(37) 50.59 SD 02473, 0K112A Hight Bearing Temp Trip Temp Bypass, Revision 0
(38) 50.59 SD 02500, Housekeeping Patch Installed Over Pinhole Leak on HRC205-2 ESW Piping, Revision 0

71111.19 - Post-Maintenance Testing

Post-Maintenance Test (IP Section 03.01) (3 Samples)

The inspectors evaluated the following post-maintenance test activities to verify system operability and functionality:

(1) Unit 1, bypass valve 'D' O-ring replacement during forced outage on August 2, 2021
(2) Unit 2, residual heat removal flow verification after breaker motor cut out switch replacement on August 12, 2021
(3) Unit Common, 'C' emergency diesel generator 5-year overhaul on August 30, 2021

71111.20 - Refueling and Other Outage Activities

Refueling/Other Outage (IP Section 03.01) (1 Sample)

(1) The inspectors evaluated Unit 1 forced outage, scram due to turbine trip from an electrical short in the C phase of the Isophase ducting, activities from July 21 to July 27, 2021.

71111.22 - Surveillance Testing

The inspectors evaluated the following surveillance tests:

Inservice Testing (IP Section 03.01) (1 Sample)

(1) Unit Common, 'A' loop emergency service water comprehensive flow surveillance on August 23, 2021

71114.06 - Drill Evaluation

Select Emergency Preparedness Drills and/or Training for Observation (IP Section 03.01) (1 Sample)

(1) Full participation emergency planning practice drill on August 5, 2021

RADIATION SAFETY

71124.08 - Radioactive Solid Waste Processing and Radioactive Material Handling, Storage, and Transportation

Radioactive Material Storage (IP Section 03.01) (2 Samples)

The inspectors evaluated the licensee's performance in controlling, labelling, and securing radioactive materials as follows:

(1) Low-level radwaste storage facility
(2) 660-foot elevation of the radwaste facility

Radioactive Waste System Walkdown (IP Section 03.02) (2 Samples)

The inspectors walked down accessible portions of the solid radioactive waste systems and evaluated system configuration and functionality, including:

(1) Spent resin tank room and associated equipment
(2) Filter disposal in the radwaste liner processing area

Waste Characterization and Classification (IP Section 03.03) (2 Samples)

The inspectors evaluated the licensee's characterization and classification of radioactive waste, including:

(1) Results of the 10 CFR, Part 61, characterization of dry active waste taken in 2019
(2) Results of the 10 CFR, Part 61, characterization of control rod drive mechanisms taken in 2021

Shipment Preparation (IP Section 03.04) (1 Sample)

Complete – opportunity to apply full procedure not available in accordance with IMC 0306.

(1) There were no shipments of radioactive material during the inspection period.

Shipping Records (IP Section 03.05) (4 Samples)

The inspectors evaluated the following non-excepted radioactive material shipments through a record review:

(1) 20-009
(2) 20-063
(3) 21-006
(4) 21-023

OTHER ACTIVITIES – BASELINE

71151 - Performance Indicator Verification

The inspectors verified licensee performance indicators submittals listed below:

MS05: Safety System Functional Failures (IP Section 02.04) (2 Samples)

(1) Unit 1 (July 1, 2020, through June 30, 2021)
(2) Unit 2 (July 1, 2020, through June 30, 2021)

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) Wall thinning in the Unit 2 reactor vessel bottom head drain piping

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 reports (LERs):

(1) LER 05000388/2020-001-01, Manual Scram Due to Rising Main Condenser Backpressure Caused by Failure of an Offgas Recombiner Inlet Valve (ADAMS Accession No. ML20310A258). The inspectors determined that the cause of the condition described in the LER was not reasonably within the licensee's ability to foresee and correct and therefore was not reasonably preventable. No performance deficiency nor violation of NRC requirements was identified.

INSPECTION RESULTS

<table>
<tr><td colspan="4">Failure to Perform an Appropriate Critique to Ensure an Incorrect Emergency Action Level Classification Was Corrected</td></tr>
<tr><td>Cornerstone</td><td>Significance</td><td>Cross-Cutting Aspect</td><td>Report Section</td></tr>
<tr><td>Emergency Preparedness</td><td>Green
NCV 05000387,05000388/2021003-01
Open/Closed</td><td>[H.9] - Training</td><td>71111.05</td></tr>
<tr><td colspan="4">The inspectors identified a Green NCV of 10 CFR 50.54(q)(2) when the licensee failed to follow and maintain an emergency plan that meets the requirements of 10 CFR Part 50, Appendix E, and the planning standards of 10 CFR 50.47(b). Specifically, the licensee failed to identify and correct a weakness when performing emergency classifications during a fire drill as required by 10 CFR 50.47(b)(14).</td></tr>
<tr><td colspan="4">Description: During the performance of an announced fire drill, scenario 18A, on June 23, 2021, in which the non-safety-related engineered safeguards system transformer (OX211) was simulated as on fire, the shift manager and shift technical assistant made an EAL</td></tr>
</table>

classification of SA8.1 hazardous event affecting safety systems needed for the current operating mode (mode 1, 2, or 3). The shift manager based the call on any occurrence of table S-4 event (fire), and the event damage has caused indications of degraded performance of at least one train of safety system needed for the current operating mode or the event has caused visible damage to a safety system component or structure needed for the current operating mode. Specifically, the crew made the declaration based on their misunderstanding that the transformer was safety related. A critique at the conclusion of the fire drill, which was observed by the resident inspectors, determined that the EAL classification was correct and the critique writeup documented the EAL classification as satisfactory with no comments. Inspectors reviewed the critique package and determined that this was incorrect because, as previously stated, the transformer is non-safety-related and the EAL criteria was not met. In review of inspectors concerns and following benchmarking of five other licensees, the licensee concluded that the EAL classification that was made by the crew was incorrect and the weakness had not been appropriately critiqued. A review was conducted of the number of times this drill was performed from 2015 to 2021. The licensee determined that scenario 18A had been performed 14 times, of which the EAL classification was evaluated as being satisfactory 10 times and not being observed 4 times.

Corrective Actions: The licensee entered the issue into the corrective action program and identified corrective actions to conduct operator training regarding the review of safety-related versus non-safety-related classification of systems.

Corrective Action References: CR-2021-09708, CR-2021-10886, and AR 2021-09730

Performance Assessment:

Performance Deficiency: The inspectors determined that the failure to critique an incorrect EAL classification during a training evolution was a performance deficiency that was within the licensee's ability to foresee and correct and should have been prevented. Specifically, 10 CFR 50.54(q)(2) requires a licensee to follow and maintain an emergency plan that meets the requirements of 10 CFR Part 50, Appendix E; and the planning standards of 10 CFR 50.47(b) require a licensee to identify and correct weaknesses.

Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the Emergency Response Organization Performance attribute of the Emergency Preparedness cornerstone and adversely affected the cornerstone objective to ensure that the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. Specifically, because the licensee did not effectively identify and critique an emergency preparedness drill performance weakness during an announced fire drill, this caused a missed opportunity to identify and correct a drill-related performance deficiency. Failing to identify and correct operators' understanding of the applicability of the EAL could result in an incorrect implementation of the emergency plan in an actual emergency.

Significance: The inspectors assessed the significance of the finding using Appendix B, "Emergency Preparedness Significance Determination Process." The inspectors evaluated the finding using IMC 0609, Attachment 4, "Initial Characterization of Findings," issued December 20, 2019. The attachment instructs the inspectors to utilize IMC 0609, Appendix B, "Emergency Preparedness Significance Determination Process," issued September 22, 2015, when the finding is in the licensee's Emergency Preparedness cornerstone. The inspectors determined that this finding was a critique finding related to planning standard 10 CFR 50.47(b)(14) where the critique process did not identify a classification weakness during a limited facility interaction drill (fire drill) in which there is a limited team of evaluators. Therefore, using Figure 5.14-1, "Significance Determination for Critique Findings," the inspectors determined the finding was of very low safety significance (Green).

Cross-Cutting Aspect: H.9 - Training: The organization provides training and ensures knowledge transfer to maintain a knowledgeable, technically competent workforce and instill nuclear safety values. Specifically, the licensee did not show an adequate understanding of component designation by having the drill guide provide the incorrect classification; and the deficiency was not identified during the critique of the training fire drill on 10 separate occasions.

Enforcement:

Violation: 10 CFR 50.54(q)(2) requires, in part, that the licensee shall follow and maintain the effectiveness of an emergency plan that meets the requirements in Appendix E to this part and the planning standards of 10 CFR 50.47(b). 10 CFR 50.47(b)(14) requires, in part, that periodic drills are conducted to develop and maintain key skills, and that deficiencies identified as a result of exercises are corrected. Section IV.F.2.g of Appendix E to 10 CFR Part 50 requires that all exercises, drills, and training that provide performance opportunities must provide for formal critiques in order to identify weak or deficient areas that need correction. Any weaknesses or deficiencies that are identified in a critique must be corrected. Contrary to the above, the licensee failed to identify and correct a performance weakness during the critique of the June 23, 2021, fire drill. Specifically, the licensee did not identify that the incorrect EAL determination was made by the shift manager. In review of the issue, it was identified that the violation had occurred on 10 other occasions as far back as September 23, 2015.

Enforcement Action: This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy.

Electrohydraulic Control System Leak Due to Improper Maintenance			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green FIN 05000387/2021003-02 Open/Closed	[H.12] - Avoid Complacency	71111.12
The inspectors identified a finding (FIN) of very low significance (Green) when plant maintenance personnel failed to properly torque the tie rods that hold the bottom plate on the 'D' main turbine steam bypass valve which resulted in a leak on the EHC system.			

Description: The main turbine steam bypass system is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. It allows excess steam flow from the reactor to the condenser without going through the turbine. The full bypass capacity of the system is approximately 22 percent of the nuclear steam supply system rated steam flow. The main turbine bypass system fast opening and pressure regulation modes are required to be operable to limit the pressure increase in the main steam lines and reactor pressure vessel during transients that cause a pressurization so that the safety limit minimum critical power ratio and linear heat generation rate are not exceeded. Licensing analyses credit an operable main turbine bypass system as having both the bypass valve fast opening mode and pressure regulation mode. The fast-opening mode is required for transients initiated by a turbine control valve or turbine stop valve closure. The pressure regulation mode is required for transients where the power increase exceeds the capability of the turbine control valves.

The tie rods on the bypass valves provide proper support for the bottom plate of the bypass valves. These tie rods are pressure-retaining components used to ensure that the bypass valves are available to control pressure during post-scram recovery under normal circumstances. High-pressure EHC fluid is supplied to the bypass valve to position the valve in either the open or closed position depending on the system demand. The tie rods maintain the integrity of the EHC pressure boundary during normal valve operation.

On June 12, 2021, the licensee identified an adverse trend in EHC reservoir levels, indicating a potential leak. Subsequently, while investigating on June 17, 2021, the operators identified a leak on bypass valve 'D'. On June 26, 2021, station personnel made entry to effect repairs, documenting in CR-2021-09661, that of the four tie rods, one was on the scaffolding, one was extremely loose and near falling out, and the two remaining tie rods were not torqued. As part of the repair plan, station personnel reinstalled the missing tie rod and torqued the remaining tie rods. In addition, it was identified during planning that the 'D' and 'E' bypass valves were the two most recently worked valves during the prior refueling outage in the spring of 2020, and performed an extent of condition, noting that the tie rods on 'E' bypass valve were torqued, with torque paint applied.

The NRC inspectors reviewed work orders from the Unit 1 refueling outage in 2020 and discovered that both 'D' and 'E' bypass valves were marked torqued to 250 ft.-lbs. However, from comparison to the work performed on June 17, 2021, the torque values were listed as 550 ft.-lbs. The inspectors requested clarification from the licensee on the correct torque values for the tie rods, which was later determined to be 750 ft.-lbs. (CR-2021-12291). The licensee performed an extent of condition based on this inspection effort and discovered that 8 of the 10 bypass valves were not torqued to the design value of 750 ft.-lbs., and the remaining two bypass valves did not have torque values listed in their prior work orders. The licensee performed a prompt operability determination (CR-2021-12659) and concluded that at 250 ft.-lbs., the lowest currently installed torque value, the bypass valve would remain in the operable-degraded state until the deficiency could be corrected at the next outage.

Corrective Actions: The licensee maintenance personnel reinstalled the missing tie rod, returned all bolts to the torqued condition, performed an extent of condition for remaining bypass valves, and a prompt operability determination to ensure that the other bypass vales would remain operable until deficient torque values could be corrected at the next opportunity.

Corrective Action References: CR-2021-09661, CR-2021-12291, and CR-2021-12659

Performance Assessment:

Performance Deficiency: Susquehanna Work Order ERPM 2190209, Work Instruction M8758-04, Section 6.5, specifies installing and torqueing tie rods in accordance with Torqueing Guidelines, MT-GM-015, during reassembly of the bypass valve. Contrary to the above, the licensee did not install and torque tie rods in accordance with procedural requirements on April 22, 2020, which resulted in the subsequent EHC leak.

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, failure to appropriately torque the tie rods on the 'D' bypass valve resulted in a significant EHC fluid leak from the valve and challenged the reliability of it to operate. Significance: The inspectors assessed the significance of the finding using Appendix A, "The Significance Determination Process for Findings At-Power." This finding was screened to Green using the Mitigating Systems screening questions because it did not involve a design deficiency, did not represent the loss of a train of technical specification equipment for longer than its allowed outage time, and did not represent loss of a probabilistic risk analysis system or function. 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. Specifically, during review of the bypass valve work orders, the NRC inspectors noted that the instructions to complete the installation and torqueing of the tie rods was marked as complete; however, the tie rods were not torqued. The NRC inspectors determined that individual contributors did not perform a thorough review of the planned activity to ensure a successful outcome and did not use an appropriate error reduction tool, such as peer checking the as-left torque settings, that would have prevented this event from occurring.
Enforcement: Inspectors did not identify a violation of regulatory requirements associated with this finding.

Observation: Wall Thinning in the Unit 2 Reactor Vessel Bottom Head Drain Piping	71152
The NRC inspectors performed a detailed review of the licensee's root cause analysis and corrective actions associated with condition report CR-2021-04937 for wall thinning identified in the Unit 2 reactor pressure vessel bottom head drain line during the spring 2021 refueling outage. Specifically, the licensee performed an ultrasonic thickness examination of a 2-inch x 4-inch expander (Component ID DBA-221-1-9625-X) located in the bottom head drain line. Near the weld on the 2-inch side of the pipe expander, the licensee discovered the lowest measured wall thickness (0.125 inches) was less than the minimum allowable wall thickness (0.139 inches). As part of the review, the inspectors conducted in-person interviews with the responsible licensee staff to discuss the results of the root cause investigation, the extent of condition for similar components in Unit 1, and the corrective action activities to address this condition. The licensee completed a root cause analysis in accordance with their procedures and determined the direct cause of the wall thinning was due to flow accelerated corrosion (FAC) in the expander and adjacent bottom head drain piping. The licensee completed a Failure Analysis-Support/Refute Matrix, Equipment Failure Checklist, Barrier Analysis, and Organizational and Programmatic Worksheet from their corrective action procedures to evaluate the problem. The licensee concluded the root cause was "the bridging strategy to manage station FAC risk over-relied on industry guidance and Checworks when monitoring wear rates for the bottom head drain piping." The licensee documented the supporting basis that degradation of the expander would have been identified prior to approaching the minimum wall thickness if the station had taken a more proactive approach to inspect the expander. In response to the root cause, the licensee revised their procedure	

NEPM-QA-1172, "Guidelines for Flow-Accelerated Corrosion Program Activities," to include a decision tree with requirements for additional risk mitigation actions if there is a high difficulty and/or uncertainty when performing the ultrasonic thickness exams. The licensee also developed a contingency plan for Unit 1 to ultrasonically examine the bottom head drain line components during the next scheduled outage and replace the expander, if necessary. The inspectors noted there was no indication of similar wall thinning in Unit 1 based on a review of past thickness data.

In response to the identified wall thinned condition on the pipe expander, the licensee performed immediate corrective actions to evaluate the as-found condition to applicable ASME Code Section III design requirements for this ASME class 1 component and found the expander met design requirements. The licensee's evaluation also assessed the as-found condition for one additional cycle of operation, including conservative assumptions regarding fatigue cycles, to justify continued operation of Unit 2. The licensee staff performed valve line-up changes via their modification process to isolate flow through the Unit 2 bottom head drain line to prevent further wear from FAC. The licensee also considered other potential degradation mechanisms (pitting and crevice corrosion) as a result of the modification and concluded the expander met design requirements for an additional operating cycle.

The NRC inspectors reviewed the analysis and modification and documented the results in the Susquehanna Steam Electric Station, Units 1 and 2 – Integrated Inspection Report 05000387/2021002 and 05000388/2021002 (ML21216A077). The inspectors determined that the licensee planned corrective actions to replace the Unit 2 degraded expander with non-susceptible material during the next refueling outage, conduct metallurgical examinations and analysis of the removed expander, and update the root cause report based on the results of the failure analysis.

The inspectors further reviewed the root cause report, the associated Checworks model, ultrasonic thickness examination history, FAC procedures, industry guidance documents, extended power uprate analysis, and original fabrication records. The inspectors noted the flow through the bottom head drain line was initially designed for 63 GPM and was increased up to an average of 240 GPM due to various modifications and power uprates. While this flow is high compared to the diameter of piping, the inspectors verified that the licensee appropriately accounted for the flow in their FAC susceptibility analyses and had entered this flowrate into their Checworks model for predicting wear. The inspectors further noted that the extent of the FAC wear was localized to the 2-inch x 4-inch expander and that limited wear was measured on the upstream elbow, which was unexpected given the conditions for FAC susceptibility increases in elbows due to geometrically induced flow disturbances.

The inspectors noted the root cause analysis investigation effort did not appear to include retrieval of original fabrication pre-service weld examinations and fit-up of the expander. In requesting and reviewing these documents, the inspectors noted that three weld repairs, including two that reopened the weld root, were performed on the 2-inch side of the expander at the time of construction which is the area of interest for wall thinning. The inspectors viewed these repairs as potential contributors to the localized FAC wear given the potential for geometric discontinuities on the inner diameter from "weld root drop-through" which could cause flow disturbances that increase FAC wear, or "weld undercut" which could show itself as a thinned area. The inspectors noted the causes of the accelerated wear at this location cannot be determined with accuracy until the component is cut-out from the system and examined.

The inspectors determined the licensee's overall response to the issue was commensurate with the safety significance, was timely, and included appropriate corrective actions that were focused to correct the problem. However, the inspectors' review of the root cause report identified the licensee did not thoroughly consider the potential for a fabrication induced geometric discontinuity contributing to the localized FAC at the expander. Further, the inspectors requested a copy of the radiography films referenced in the fabrication records and discovered that the licensee had lost the films in 2012 when a box containing the films for these welds was removed from the storage vault and not returned. The licensee previously identified the loss of these quality assurance records in 2019 and captured the issue in their corrective action program as condition reports CR-2020-03141 and CR-2020-09118. The inspectors determined this issue was of minor safety significance using IMC 0612, Appendix B, more-than-minor screening questions, because there was quality assurance documentation (other than the radiography film) showing satisfactory completion of the pre-service welding. This failure to retain quality assurance records constitutes a minor violation that is not subject to enforcement action in accordance with the NRC's Enforcement Policy.

EXIT MEETINGS AND DEBRIEFS

The inspectors verified no proprietary information was retained or documented in this report.

- On August 25, 2021, the inspectors presented the Unit 2 bottom head drain inspection results to Ms. Melisa Krick, Regulatory Assurance Manager, and other members of the licensee staff.
- On August 26, 2021, the inspectors presented the radwaste and shipping inspection results to Mr. Kevin Cimorelli, Site Vice President, and other members of the licensee staff.
- On September 23, 2021, the inspectors presented the evaluations of changes, tests, and experiments inspection results to Mr. Kevin Cimorelli, Site Vice President, and other members of the licensee staff.
- On October 28, 2021, the inspectors presented the integrated inspection results to Mr. Brad Berryman, Chief Nuclear Officer, and other members of the licensee staff.

DOCUMENTS REVIEWED

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
71111.01	Procedures	NDAP-00-0030	Severe Weather/Natural Disaster Preparation	Revision 17
71111.04	Drawings	E105952	Unit 2 P&ID Core Spray	Revision 31
71111.05	Corrective Action Documents Resulting from Inspection		CR-2021-12398, CR-2021-12404	
	Fire Plans	FP-013-169	Equipment and Battery Rooms Unit 1 East Side El. 771'	Revision 4
		FP-013-170	Equipment and Battery Rooms Unit 2 West Side El. 771'	Revision 5
		FP-013-171	Equipment and Battery Rooms Unit 2 East Side El. 771'	Revision 4
		FP-013-189	Diesel Generator Bay 'A', FZ 0-41A	Revision 4
		FP-113-112	Unit 1, Equipment Area, FZ 1-3C-N, S, W	Revision 5
		FP-213-239	RCIC Pump Room Elevation 645'	Revision 7
		FP-213-240	RHR Pump Room Elevation 645'	Revision 5
		FP-213-242	Sump Pump Room Elevation 645'	Revision 6
	Miscellaneous	U1CRFWP	Fire-Unit 1 "C" Reactor Feedwater Pump with Extension into the Oil Sump Area, 656'	09/14/2021
71111.11Q	Procedures	JIT-OP-2106	Plant Startup JITT	Revision 0
71111.12	Corrective Action Documents		CR-2020-06582, CR-2020-10072, CR-2021-09255, CR-2021-09661, CR-2021-09818	
	Corrective Action Documents Resulting from Inspection		CR-2021-12291	
	Work Orders		ERPM 2190209, ERPM 2190210, ERPM 2448047, RTPM 2203693	
71111.13	Miscellaneous		Risk Mitigating Actions for Yellow PRA Risk	08/24/2021
	Procedures	SO-016-002	Quarterly Common RHRSW/ESW (ESW Spray Pond Valve)	Revision 23
		SO-054-004	Unit 1, Quarterly ESW/TBCCW and ESW/RBCCW Isolation Valve Exercising	Revision 25
	Work Orders		PCWO 2457034-0	

71111.15	Corrective Action Documents		CR-2021-09456, CR-2021-09559, CR-2021-12739, CR-2021-13814	
71111.15	Corrective Action Documents Resulting from Inspection		AR-EWR-2021-09552, CR-2021-09529, CR-2021-11076, CR-2021-11107	
71111.15	Drawings		C-1761 Sheet 6, M-2111 Sheet 1	
71111.15	Operability Evaluations		ACT-01-CR-2021-12561; ACT-01-CR-2021-12566, ACT-01-CR-2021-12659	
71111.17T	Corrective Action Documents Resulting from Inspection		CR-2021-13922	
71111.17T	Procedures	NDAP-QA-0726	10 CFR 50.59 and 10 CFR 72.48 Implementation	Revision 24
71111.19	Procedures	SO-249-B02	Quarterly RHR System Flow Verification DIV II	Revision 29
71111.19	Work Orders		ERPM 2300392, ERPM 2349219, PCSO 2400204, PCWO 2263093, PCWO 2274272, PCWO 2359055, PCWO 2402167-24, PCWO 2439539-1, PCWO 2439539-2, PCWO 2439539-3, PCWO 2452510-0, RACT 2146835, RTSV 2274269	
71111.22	Procedures	SO-054-A08	Comprehensive ESW Flow Verification Loop A	Revision 14
71153	Corrective Action Documents		CR-2021-02351	02/14/2021

<u>November 4, 2021</u> – Letter from Glenn T. Dentel, Chief Engineering Branch 2 Division of Operating Reactor Safety to Brad Berryman Senior Vice President and Chief Nuclear Officer Susquehanna Nuclear, LLC with subject of SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 – TEMPORARY INSTRUCTION 2515/194 INSPECTION REPORT 05000387/2021011 AND 05000388/2021011

On October 27, 2021, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Susquehanna Steam Electric Station, Units 1 and 2 and discussed the results of this inspection with Mr. Kevin Cimorelli, 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: 05000387 and 05000388

License numbers: NPF-14 and NPF-22

Report numbers: 05000387/2021011 and 05000388/2021011

Enterprise Identifier: I-2021-011-0020

Licensee: Susquehanna Nuclear, LLC

Facility: Susquehanna Steam Electric Station, Units 1 and 2

Location: 769 Salem Blvd., Berwick, PA

Inspection dates: October 25, 2021 to October 27, 2021

Inspectors: A. Patel, Senior Reactor Inspector

F. Arner, Senior Reactor Analyst

Approved by: Glenn T. Dentel, Chief Engineering Branch 2 Division of Operating Reactor Safety

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting a Temporary Instruction 2515/194 Inspection at Susquehanna Steam Electric Station, Units 1 and 2, 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), 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. However, all the inspection activities were performed onsite. The inspections documented below met the objectives and requirements for completion of the IP.

OTHER ACTIVITIES – TEMPORARY INSTRUCTIONS, INFREQUENT AND ABNORMAL

2515/194 - 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)

The inspectors reviewed the licensee's implementation of the "Nuclear Energy Institute Voluntary Industry Initiative," (ADAMS Accession No. ML19163A176) dated June 6, 2019. This included reviewing how the licensee updated their licensing basis to reflect the need to protect against open phase conditions.

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) (1 Sample)

(1) Susquehanna Nuclear selected the open phase detection system designed and manufactured by PSSTech as the design vendor for the open phase condition system at Susquehanna Steam Electric Station. The open phase protection system is designed to protect the offsite power sources from a loss of phase condition. Startup Transformers T10 and T20 provide two 230 kV independent offsite power sources to the Susquehanna station from the bulk power system. Startup Transformers T10 and T20 provide offsite power for the Engineered Safeguard Auxiliary buses through 13.8kV Start Up Bus 10 (OA103) and Bus 20 (OA104). Four PSStech open phase detection systems have been installed on the high side of Startup Transformers T10 and T20. The relays are wired to provide annunciation if a loss of phase condition is detected. Alarms from the open phase detection systems, including open phase condition alarms and panel trouble alarms, are annunciated in the control room at the "Start Up XFMR Trouble" alarm windows via the transformer's control panel. Additionally, the status of individual open phase detection system alarm conditions (including open phase detected, channel injection abnormal, and Injection source failure) are available on the plant process computer system.

In lieu of automatic open phase protective actions, Susquehanna Nuclear implemented an alarm only strategy which relies on proper operator actions to diagnose and respond to an open phase condition. At the end of this inspection the PSSTech relays were monitoring the associated power sources and would provide main control room annunciation if a loss of one or two phase conditions was detected or if a relay was non-functional.

INSPECTION RESULTS

Observation: Temporary Instruction 2515/194 - Section 03.01(a) and (c) Results	2515/194

Based on discussions with Susquehanna Nuclear staff, review of design and testing documentation, and walkdowns of installed equipment, the inspectors had reasonable assurance that Susquehanna Nuclear is appropriately implementing the voluntary industry initiative at Susquehanna Steam Electric Station, Units 1 and 2. The inspectors verified the following criteria:

Detection, Alarms and General Criteria

1. [03.01(a)(1)] Open phase conditions are detected and alarmed in the control room.
2. [03.01(a)(2)] Open phase condition detection circuits are sensitive enough to identify an OPC for all credited transformer loading conditions (high and low loading). In addition, enhanced monitoring criteria have been proceduralized when automatic detection is out of service.
3. [03.01(a)(3)] The open phase condition design and protective schemes minimize misoperation or spurious action in the range of voltage unbalance normally expected in the transmission system that could cause separation from an operable off-site power source. Additionally, Susquehanna Nuclear has demonstrated that the actuation circuit design does not result in lower overall plant operation reliability.
4. [03.01(a)(4)] No Class-1E circuits were replaced with non-Class-1E circuits in this design.
5. [03.01(a)(5)] The Updated Final Safety Analysis Report was updated to discuss the design features and analyses related to the effects of any open phase condition design vulnerability.
6. [03.01(a)(6)] The open phase condition detection and alarm components are maintained in accordance with Susquehanna Nuclear's procedures or maintenance program, and periodic tests, calibrations setpoint verifications or inspections (as applicable) have been established.

Use of Risk-Informed Evaluation Method

1. [03.01(c)(1)] The plant configuration matched the changes made to the probabilistic risk assessment model to address an open phase condition, and the logic of the probabilistic risk assessment model changes is sound.
2. [03.01(c)(2)] The procedures which validate that the open phase condition alarm would identify the proper indication to validate the open phase conditions at all possible locations
3. [03.01(c)(3)] Observations associated with procedure(s) and operator actions required to respond to an open phase condition alarm and potential equipment trip match the Human Reliability Analysis.
4. [03.01(c)(4)] Assumptions listed in the NEI 19-02 Appendix A evaluation and the sensitivity analyses listed in Section 5 of the evaluation.

5. [03.01(c)(5)] Assumptions, procedures, operator actions and Susquehanna Nuclear's analyses specified above are consistent with the plant-specific design and licensing basis, including:

(a) Initiating events considered in the analysis

(b) Boundary conditions specified in Attachment 1 of the NEI Voluntary Industry Initiative, Revision 3

(c) Operating procedures for steps taken to recover equipment assumed tripped/locked out or damaged due to the open phase conditions (or use of alternate equipment)

(d) Where recovery was assumed in the probabilistic risk assessment analysis for tripped electric equipment, restoration of the equipment was based on analyses that demonstrate that automatic isolation trips did not result in equipment damage

There were no findings or exceptions identified during the inspection.

EXIT MEETINGS AND DEBRIEFS

The inspectors verified no proprietary information was retained or documented in this report.

- On October 27, 2021, the inspectors presented the Temporary Instruction 2515/194 Inspection results to Mr. Kevin Cimorelli, Site Vice President, and other members of the licensee staff.

DOCUMENTS REVIEWED

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
2515/194	Corrective Action Documents Resulting from Inspection	CR-2021-15357	Potential Knowledge Gap related to OPC	
		CR-2021-15358	Enhancement/Clarification for OPC Procedures	
		CR-2021-15359	AR-015-001 Simulator Alarm response was 3 revisions behind the current revision in NIMS	
	Engineering Evaluations	EC-RISK-0029	Open Phase Condition (OPC) Evaluation	Revision 1
	Procedures	AR-015-001	13.8/4kV Switchgear Distribution and Diesel Generators A, B, & C 0C653	Revision 59
		LA-0X103-001	0X103 Startup Transformer Local Alarm Responses	Revision 10
		LA-0X104-001	0X104 Startup Transformer Local Indicating Light Alarms	Revision 11

November 8, 2021 – Letter from Mel Gray, Chief Engineering Branch 1 Division of Operating Reactor Safety to Brad Berryman Senior Vice President and Chief Nuclear Officer Susquehanna Nuclear, LLC with subject of SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1 – INFORMATION REQUEST TO SUPPORT POST-APPROVAL SITE INSPECTION FOR LICENSE RENEWAL; INSPECTION REPORT 05000387/2022011

The purpose of this letter is to notify you that the U.S. Nuclear Regulatory Commission (NRC) Region I staff will conduct a license renewal post-approval inspection at your Susquehanna Steam Electric Station, Unit 1. Niklas Floyd, 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 71003, "Post-Approval Site Inspection for License Renewal," dated July 1, 2016 (ADAMS Accession No. ML16013A260). This inspection is described as a Phase 2 license renewal inspection in the referenced inspection procedure and is conducted three months to one year prior to the plant entering its period of extended operation.

The inspection will assess the adequacy of the planned and/or completed activities and programs described in the regulatory commitments and license conditions added as part of your renewed license. This inspection will also evaluate the need for additional follow-up inspections (Phase 3) under Inspection Procedure 71003 or as part of the Reactor Oversight Program. Finally, an inspector will perform a license renewal inspection

(Phase 1) in April 2022 to observe implementation of aging management programs and activities that are only accessible during a scheduled plant outage.

This onsite Phase 2 inspection is scheduled for the weeks of January 11 and January 25, 2022.

In order to minimize the 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 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 and in-office preparation activities. During the in-office preparation activities, the team will identify as much as possible the information and activities to be reviewed during the inspection. This information should be provided to the lead inspector by December 27, 2021.

• The second group includes the additional information required for the team to review the selected activities. This information should be provided to the team by January 4, 2021 or made available upon arrival onsite January 11, 2022.

If there are any questions about the inspection or the information requested in the enclosure, please do not hesitate to contact the lead inspector at 610-337-5282 or via e-mail at Niklas.Floyd@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."

INFORMATION REQUEST
POST-APPROVAL SITE INSPECTION FOR LICENSE RENEWAL – PHASE II

Inspection Report:	05000387/2022011
Onsite Inspection Dates:	January 10, through January 14, 2022; and January 24, through January 28, 2022
Inspection Procedure:	Inspection Procedure 71003, Post-Approval Site Inspection for License Renewal, dated July 1, 2016
Lead Inspector:	Niklas Floyd, Senior Reactor Inspector 610-337-5282 Niklas.Floyd@nrc.gov

I. Information Requested Prior to the On-site Inspection Week (General Info)

The following information is requested by December 27, 2021, or sooner, to facilitate inspection preparation. This section of the request identifies general information that will help the inspectors to understand the applicable administrative procedures and overall completion status of license renewal activities. As such, the information requested herein is not aimed to a particular regulatory commitment, license condition, aging management program (AMP), or time-limited aging analysis (TLAA).

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, etc.). The files should contain descriptive names, 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) Sections of the quality assurance program manual applicable to license renewal activities.

(2) Copies of the following procedures:

- Correction Action Program
- Operating Experience
- Regulatory Commitment Change Process

(3) Updated Final Safety Analysis Report (UFSAR) supplement for license renewal submitted with the license renewal application (LRA), including the latest version of the UFSAR supplement submitted to the NRC per 10 CFR 50.71.

(4) Post-approval evaluations or assessments performed to verify that in-scope structures, systems, or components (SSCs) were not excluded from the LRA and appropriate aging management has been considered in accordance with 10 CFR 54.37(b).

(5) List of plant modifications with a brief description of the modification scope that were approved and/or implemented from the date the LRA was submitted to the date the renewed license was issued.

(6) Self-assessments performed after the renewed license was issued associated with the implementation of license renewal commitments, including AMPs and TLAAs.

(7) "Gap analyses" performed to assess the differences between the current AMPs and the recommendations in the latest revision of NUREG-1801, "Generic Aging Lessons Learned (GALL)," and subsequent NRC Interim Staff Guidance.

(8) Site-specific responses and/or evaluations for NRC generic communications associated with license renewal that were issued after the renewed license was granted.

(9) List of your license renewal commitments with a description of the overall completion status of action items associated with each commitment, including AMPs and TLAAs.

(10) Description of administrative controls that will be used to ensure that all activities due prior to (and during) the period of extended operation (PEO) will be satisfactorily completed as described in the license renewal documents.

(11) Description of plans or processes that would be used during the PEO to continuously incorporate operating experience into license renewal activities.

(12) Internal/external self-assessment and associated corrective action documents generated in preparation for this inspection.

(13) License renewal organizational chart and points of contact for this inspection, including the appropriate regulatory assurance, engineering, and license renewal staff.

II. Information Requested to Be Available by January 4, 2022 (Specific Requests for each License Renewal Commitment, including associated AMP and TLAA)

The following information should be separated for each regulatory commitment, especially if provided electronically (e.g., a folder for each commitment named after the commitment that includes the information requested below). The commitment should include the associated AMP and TLAAs. This section is focused on documentation demonstrating completion of each specific license renewal commitment.

(1) Program basis documents and administrative procedures describing key program attributes such as program objectives, scope, detection and monitoring methods, administrative controls, acceptance criteria, corrective actions, and scope expansion requirements.

(2) Implementing procedures for program activities (e.g., visual examination procedures, ultrasonic examination procedures, maintenance procedures, system walkdowns, etc).

(3) Updated license renewal scoping drawings showing the SSCs within the scope of the AMP.

(4) Applicable sections of the LRA and the NRC's safety evaluation report.

INFORMATION REQUEST
POST-APPROVAL SITE INSPECTION FOR LICENSE RENEWAL – PHASE II

(5) Applicable Requests for Additional Information issued by NRC's technical reviewers during the review of the LRA and corresponding responses.

(6) Copy of or ready access to the key industry standards that will be followed during the PEO for the implementation of the program (e.g., American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Electric Power Research Institute Guidelines, Nuclear Energy Institute Guidelines, etc).

(7) Description of changes made to the license renewal commitment, including the associated technical and regulatory evaluations supporting the change.

(8) Description of activities completed to meet license conditions, license renewal commitments, and the UFSAR. For example, this should include completed Work Orders with associated inspection reports describing the work done and the results of the inspection.

(9) Description of instances where examination scope expansion was required based on the initial examination of an SSC for license renewal. Please describe the logic to select the scope expansion, and any resulting changes to the AMP.

(10) Description of instances where examination scope changes were required based on limitations encountered during the initial examination of an SSC for license renewal. Please describe the actions taken, and any resulting changes to the AMP.

(11) Description of pending activities due prior to the PEO, which are necessary to meet license conditions, license renewal commitments, and the activities described in the UFSAR. For example, this may include a list of pending Work Orders with a brief description of the work to be performed.

(12) Copy of any correspondence between the licensee and the NRC, after the license was issued, associated with regulatory issues affecting the license renewal commitment. For example, this may include notifications of commitment changes, or license amendment requests, affecting a license condition or a license renewal commitment.

(13) List of Corrective Action Program documents (e.g., Condition Reports) associated with the implementation of the AMP. This includes unacceptable aging effects identified during the implementation of license renewal activities and programmatic deficiencies requiring resolution.

(14) Copy of evaluations performed for applicable external/internal Operating Experience issues associated with license renewal. This request is limited to Operating Experience items issued after the renewed operating license was granted.

INFORMATION REQUEST
POST-APPROVAL SITE INSPECTION FOR LICENSE RENEWAL – PHASE II

III. Information Requested to be Provided Throughout the Inspection

(1) Copies of any corrective action documents generated as a result of the team's questions or queries during this inspection.

(2) Copies of the list of questions submitted by the team members and the status/resolution of the information requested (please provide daily during the inspection to each team member).

December 13, 2021 – Letter from Mel Gray, Chief Engineering Branch 1 Division of Operating Reactor Safety to Brad Berryman Senior Vice President and Chief Nuclear Officer Susquehanna Nuclear, LLC with subject of SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 – DESIGN BASIS ASSURANCE INSPECTION (PROGRAMS) INSPECTION REPORT 05000387/2021010 AND 05000388/2021010

On November 19, 2021, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Susquehanna Steam Electric Station, Units 1 and 2 and discussed the results of this inspection with Mr. Kevin Cimorelli, 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:	05000387 and 05000388
License Numbers:	NPF-14 and NPF-22
Report Numbers:	05000387/2021010 and 05000388/2021010
Enterprise Identifier:	I-2021-010-0027
Licensee:	Susquehanna Nuclear, LLC
Facility:	Susquehanna Steam Electric Station, Units 1 and 2
Location:	Berwick, PA
Inspection Dates:	November 1, 2021 to November 19, 2021
Inspectors:	P. Cataldo, Senior Reactor Inspector M. Farnan, Mechanical Engineer D. Kern, Senior Reactor Inspector
Approved By:	Mel Gray, Chief Engineering Branch 1 Division of Operating Reactor Safety

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting a design basis assurance inspection (programs) inspection at Susquehanna Steam Electric Station, Units 1 and 2, 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.

REACTOR SAFETY

71111.21N.02 - Design-Basis Capability of Power-Operated Valves Under 10 CFR 50.55a Requirements

Power-Operated Valve Review (IP Section 03) (11 Samples)

The inspectors:

a. Determined whether the sampled power-operated valves (POVs) are being tested and maintained in accordance with NRC regulations along with the licensee's commitments and/or licensing bases.

b. Determined whether the sampled POVs are capable of performing their design-basis functions.

c. Determined whether testing of the sampled POVs is adequate to demonstrate the capability of the POVs to perform their safety functions under design-basis conditions.

d. Evaluated maintenance activities including a walkdown of the sampled POVs (if accessible).

(1) HV01222B, RHRSW - Ultimate Heat Sink Spray Bypass Isolation Valve
(2) HV112F073A, RHRSW Loop 'A' Crosstie Valve
(3) HV151F028A, RHR Loop 'A' Suppression Pool Spray Test Shutoff Valve
(4) HV155F003, HPCS - HPCI Steam Supply Outboard Isolation Valve
(5) HV250F046, RCIC Lube Oil Cooler Water Supply Valve
(6) HV255F001, HPCI Turbine Steam Supply Valve
(7) HV255F006, HPCS - HPCI Injection Valve
(8) HV141F022A, 'A' Inboard Main Steam Isolation Valve
(9) HV25723, Containment Atmosphere Control - Drywell Air Purge Isolation Valve
(10) SV12654A, Containment Instrument Gas - to Main Steam PSV1F013 GJM
(11) XV247F010A/B, Control Rod Drive Scram Discharge Volume Vent Valves

INSPECTION RESULTS

No findings were identified.

EXIT MEETINGS AND DEBRIEFS

The inspectors verified no proprietary information was retained or documented in this report.

• On November 19, 2021, the inspectors presented the design basis assurance inspection (programs) inspection results to Mr. Kevin Cimorelli, Site Vice President, and other members of the licensee staff.

DOCUMENTS REVIEWED

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
71111.21N.02	Calculations	EC-006-0506	HV151F028A Thermal Overload Calculation	Revision 3
		EC-049-1038	HV151F028A Weak Link Seismic Calculation	Revision 3
		EC-088-0505	Unit 1 and Unit 2 Class 1E 250 VDC System Voltage Drop Calculation EE5	Revision 8
		EC-VALV-0569	V151F028A System Design Basis Calculation	Revision 22
		EC-VALV-1073	HV151F028A Component Design Basis	Revision 49
		EC-VALV-1109	Degradation Assessment Supporting Annual MOV Performance Assessment Report	Revision 13
		MDS-06	Verification of Motor-Operated Valve Functionality	Revision 18
		MDS-08	Periodic Performance Assessment for SSES Motor-Operated Valves	Revision 13
	Corrective Action Documents	CR 2021-02179		
		CR 2021-10413		
	Corrective Action Documents Resulting from Inspection	CR 2021-15644		
		CR 2021-15645		
		CR 2021-15646		
		CR 2021-15702		
		CR 2021-15705		
		CR 2021-15706		
		CR 2021-15709		
		CR 2021-15711		
		CR 2021-15721		
		CR 2021-15821		
		CR 2021-16042		
		CR 2021-16387		
		CR 2021-16439		
		CR 2021-16461		
	Drawings	M-112	Unit 1 RHR Service Water System	Revision 55
		M-2147	Unit 2 Control Rod Drive Part B	Revision 38
		M-2150	Unit 2 RCIC Turbine Pump	Revision 32
	Engineering	EQAR-084	Limitorque Actuator Environmental Qualification	Revision 23

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
	Evaluations		Assessment Report	
	Miscellaneous	ASME OMB Code-2006	2006 Addenda to ASME OM Code-2004 for Operation and Maintenance of Nuclear Power Plants	08/31/2006
		SUS-ISTPLN-100.0	Unit 1 Inservice Testing Program Plan	Revision 8
		SUS-ISTPLN-200.0	Unit 2 Inservice Testing Program Plan	Revision 11
	Procedures	NDAP-QA-0017	Motor Operated Valve Program	Revision 18
		NDAP-QA-1170	Air Operated Valve Program	Revision 5
	Work Orders	ERPM 1975504		
		PCWO 2419208		
		RTPM 2014246		