UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE RD, STE 102
KING OF PRUSSIA, PENNSYLVANIA 19406-1415

October 27, 2022

EA-22-071

David P. Rhoades
Senior Vice President
Constellation Energy Generation, LLC
President and Chief Nuclear Officer (CNO)
Constellation Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 – INTEGRATED INSPECTION REPORT 05000277/2022003 AND 05000278/2022003 AND PRELIMINARY WHITE FINDING AND APPARENT VIOLATION

Dear David Rhoades:

On September 30, 2022, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Peach Bottom Atomic Power Station, Units 2 and 3. On October 14, 2022, the NRC inspectors discussed the results of this inspection with Dave Henry, Site Vice President, and other members of your staff. The results of this inspection are documented in the enclosed report.

The enclosed inspection report discusses a finding with an associated apparent violation that the NRC has preliminarily determined to be White, a finding of low to moderate safety significance. The finding is associated with an apparent violation of Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," because Constellation failed to accomplish an activity affecting quality using a procedure appropriate to the circumstances. As a result of using the procedure that was not appropriate to the circumstances, an operator took an action that caused a Unit 2 reactor scram, primary containment isolation system Group I isolation, safety-relief valve actuation, and loss of the normal heat sink which required emergency core cooling systems to maintain level and pressure. The finding was assessed based on the best available information, using Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power.” The finding and basis for the NRC’s preliminary significance determination is described in the enclosed report.

We are considering escalated enforcement for the apparent violation consistent with our Enforcement Policy, which can be found at http://www.nrc.gov/about-nrc/regulatory/enforcement/enforce-pol.html. Because we have not made a final determination, no notice of violation is being issued at this time. Please be aware that further NRC review may prompt us to modify the number and characterization of the apparent violation.

We intend to issue our final significance determination and enforcement decision, in writing, within 90 days from the date of this letter. The NRC's significance determination process (SDP) is designed to encourage an open dialogue between your staff and the NRC; however, neither the dialogue nor the written information you provide should affect the timeliness of our final determination.

Before we make a final decision, you may choose to communicate your position on the facts and assumptions used to arrive at the finding and assess its significance by either (1) attending and presenting at a regulatory conference or (2) submitting your position in writing. The focus of a regulatory conference is to discuss the significance of the finding and not necessarily the root cause(s) or corrective actions associated with the finding.

If you choose to respond in writing, please mark the response "Response to Preliminary White Finding in Inspection Report 05000277/2022003 and 05000278/2022003; EA-22-071," and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, Region I, and a copy to the NRC Senior Resident Inspector at Peach Bottom Atomic Power Station, Units 2 and 3 within 40 days of the date of this letter.

If you request a regulatory conference, it should be held within 40 days of your receipt of this letter. Please provide information you would like us to consider or discuss with you at least 10 days prior to any scheduled conference. If you choose to attend a regulatory conference, it will be open for public observation.

If you choose not to request a regulatory conference or to submit a written response, you will not be allowed to appeal the NRC's final significance determination.

Please contact Jon Greives at 610-337-5337, or in writing, within 10 days from the issue date of this letter to notify the NRC of your intentions. If we have not heard from you within 10 days, we will continue with our significance determination and enforcement decision.

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 violations or the significance or severity of the violations documented in this inspection report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement; and the NRC Resident Inspector at Peach Bottom Atomic Power Station, Units 2 and 3.

If you disagree with a cross-cutting aspect assignment 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 Peach Bottom Atomic Power Station, Units 2 and 3.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with 10 CFR 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Sincerely,

Daniel S. Collins

Digitally signed by Daniel S. Collins
Date: 2022.10.27 08:19:24 -04'00'

Daniel S. Collins, Director
Division of Operating Reactor Safety

Docket Nos. 05000277 and 05000278
License Nos. DPR-44 and DPR-56

Enclosure:
Inspection Report 05000277/2022003 and 05000278/2022003 w/Attachment: Detailed Risk Evaluation

cc w/ encl: Distribution via LISTSERV

SUBJECT: PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 – INTEGRATED INSPECTION REPORT 05000277/2022003 AND 05000278/2022003 AND PRELIMINARY WHITE FINDING AND APPARENT VIOLATION DATED OCTOBER 27, 2022

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U.S. NUCLEAR REGULATORY COMMISSION
Inspection Report

Docket Numbers:	05000277 and 05000278
License Numbers:	DPR-44 and DPR-56
Report Numbers:	05000277/2022003 and 05000278/2022003
Enterprise Identifier:	I-2022-003-0039
Licensee:	Constellation Energy Generation, LLC
Facility:	Peach Bottom Atomic Power Station, Units 2 and 3
Location:	Delta, PA 17314
Inspection Dates:	July 1, 2022 to September 30, 2022
Inspectors:	P. Boguszewski, Senior Resident Inspector E. Brady, Project Engineer C. Dukehart, Resident Inspector N. Eckhoff, Health Physicist B. Edwards, Health Physicist N. Floyd, Senior Reactor Inspector E. Miller, Reactor Inspector S. Rutenkroger, Senior Resident Inspector
Approved By:	Daniel S. Collins, Director Division of Operating Reactor Safety

Enclosure

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting an integrated inspection at Peach Bottom Atomic Power Station, Units 2 and 3, in accordance with the Reactor Oversight Process. The Reactor Oversight Process is the NRC's program for overseeing the safe operation of commercial nuclear power reactors. Refer to https://www.nrc.gov/reactors/operating/oversight.html for more information.

List of Findings and Violations

High-Pressure Service Water Discharge Check Valve Closure Testing Inadequate			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000277,05000278/2022003-01 Open/Closed	[P.2] - Evaluation	71111.15
The inspectors identified a Green non-cited violation (NCV) of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.55(a)(f)(4) because Constellation did not perform adequate in-service testing on check valves within the scope of the American Society of Mechanical Engineers (ASME) OM Code. Specifically, high-pressure service water (HPSW) discharge check valve testing did not adequately verify valve closure in accordance with in-service testing program requirements. As a result, Constellation did not adequately correct long-standing degraded conditions with check valves sticking open.			

Failure to Determine and Correct Adverse Condition Leads to Reactor Scram			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Initiating Events	Green FIN 05000277/2022003-02 Open/Closed	[H.13] - Consistent Process	71153
A self-revealed Green finding was identified when Constellation failed to perform an adequate functional review of an action request. Specifically, Constellation failed to identify jet compressor suction pressure control valve CV-2-08-8417A was affected by a nearby steam leak when doing a functional review of the components in the area.			

Loss of Reactor Protection System Power and Unit Scram Due to Operator Error			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Initiating Events	Preliminary White AV 05000277/2022003-03 Open EA-22-071	[H.1] - Resources	71153
The inspectors identified a self-revealing preliminary White finding and an associated apparent violation (AV) of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," because Constellation failed to accomplish an activity affecting quality using a procedure appropriate to the circumstances. Specifically, Constellation failed to implement the pre-planned "Partial" of procedure SO 60F.1.A-2/3, "Reactor Protection System MG Set and Power Distribution system Startup from Dead Bus Condition," when responding to the trip of the Unit 2 'A' reactor protection system (RPS) motor generator (MG) set while Unit 2 'B' RPS was supplied via the alternate feed and instead used the "clean" copy of procedure SO			

60F.1.A-2, which was not appropriate to the circumstances. As a result, an operator performed a procedure step which opened the output breakers associated with the alternate RPS feed causing a Unit 2 reactor scram, primary containment isolation system (PCIS) Group I isolation, safety-relief valve (SRV) actuation, and loss of the normal heat sink which required emergency core cooling systems (ECCS) to maintain level and pressure.

Additional Tracking Items

Type	Issue Number	Title	Report Section	Status
LER	05000277/2021-003-00	Unit 2 Manual Reactor Scram Due to Degrading Condenser Vacuum	71153	Closed
LER	05000277/2022-001-00	Unit 2 Automatic Reactor Scram Due to Loss of Power to Both RPS Buses	71153	Closed

PLANT STATUS

Unit 2 began the inspection period at rated thermal power (RTP). Operators reduced power as needed on July 1, July 15, July 23, and August 7, 2022, for control rod pattern adjustments, and returned to RTP within a day. Operators reduced power as needed on August 1, August 10, August 15, August 19, August 22, August 24, August 26, August 28, September 1, September 4, and September 17, 2022, to cycle main condenser waterbox inlet valves, and returned to RTP within a day. On August 5, 2022, operators initially reduced power to approximately 55 percent for main condenser waterbox cleaning and then reduced power to approximately 33 percent due to degrading main condenser vacuum. Operators returned the unit to RTP the following day. The unit remained at or near RTP for the remainder of the inspection period.

Unit 3 began the inspection period at RTP. On July 30 and September 2, 2022, operators reduced power to approximately 81 percent for main condenser waterbox inlet valve cycling and returned the unit to RTP the following day. On September 9, 2022, operators reduced power to approximately 66 percent for a control rod sequence exchange, main turbine valve maintenance and testing, and main condenser waterbox inlet valve cycling and returned the unit to RTP the following day. The unit remained at or near RTP for the remainder of the inspection period.

INSPECTION SCOPES

Inspections were conducted using the appropriate portions of the inspection procedures (IPs) in effect at the beginning of the inspection unless otherwise noted. Currently approved IPs with their attached revision histories are located on the public website at http://www.nrc.gov/reading-rm/doc-collections/insp-manual/inspection-procedure/index.html. Samples were declared complete when the IP requirements most appropriate to the inspection activity were met consistent with Inspection Manual Chapter (IMC) 2515, “Light-Water Reactor Inspection Program - Operations Phase.” The inspectors performed activities described in IMC 2515, Appendix D, “Plant Status,” observed risk significant activities, and completed on-site portions of IPs. 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

71111.01 - Adverse Weather Protection

Seasonal Extreme Weather Sample (IP Section 03.01) (1 Sample)

The inspectors evaluated readiness for seasonal extreme weather conditions prior to the onset of extreme temperatures for the following system:

(1) Offsite and onsite alternating current primary and alternate power systems, electrical switchgear rooms, cable spreading room, and reactor building closed loop cooling water systems on August 9, 2022

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) Unit common, 'E-4' emergency diesel generator (EDG) during 'E-3' EDG maintenance and testing on July 20, 2022
(2) Unit common, motor driven fire pump during diesel driven fire pump (DDFP) maintenance on August 29 and 30, 2022
(3) Unit 2, 'A' and 'B' loops of the core spray system on September 21, 2022

71111.05 - Fire Protection

Fire Area Walkdown and Inspection Sample (IP Section 03.01) (4 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) Unit common, circulating water pump structure, general area, PF-144, on July 7, 2022
(2) Unit common, turbine building hatch area, PF-81, on July 21, 2022
(3) Unit common, turbine building deck area, PF-80, on July 21, 2022
(4) Unit common, EDG building general area, PF-132, on September 14, 2022

Fire Brigade Drill Performance Sample (IP Section 03.02) (1 Sample)

(1) The inspectors evaluated the onsite fire brigade training and performance during an unannounced fire drill on September 2, 2022

71111.11Q - Licensed Operator Requalification Program and Licensed Operator Performance

Licensed Operator Requalification Training/Examinations (IP Section 03.02) (1 Sample)

(1) The inspectors observed and evaluated licensed operator requalification training in the simulator on September 19, 2022

71111.12 - Maintenance Effectiveness

Maintenance Effectiveness (IP Section 03.01) (3 Samples)

The inspectors evaluated the effectiveness of maintenance to ensure the following structures, systems, and components (SSCs) remain capable of performing their intended function:

(1) Unit 2, HPSW system through September 27, 2022
(2) Unit 3, HPSW system through September 27, 2022
(3) Unit 3, control rod drive system hydraulic control unit directional control valves through September 28, 2022

71111.13 - Maintenance Risk Assessments and Emergent Work Control

Risk Assessment and Management Sample (IP Section 03.01) (6 Samples)

The inspectors evaluated the accuracy and completeness of risk assessments for the following planned and emergent work activities to ensure configuration changes and appropriate work controls were addressed:

(1) Unit common, DDFP tripped during testing on July 12, 2022
(2) Unit 3, reactor core isolation cooling (RCIC) steam line break temperature instrumentation risk-informed completion time implementation on July 25, 2022
(3) Unit 3, ‘B’ RHR testing and circuit breaker swap on July 26, 2022
(4) Unit 2, ‘B’ and ‘D’ HPSW maintenance, ‘D’ residual heat removal (RHR) heat exchanger cleaning and maintenance, and risk informed completion time implementation on August 8, 2022
(5) Unit 3, ‘B’ and ‘D’ HPSW unavailable for cross-tie valve maintenance on August 10, 2022
(6) Unit 3, ‘B’ loop HPSW unavailable for maintenance on September 8, 2022

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) Unit 2, control rod ‘26-03’ temperature indication intermittently displaying ‘OTD’ on July 22, 2022
(2) Unit 2, HPSW flow transmitter and indicator found out of tolerance on July 27, 2022
(3) Unit 3, ‘B’ HPSW pump discharge check valve was open with the ‘B’ HPSW pump shut down and the ‘D’ HPSW pump running on August 12, 2022
(4) Unit 2, ECCS compensated level system backup power ready light extinguished on August 29, 2022
(5) Unit 2, high-pressure coolant injection (HPCI) pump steam emission valve packing leak on September 7, 2022
(6) Unit 2, 'B' standby liquid control pump oil level lower than the required minimum static level on September 26, 2022
(7) Unit 2, 'B' loop core spray full flow test valve suspected leak-by to the torus on September 19, 2022

71111.18 - Plant Modifications

Severe Accident Management Guidelines (SAMG) Update (IP Section 03.03) (1 Sample)

(1) The inspectors verified that the site severe accident management guidelines for Units 2 and 3 were updated following the boiling water reactors owners' group issuance of Revision 4 of the severe accident technical guidelines in June 2018

71111.19 - Post-Maintenance Testing

Post-Maintenance Test Sample (IP Section 03.01) (8 Samples)

The inspectors evaluated the following post-maintenance testing activities to verify system operability and/or functionality:

(1) Unit 3, 'B' core spray loop planned maintenance on July 11, 2022
(2) Unit 3, ‘B’ and ‘D’ HPSW cross-tie valve maintenance on August 12, 2022
(3) Unit 3, ‘B’ HPSW pump discharge check valve maintenance on September 8, 2022
(4) Unit 2, post-maintenance for 125v battery charger 2CD003-1 and troubleshooting of charger cooling fans on September 13, 2022
(5) Unit 2, 'A' standby liquid control explosive (squib) valve electrical connector replacement on September 13, 2022
(6) Unit common, 'C' emergency service water (ESW) isolation valve maintenance on September 21, 2022
(7) Unit 2, reactor protection system (RPS) Rosemount trip units 'PIS-2-05-012D,' 'LIS-2-02-3-101D,' 'PSL-2-02-3-055D,' and 'LIS-2-02-3-099D' replaced on September 27, 2022
(8) Unit common, 'E-4' EDG crankcase vacuum orifice plate replacement on September 29, 2022

71111.22 - Surveillance Testing

The inspectors evaluated the following surveillance testing activities to verify system operability and/or functionality:

Surveillance Tests (other) (IP Section 03.01) (3 Samples)

(1) Unit 2, HPSW pump, valve and flow functional and in-service test on July 7, 2022
(2) Unit 2, 'B' RHR loop pump, valve, flow and unit cooler functional and in-service test on July 12, 2022
(3) Unit 2, HPSW pump, valve, and flow functional and in-service test on September 13, 2022

FLEX Testing (IP Section 03.02) (1 Sample)

(1) Unit common, ‘00G311’ FLEX diesel generator partially loaded run, on September 30, 2022

71114.06 - Drill Evaluation

Drill/Training Evolution Observation (IP Section 03.02) (2 Samples)

The inspectors evaluated:

(1) Limited scope emergency preparedness drill conducted on August 30, 2022
(2) Limited scope emergency preparedness drill conducted on September 29, 2022

RADIATION SAFETY

71124.06 - Radioactive Gaseous and Liquid Effluent Treatment

Walkdowns and Observations (IP Section 03.01) (4 Samples)

The inspectors evaluated the following radioactive effluent systems during walkdowns:

(1) Unit 2 Vent Stack, wide range radiation monitor, and air sampling system
(2) Unit 3 Vent Stack, wide range radiation monitor, and air sampling system
(3) Main Stack, wide range radiation monitor, and air sampling system
(4) Unit 2 and 3 floor drain tank sampling system

Sampling and Analysis (IP Section 03.02) (4 Samples)

Inspectors evaluated the following effluent samples, sampling processes and compensatory samples:

(1) Continuous plant vent sample for Unit 2 low gas radiation monitor, RE-3979, filter and cartridge change-out
(2) Observed liquid effluent sample from the unit 2 and 3 drain floor tank sampling system
(3) Observed isotopic analysis of charcoal filter from Unit 3 vent stack radiation monitor, 3BF370
(4) Continuous plant vent sample for Main Stack wide range gas radiation monitor, 00S853, filter and cartridge change-out

Dose Calculations (IP Section 03.03) (2 Samples)

The inspectors evaluated the following dose calculations:

(1) Dose calculations and permit generation from Unit 2 vent stack samples
(2) Dose calculations and permit generation from Main stack samples

Abnormal Discharges (IP Section 03.04) (2 Samples)

The inspectors evaluated the following abnormal discharges:

(1) Groundwater tritium plume detected in 2021, which flows into plant intake and eventually normal discharge canal
(2) Release of Cs-138 from Turbine building ventilation in March of 2020

OTHER ACTIVITIES – BASELINE

71151 - Performance Indicator Verification

The inspectors verified Constellation's performance indicators submittals listed below for the period October 1, 2021 through September 30, 2022:

MS07: High-Pressure Injection Systems (IP Section 02.06) (2 Samples)

(1) Unit 2 high-pressure injection systems
(2) Unit 3 high-pressure injection systems

MS08: Heat Removal Systems (IP Section 02.07) (2 Samples)

(1) Unit 2 heat removal systems
(2) Unit 3 heat removal systems

MS09: Residual Heat Removal Systems (IP Section 02.08) (2 Samples)

(1) Unit 2 RHR systems
(2) Unit 3 RHR systems

MS10: Cooling Water Support Systems (IP Section 02.09) (2 Samples)

(1) Unit 2 cooling water support systems
(2) Unit 3 cooling water support systems

71152A - Annual Follow-up Problem Identification and Resolution

Annual Follow-up of Selected Issues (Section 03.03) (3 Samples)

The inspectors reviewed the licensee’s implementation of its corrective action program (CAP) related to the following issues:

(1) Unit common, 'E-3' EDG testing did not open the 'E-32' circuit breaker when required on October 20, 2021 (Issue report (IR) 4454298)
(2) Unit 3, elevated vibrations on the HPCI steam supply piping (IR 4175898)
(3) Unit 2, manual reactor scram on November 14, 2021 (IR 04460767)

71153 - Follow Up 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 05000277/2022-001-00 for Peach Bottom Atomic Power Station (PBAPS), Unit 2, Automatic Reactor Scram Due to Loss of Power to Both RPS Buses (ADAMS Accession No. ML22196A020)

    The inspection conclusions associated with this LER are documented in this report under Inspection Results Section, AV 05000277/2022003-03.

(2) LER 05000277/2021-003-00, Manual Reactor Scram Due to Degrading Condenser Vacuum (ADAMS Accession No. ML22012A400)

    The inspection conclusions associated with this LER are documented in this report under Inspection Results Section, FIN 05000277/2022003-01.

INSPECTION RESULTS

High-Pressure Service Water Discharge Check Valve Closure Testing Inadequate			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000277,05000278/2022003-01 Open/Closed	[P.2] - Evaluation	71111.15

The inspectors identified a Green non-cited violation (NCV) of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.55(a)(f)(4) because Constellation did not perform adequate in-service testing on check valves within the scope of the American Society of Mechanical Engineers (ASME) OM Code. Specifically, high-pressure service water (HPSW) discharge check valve testing did not adequately verify valve closure in accordance with in-service testing program requirements. As a result, Constellation did not adequately correct long-standing degraded conditions with check valves sticking open.

Description: The PBAPS HPSW system provides cooling water for RHR system heat exchangers and consists of two independent and redundant subsystems. Each subsystem is made up of a header and two 4500 gpm pumps. Each HPSW pump has a discharge check valve, which opens to permit required pump flow and closes when the pump is off to prevent reverse flow if the second pump in the loop is operating. By design, manual initiation of the HPSW subsystem and associated RHR system provides the required cooling flow to two RHR heat exchangers within each containment cooling/spray subsystem. The discharge check valves have a specified design basis safety function both to open and close.

On August 12, 2022, the inspectors observed post-maintenance testing of the Unit 3 ‘B’ and ‘D’ HPSW pumps following completion of maintenance on the Unit 2 to Unit 3 HPSW cross-tie valves. The testing sequence involved both pumps running at the same time followed by shutdown of the ‘B’ HPSW pump with the ‘D’ HPSW pump continuing to run for about a minute. The inspectors noted that the flow through the ‘B’ RHR heat exchanger was about 3300 gpm and steady with the ‘D’ HPSW pump running alone, which was significantly less than the expected flowrate of about 5000 gpm. The inspectors asked the operators about the observed indications, and the equipment operators reported that the ‘B’ HPSW pump discharge check valve ‘502B’ was observed to be about full open following the ‘B’ HPSW pump shutdown with the ‘D’ HPSW pump continuing to run. In addition, an open equipment deficiency tag was present on ‘502B’ due to the check valve first sticking open in 2016. A condition report was not initiated for the issue until further questioning by the inspectors the following week, at which time the senior reactor operators declared the check valve operable.

The inspectors reviewed the HPSW operating, shutdown, and testing procedures, including quarterly in-service testing, interviewed equipment operators regarding check valve performance, and reviewed past condition reports on check valves. The inspectors identified that the procedure testing sequence for the discharge check valves did not ensure closing of the check valves was observed per in-service testing requirements. The surveillance test for each unit includes all four HPSW pumps and allows performing the test in any sequence. The closing test for each check valve was performed following the start of the opposite pump. However, the test sequence did not ensure a prior opening of the check valve to test the closing function. In particular, the normal shutdown procedure for the HPSW pumps required verification of closure of the check valve. In accordance with this step, equipment operators described manually closing stuck open check valves. As a result of the combination of the surveillance test sequence, the pump shutdown procedure verification, and equipment operators manually closing valves, the check valves' in-service monitored safety function to close was effectively pre-conditioned and/or not tested.

In addition, the inspectors identified previous condition reports in which a HPSW check valve was identified to be stuck open in the surveillance test, the operator manually closed the check valve, and the condition was not identified as a failure of the in-service testing (IR 2693528 on July 17, 2016; IR 4204464 on December 19, 2018; IR 4218337 on February 8, 2019; and IR 4466868 on December 15, 2021). Finally, the inspectors identified that ER-PB-321-1000, "Inservice Testing (IST) Program Plan, 5th Ten Year Interval," the IST program plan for the PBAPS 5th ten-year interval, required such valves to be tested in the closed direction and that check valves which fail to exhibit the required change of disk position per this testing shall be considered inoperable.

The 2012 ASME OM Code Edition, the code edition of record for the interval, ISTC-3522, "Category C Check Valves," requires each Category C check valve to be exercised or examined in a manner that verifies obturator travel by using the methods in ISTC-5221. The HPSW discharge check valves are Category C, and the obturator is the valve closure member which stops fluid flow when pressed against the valve seat (e.g., the disk of the check valve). ISTC-5221, "Valve Obturator Movement," requires necessary valve obturator movement during exercise testing to be demonstrated by performing both an open and a close test and the exercise test to verify that on cessation or reversal of flow the obturator has traveled to the seat (for check valves that have a safety function in the close direction). ISTC-5224, "Corrective Action," requires that if a check valve fails to exhibit the required change of obturator position, it shall be declared inoperable and that a retest showing acceptable performance shall be run following any required corrective action before the valve is returned to service.

Based on inadequate check valve testing and the '502B' check valve observed to be in the fully open position during sustained backpressure and reverse flow produced by the running 'D' HPSW pump on August 12, 2022, the inspectors questioned the current operability of check valve '502B' and its 'B' HPSW subsystem. The inspectors noted that the Updated Final Safety Analysis Report described single failures for the safety analysis to include the stopping of any single component, such as an operating HPSW pump. The inspectors also noted that Constellation's past engineering evaluation which the current operability determination referenced, described assurance of check valve closure based on the force applied from backpressure/reverse flow exceeding manual closure force. Finally, the inspectors noted that check valve '502B' was last successfully exercised and/or tested on June 30, 2022. However, the prior testing did not ensure valve closure with backpressure since evidence existed that equipment operators manually closed stuck open check valves. In addition, operators did not necessarily initiate new condition reports for present conditions when an open work request and previous condition report existed for a deficiency.

As a result of the inspectors' questions, Constellation tested check valve '502B' on September 5, 2022, using a revised test procedure in which the valve was observed to open when the 'B' HPSW pump was run, stick open when the 'B' HPSW pump was shutdown, and then close as required after the 'D' HPSW pump was started. Constellation then performed corrective maintenance on '502B' on September 8, 2022. Maintenance technicians identified the cause of the sticking open to be excessive packing tightness and replaced the shaft, bushings, and packing of the valve. Engineering also evaluated and approved, and maintenance installed, Teflon packing which significantly reduced the resistance to swingarm motion. Constellation completed the revised testing of all Unit 2 and Unit 3 HPSW pump discharge check valves on September 13, 2022.

Corrective Actions: Constellation revised the surveillance test procedures and briefed the operating crews to ensure proper check valve closure testing, tested '502B' successfully, performed corrective maintenance on '502B' that reduced friction to motion, and completed revised testing of all HPSW pump discharge check valves.

Corrective Action References: IR 4516972 and IR 4522304

Performance Assessment:

Performance Deficiency: The inspectors determined that the failure to perform adequate closure testing of the HPSW discharge check valves and failure to correct the sticking open of check valve '502B' in a timely manner was a failure to meet ASME OM Code and corrective action requirements, was reasonably within Constellation's ability to foresee and correct, and should have been prevented. Specifically, testing of the HPSW pump discharge check valves did not ensure in-service closure testing was conducted in accordance with the IST program plan and ASME OM Code, and sticking open of '502B' was first identified in 2016 and not resolved prior to the valve being stuck open with backpressure and reverse flow.

Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the HPSW pump discharge check valves' safety function includes closing, and the '502B' check valve was observed to be open with sustained backpressure and reverse flow.

Significance: The inspectors assessed the significance of the finding using IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." The inspectors determined that the finding was of very low safety significance (i.e., Green). Specifically, the probabilistic risk assessment (PRA) safety function of the HPSW system requires a single HPSW pump providing cooling flow to a RHR heat exchanger. In addition, a check valve sticking open and not closing would not, by itself, cause a loss of the PRA safety function of a train or system absent additional concurrent failures. Finally, the inspectors considered the result of the revised IST tests in which all HPSW discharge check valves did successfully close either upon pump shutdown or with sustained backpressure and reverse flow applied for a specified time. Therefore, the finding screened to Green based on the determination that the finding did not cause a loss of the PRA safety function for a train or system.

Cross-Cutting Aspect: P.2 - Evaluation: The organization thoroughly evaluates issues to ensure that resolutions address causes and extent of conditions commensurate with their safety significance. Constellation personnel did not thoroughly evaluate within the CAP, the sticking open of HPSW pump discharge check valves when the condition occurred, recurred, and/or was a long-standing degraded condition. In particular, the sticking open of '502B' was first identified in 2016, and corrective actions were not prioritized to resolve the degraded condition because prior evaluations accepted the condition. In addition, the prior engineering evaluation was re-used and re-affirmed when new condition reports were initiated for HPSW pump discharge check valve sticking.

Enforcement: Violation: 10 CFR 50.55a(f)(4) requires, in part, that valves that are within the scope of the ASME OM Code must meet the in-service test requirements. 2012 ASME OM Code Edition, the code edition of record for the interval, ISTC-3522, “Category C Check Valves,” requires each Category C check valve to be exercised or examined in a manner that verifies obturator travel by using the methods in ISTC-5221. ISTC-5221, “Valve Obturator Movement,” requires necessary valve obturator movement during exercise testing to be demonstrated by performing both an open and a close test and the exercise test to verify that on cessation or reversal of flow the obturator has traveled to the seat. ISTC-5224, “Corrective Action,” requires that if a check valve fails to exhibit the required change of obturator position, it shall be declared inoperable. ER-PB-321-1000, “IST Program Plan, 5th Ten Year Interval,” the IST program plan for the PBAPS 5th ten-year interval, required Category C check valves to be tested in the closed direction and that check valves which fail to exhibit the required change of disk position per this testing shall be considered inoperable. Contrary to this, until September 13, 2022, the HPSW pump discharge check valves were within the scope of the ASME OM Code but did not meet the in-service test requirements. Specifically, exercise tests were performed on the HPSW pump discharge check valves that did not verify that on cessation or reversal of flow the obturator has traveled to the seat, and check valves that failed to exhibit the required change of obturator position during testing were not declared inoperable. Enforcement Action: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.

Observation: Root Cause Investigation Report 4460767 Reviewed for Unit 2 Reactor Scram	71152A
The inspectors performed a detailed review of Constellation’s root cause evaluation associated with IR 4460767, "Peach Bottom Unit 2 Manual Reactor Scram." The root cause was written to capture and address the Peach Bottom Unit 2 reactor scram that occurred on November 14, 2021. Constellation captured the event accurately, in an appropriate manner, and identified actions to perform in response to the event which were reasonable. As part of this inspection, the inspectors evaluated LER 05000277/2021-003-00 and the conclusions associated with this LER are documented in this report under the Inspection Results Section, FIN 05000277/2022003-01. Based on the documents reviewed and discussions with Constellation personnel, the inspectors determined that, in general, Constellation identified problems and entered them into the CAP at a low threshold, corrective actions were commensurate with the safety significance, and appropriately addressed the deficiency.	

Observation: Elevated Vibrations on the Unit 3 HPCI Steam Supply Piping	71152A
The inspectors reviewed the licensee’s evaluations and corrective actions, including 40 issue reports, associated with the elevated vibrations on the steam supply line for the Unit 3 HPCI system. Based on discussions with the licensee staff and review of plant documentation, the inspectors noted the elevated vibrations on the steam line have likely existed since initial plant start-up. The inspectors noted several conditions have occurred in the past four years such as a through-wall leak on the HPCI steam sensing line in 2018 that required a plant shutdown for repair and sheared bolting on a support for a steam drain valve in 2020. Licensee staff performed a cause investigation in response to the 2018 sensing line leak, performed vibration analysis of the HPCI system which included instrumenting the 10-inch	

steam supply and the two 1-inch steam sensing lines, generated new piping models of the HPCI system, and reanalyzed the piping stresses based on the results of the vibration analysis. The licensee determined the likely cause of the vibrations was due to acoustic excitation of the system (an inherent design characteristic from the piping configuration) and concluded the elevated vibrations would not impact the fatigue limits of the piping. The inspectors observed the licensee's planned corrective actions to maintain monitoring of HPCI system piping supports to help ensure the piping remained unimpacted from continued vibration. The inspectors did not identify any findings or violations.

Observation: 'E-3' EDG Testing Did Not Open the 'E-32' Circuit Breaker	71152A
On October 20, 2021, during the ‘E-3’ EDG 24-hour endurance surveillance test, the ‘E-32’ emergency auxiliary switchgear bus output circuit breaker failed to trip when required once technicians installed a jumper to simulate a loss-of-coolant accident signal. The cause of the ‘E-32’ breaker failure to trip was high resistance in the control wire/cable. To restore the ‘E-3’ EDG, technicians swapped the control function to a spare conductor after verifying it was in good condition. Constellation later performed more extensive cable testing which indicated more degradation. Therefore, in March 2022, Constellation installed new control cable for the ‘E-3’ EDG. Constellation sent portions of the removed cable to a vendor for further testing and failure analysis. The vendor determined that the degradation was primarily due to 1) gradual infiltration of moisture into the cable, 2) conductor degradation and the formation of copper hydroxide and other contaminants through chemical reactions involving the conductor and the water that infiltrated the cable, and 3) migration of these contaminants into the insulation polymers, inducing degradation of the materials. This progressive age-related degradation process led to the formation of ionic compounds in the cable that increased the leakage current through the insulation polymer, which eventually resulted in the high impedance/low insulation resistance fault that caused the cable to fail. The inspectors observed the cable monitoring and cable replacement and reviewed Constellation’s evaluations and the vendor report. The inspectors noted the degraded wires were low voltage control cables routed largely through underground and embedded conduit and the failure was a progressive age-related degradation process. The inspectors determined Constellation’s corrective actions, including actions to develop and implement extent of condition and ongoing low voltage cable testing and monitoring, were commensurate with the safety significance and appropriately addressed the deficiency. The inspectors did not identify any findings or violations of more than minor significance. Based on the documents reviewed and discussions with Constellation personnel, the inspectors noted that, in general, Constellation identified problems and entered them into the CAP at a low threshold.	

Failure to Determine and Correct Adverse Condition Leads to Reactor Scram			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Initiating Events	Green FIN 05000277/2022003-02 Open/Closed	[H.13] - Consistent Process	71153
A self-revealed Green finding was identified when Constellation failed to perform an adequate functional review of an action request. Specifically, Constellation failed to identify jet			

compressor suction pressure control valve CV-2-08-8417A was affected by a nearby steam leak when doing a functional review of the components in the area.
Description: On May 9, 2019, a steam leak was identified in the Unit 2 ‘A’ jet compressor room from the offgas system. Constellation entered the issue into their CAP (IR 04237919). The one foot steam plume was noted to be from piping located about 1 foot above the floor and caused condensation to build on the ceiling. Per step 4.4.6.6 of PI-AA-120, “Issue and Identification and Screening Process,” Constellation performed a functional assessment to determine if the as found functionality of any SSC was affected by the condition described in the issue and documented the basis of the determination. Functionality assessments, as defined in OP-AA-108-115, “Operability Determinations (CM-1),” are the decision made by a senior licensed reactor operator (SRO) on the operating shift crew as to whether an identified or postulated condition has an impact on the functionality of a SSC. Furthermore, the procedure states, in part, that functionality assessments must be documented in sufficient detail to allow an individual knowledgeable in the technical discipline associated with the condition to understand the basis for the decision. The SRO wrote in the functionality assessment for IR 04237919 that the steam leak was not impacting or affecting any of the other equipment in the room. Additionally, engineering recommended to not attempt to isolate the leak at the time of discovery. However, the SRO and engineering personnel failed to capture that the steam leak affected the soft part materials (gaskets and diaphragm) of CV-2-08-8417A. In particular, the component was located near the steam leak, the room is a small confined area subject to significant heat-up given a steam leak, and the soft parts already had a four-year planned preventive maintenance replacement due to a short expected life from the typical high room temperature. The inspectors determined that a steam leak in a confined area or near other equipment would reasonably affect other SSCs and therefore would require further evaluation to assess the impact of the condition. The inspectors also noted that IR 04237919 was written without using Constellation’s steam leak template, which is a tool available to operators that provides additional information to consider for functionality assessments. Finally, the inspectors noted that Constellation determined that CV-2-08-8417A was not characterized correctly in accordance with the Single Point of Vulnerability classification program and that correct classification would have provided additional information for the functionality assessment. As a result, the soft parts of CV-2-08-8417A failed, and the valve lost the ability to regulate steam flow. The valve failure resulted in degraded condenser vacuum, and the operators inserted a Unit 2 manual reactor scram on November 14, 2021. Corrective Actions: Constellation replaced the soft parts constructed of lower temperature materials in CV-2-08-8417A with soft parts rated for higher temperatures. Constellation performed a root cause analysis to identify additional contributing causes which led to the reactor scram. Corrective Action References: IR 04237919
Performance Assessment: Performance Deficiency: The inspectors determined that the failure to perform an adequate functionality review of the offgas system steam leak in 2019 was a failure to meet PI-AA-120, Step 4.4.6.6, was reasonably within Constellation’s ability to foresee and correct, and should have been prevented. Specifically, a high temperature steam leak in a confined area would

reasonably affect other SSCs within the area and require further evaluation. The available steam leak template also described these considerations but was not used. Lastly, the misclassification of CV-2-08-8417A contributed to the insufficient functionality assessment by not flagging higher importance equipment in the room. Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the Equipment Performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inadequate functionality review resulted in CV-2-08-8417A degrading to the point where the valve could not regulate steam flow which resulted in degraded condenser vacuum and a manual scram of the unit. Significance: The inspectors assessed the significance of the finding using IMC 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power.” The inspectors determined that this finding was of very low safety significance (Green) because it did not cause a reactor trip coincident with the loss of mitigation equipment relied upon to transition the plant from the onset of a reactor trip to a stable shutdown condition. Cross-Cutting Aspect: H.13 - Consistent Process: Individuals use a consistent, systematic approach to make decisions. Risk insights are incorporated as appropriate. Constellation personnel made a functionality screening decision without reviewing available information contained in procedures and other supporting documents, and a systematic review would reasonably have resulted in a different outcome. The inspectors determined that the cross-cutting aspect reflected present performance. The inadequate functionality assessment was slightly more than three years ago. However, the organizational learning from the scram is relatively recent and a consistent approach was not established or communicated by Constellation personnel.
Enforcement: Inspectors did not identify a violation of regulatory requirements associated with this finding.

Loss of Reactor Protection System Power and Unit Scram Due to Operator Error			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Initiating Events	Preliminary White AV 05000277/2022003-03 Open EA-22-071	[H.1] - Resources	71153
The inspectors identified a self-revealing preliminary White finding and an associated apparent violation (AV) of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," because Constellation failed to accomplish an activity affecting quality using a procedure appropriate to the circumstances. Specifically, Constellation failed to implement the pre-planned “Partial” of procedure SO 60F.1.A-2/3, "Reactor Protection System MG Set and Power Distribution system Startup from Dead Bus Condition," when responding to the trip of the Unit 2 ‘A’ reactor protection system (RPS) motor generator (MG) set while Unit 2 ‘B’ RPS was supplied via the alternate feed and instead used the “clean” copy of procedure SO 60F.1.A-2, which was not appropriate to the circumstances. As a result, an operator performed a procedure step which opened the output breakers associated with the alternate RPS feed causing a Unit 2 reactor scram, primary containment isolation system (PCIS)			

Group I isolation, safety-relief valve (SRV) actuation, and loss of the normal heat sink which required emergency core cooling systems (ECCS) to maintain level and pressure.
<u>Description</u>: The PBAPS RPS is a safety-related plant protection system that consists of two trip systems (‘A’ and ‘B’). The primary function of the RPS is to initiate rapid insertion of the control rods (i.e., scram) by removing power. The RPS is a normally energized system and deenergizing a RPS trip system places that system in a tripped condition which makes the RPS fail safe in case of loss of electrical power. The RPS power is also used as a source of power for the instrumentation whose trip outputs go to normally energized logic such as the RPS, certain functions of the PCIS, the neutron monitoring system, the radiation monitoring system, the reactor instrumentation system, and the control rod drive system. Power for the two RPS power buses is supplied from two independent MG sets equipped with flywheels that have sufficient energy storage capability to prevent RPS trip actuations during short power losses. If one MG set is lost from service a half scram occurs along with associated isolations. Following the half scram, the alternate source select switch transfers the load from that MG set to the alternate feed transformer (a solid-state battery-backup system). The scram solenoids deenergized by the half scram are then reset and the plant is returned to normal operation. Return from the alternate feed transformer back to the MG set is initiated from the same transfer switch. The capability of switching to an alternate power source prevents the RPS from having to remain in a half scram condition for extended periods of time. However, the alternate feed can supply only one train at a time (i.e., either ‘A’ or ‘B’ but not both). Beginning April 20, 2022, all PBAPS 4kV buses were lined up to one source due to a planned offsite source outage window. Therefore, one RPS bus on each unit was lined up to the alternate feed (a solid state, battery back-up system) to prevent a loss of all RPS on an electrical transient or fast transfer due to loss of offsite feed. Unit 2 had the ‘B’ RPS bus aligned to the alternate feed with the ‘A’ RPS bus supplied from its MG set. An Operation’s “white paper” was created outlining the offsite source outage that included compensatory measures and plant response if there was a loss of the offsite feed in the off-normal lineup. Compensatory measures included creating a pre-planned “partial” of procedure SO 60F.1.A-2/3, “RPS MG Set and Power Distribution System Startup from Dead Bus Condition,” to re-start a tripped RPS MG set. This was necessary because the alternate feed can only supply one RPS bus at a time, and the non-marked up procedure, SO 60F.1.A-2/3, required the operator to verify the other division was on the MG set and then transfer the tripped division of RPS to the alternate feed. As such, the full non-marked up procedure did not align with the plant conditions that existed in the maintenance configuration. On May 16, 2022, an offsite electrical transient caused by a switchyard/grid equipment issue caused multiple main generator “thumps” and various control room alarms for both units. Among other equipment issues, the Unit 2 ‘A’ RPS MG set tripped causing a de-energization of the ‘A’ RPS bus, a half scram condition, Group II/III inboard primary containment isolations, and tripping of Unit 2 reactor building ventilation. The operating crew prioritized restoration of the Unit 2 ‘A’ RPS MG set which would allow re-powering the ‘A’ RPS bus, resetting the half scram, resetting the isolations, and restarting reactor building ventilation. However, to restore the ‘A’ RPS MG set the operators used a “clean” copy of SO 60F.1.A-2 and did not use the pre-planned “partial” of SO 60F.1.A-2. Due to this and subsequent errors, including an inadequate “partial” that was developed in the moment, the operators opened the output breakers from the RPS alternate feed supplying the ‘B’ RPS bus. The total loss of power to both ‘A’ and ‘B’ RPS caused a Unit 2 reactor scram and Group I isolation (main

steam isolation valves (MSIV) closure). The plant operated as designed in response to the loss of RPS power and the crew took actions to stabilize the plant in accordance with procedures. The MSIV closure caused a reactor high-pressure condition that lifted SRVs and caused both reactor recirculation pumps to trip. HPCI and RCIC were manually started to stabilize and control reactor pressure and level. Constellation performed a root cause evaluation to determine the human performance and technical human performance causes of the event. Constellation also performed a separate work group evaluation to address gaps in interactions with the grid operator, and a separate equipment causal evaluation to address equipment issues. Constellation determined that breakdowns in technical human performance occurred at all levels of the operating crew that allowed the operating crew to operate in knowledge-based mode and direct the wrong procedure steps during a high-stress situation, resulting in a reactor scram. As a result, an operator opened the output breakers associated with the alternate RPS feed leading to a reactor scram, PCIS Group I isolation, SRV actuation, and loss of the normal heat sink which required ECCS to maintain level and pressure. The inspectors observed the immediate response actions, reviewed initial corrective actions, and later reviewed the completed root cause evaluation. The inspectors determined that the evaluation properly identified the causes of the event and appropriate corrective actions. Corrective Actions: Constellation created a new procedure to provide more condition specific direction when responding to RPS MG set trips, implemented personal accountability actions, and incorporated lessons from the event in Operations' training. Corrective Action References: IR 4500178
<u>Performance Assessment</u>: Performance Deficiency: The inspectors determined that using the “clean” copy of SO 60F.1.A-2 with an inadequate partial and not using the pre-planned partial of procedure SO 60F.1.A-2 when responding to the trip of the Unit 2 ‘A’ RPS MG set was a failure to meet the requirement of 10 CFR Part 50, Appendix B, Criterion V, to accomplish an activity affecting quality using a procedure appropriate to the circumstances. This was reasonably within Constellation’s ability to foresee and correct and should have been prevented. Specifically, operators responded to the trip of Unit 2 ‘A’ RPS by using the clean copy of procedure SO 60F.1.A-2 that was marked-up in the moment and opened the output breakers of the alternate feed supplying ‘B’ RPS which caused a Unit 2 reactor scram, PCIS Group I isolation, SRV actuation, and loss of the normal heat sink and required ECCS to maintain level and pressure. Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the Human Performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors also determined the performance deficiency affected the Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the finding caused a reactor trip and loss of the normal heat sink and feedwater.

Significance: The inspectors assessed the significance of the finding using IMC 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power.” Using IMC 0609, Appendix A, “Exhibit 1-Initiating Events Screening Questions,” under B, “Transient Initiators,” the inspectors determined the finding required a detailed risk evaluation because the finding caused both a reactor trip and the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition (e.g. loss of power conversion system). A Region I Senior Reactor Analyst (SRA) performed a detailed risk evaluation. The finding was preliminarily determined to be of low to moderate safety significance (White). The dominant core damage sequences consisted of a loss of condenser heat sink, failure of high-pressure injection and failure to manually depressurize the reactor. See Attachment, “Initiating Event MSIV Closure Detailed Risk Evaluation,” for a detailed review of the quantitative and qualitative criteria considered in the preliminary risk determination. Cross-Cutting Aspect: H.1 - Resources: Leaders ensure that personnel, equipment, procedures, and other resources are available and adequate to support nuclear safety. Specifically, the operator training, direction, and monitoring implemented by leaders did not adequately improve performance and lacked sufficient accountability. Before the event leaders did not ensure correct understanding of the plant configuration and procedure to be implemented, and during the event leaders allowed inappropriate time pressure and did not ensure error reduction tools were appropriately used.
Enforcement: Violation: 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions or procedures of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions or procedures. Contrary to the above, on May 16, 2022, Constellation did not accomplish an activity affecting quality in accordance with the prescribed procedure appropriate to the circumstances. Specifically, the pre-planned procedure, a “Partial” of SO 60F.1.A-2, "Reactor Protection System MG Set and Power Distribution System Startup from Dead Bus Condition," was prescribed to respond to an anticipated transient involving the loss of the Unit 2 ‘A’ RPS power source when the ‘B’ RPS power source was supplied via the alternate feed, but this prescribed procedure was not implemented when required, and the use of a “clean” procedure SO 60F.1.A-2, marked-up in the moment, was not appropriate to the circumstances. Enforcement Action: This violation is being treated as an apparent violation pending a final significance (enforcement) determination.

EXIT MEETINGS AND DEBRIEFS

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

- On October 14, 2022, the inspectors presented the integrated inspection results to Dave Henry, Site Vice President, and other members of the licensee staff.
- On July 21, 2022, the inspectors presented the IP 71124.06 Radiological Gaseous and Liquid Effluent Treatment Systems inspection results to Ron DiSabatino, Plant Manager, and other members of the licensee staff.

DOCUMENTS REVIEWED

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
71111.04	Procedures	M-362	P&I Diagram Core Spray Cooling System	Revision 64
		M-377, Sheet 1	Diesel Generator Auxiliary Systems (Air Coolant and Jacket Coolant Systems	Revision 45
		M-377, Sheet 2	Diesel Generator Auxiliary Systems (Air Coolant and Jacket Coolant Systems)	Revision 45
		ST-O-014-350-2	Core Spray Loop 'A' Valve Alignment and Filled and Vented Verification	Revision 5
		ST-O-014-355-2	Core Spray Loop 'B' Valve Alignment and Filled and Vented Verification	Revision 5
71111.05	Procedures	PF-132	Diesel Generator Building, General Area, Elevation 127'-0"	Revision 9
		PF-132A	Diesel Generator Building, General Area (Upper Level)	Revision 4
		PF-144	Circulating Water Pump Structure, General Area	Revision 7
		PF-80	Turbine Building Common, Deck Area-Elevation 165'	Revision 10
		PF-81	Turbine Building Common, Hatch Area-Elevation 116'	Revision 10
71111.12	Corrective Action Documents	Issue Reports (IRs) 04355756 04400698 04421253 04457427 04466868 04489142 04515723 04520791 04521045 04522137 04523718		
71111.13	Corrective Action Documents	IR 04510157		
	Corrective Action Documents Resulting from	*IR 4515664		

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
	Inspection			
	Miscellaneous	RMTS Title 3.3.6.1	RCIC PCIS Group V Defeated Due to TE-5939B Failure	
	Procedures	ER-AA-600-1052	Risk Management Support of RICT	Revision 1
		ER-AA-600-1053	Calculation of RMAT and RICT for Risk Informed Completion Time Program	Revision 0
		ER-PB-600-2001	Peach Bottom RICT System Guidelines	Revision 1
		OP-AA-108-118	Risk Informed Completion Time	Revision 2
71111.15	Corrective Action Documents	IR 04475907	2BP040 SBLC Pump Sight Glass Webpage	
		IR 4501237 IR 4512496		
		IR 451239 IR 4513261		
		IR 4512496 IR 4501237		
		IR 4523239 IR 45413261		
	Corrective Action Documents Resulting from Inspection	IRs 04524722	NRC ID - 2 'B' Standby Liquid Control (SBLC) Pump Oil Level Below Static Minimum Level	
	Miscellaneous	Engineering Change 637336		
71111.19	Corrective Action Documents	AR 04521792		
		IR 04506392		
	Procedures	M-057-014	Cyberex 125 volt Battery Charger Maintenance	Revision 18
		M-506-005	Valve Packing	Revision 15
		M-510-107	Inspection and Refurbishment of Atwood and Morrill Mark No. 234 & 237 Swing Check Valves	Revision 26
		MA-AA-723-300	Diagnostic Testing of Motor Operated Valves	Revision 14
		MA-AA-723-301	Periodic Inspection of Limitorque Model SMB/SB/SBD-000 through 5 Motor Operated Valves	Revision 15
		MA-PB-742-003	"Rack Calibration of Rosemount Model 710DU Trip Units	Revision 3
		SI2A-2-RPS-	Functional Test of RPS 'D' Card File	Revision 3

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		D1F1		
		ST-O-010-306-2	'B' RHR Loop Pump, Valve, Flow and Unit Cooler Functional and Inservice Test	Revision 53
	Work Orders	04261335		
		04510502		
		05289668		
		05292004		
		WO 04235267		
71111.22	Procedures	SO 14.7.B-3	Manual Operation of Core Spray System with Discharge to Torus	Revision 4
		SO39.1.A	FLEX Generator Startup and Shutdown	Revision 5
		ST-O-032-301-2	HPSW Pump, Valve and Flow Functional and Inservice Test	
	Work Orders	05246865		
71152A	Corrective Action Documents	04237919		
		IR 4175898 IR 4193409 IR 4290810 IR 4316593 IR 4455580 IR 4456711		
		IR 04454290 IR 04454298 IR 04454795 IR 04467125 IR 04467861 IR 04487128 IR 04495492 IR 04500608		
	Engineering Evaluations	Investigation Report #4460767	Root Cause Investigation Report: Unit 2 offgas system failure led to degrading condenser vacuum necessitating a manual SCRAM.	12/19/2021
	Miscellaneous	EC 635360 EC 635851 EC 635923		

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		Report Number: PCH220601R1-F	Results of Laboratory Testing Performed on Cables Removed from Manhole 40 at Peach Bottom Atomic Power Station, July 2022	
	Procedures	OP-AA-108-115	Operability Determinations (CM-1)	Rev 24
		PI-AA-120	Issue Identification and Screening Process	Rev 12
	Work Orders	5181067 5197714 5206719 5223390 5228431		
71153	Corrective Action Documents	IR 04500178		
	Miscellaneous	COL GP-18	SCRAM Review Procedure Checklist, May 16, 2022	
		M-1-S-70, Sheet 2	Electrical Schematic Diagram Reactor Protection System MG Set Control	Revision 14
		T-BAS	Introduction to Trips and SAMPS – Bases	Revision 16
	Procedures	AD-PB-101-1003	Temporary Changes to Approved Documents and Partial Procedure Use	Revision 15
		COL 1A.7.A-2	Main Steam System Lineup After a Group I Isolation	Revision 2
		HU-AA-104-101	Procedure Use and Adherence	Revision 7
		SO 1A.1.A-2	Main Steam System Startup	Revision 9
		SO 1A.7.A-2	Main Steam System Recovery Following a Group I Isolation	Revision 5
		SO 60F.1.A-2	Reactor Protection System MG Set and Power Distribution System Startup from Dead Bus Condition	Revision 10
		SO 60F.1.A-2	Reactor Protection System MG Set and Power Distribution System Startup from Dead Bus Condition	Revision 11
		SO 60F.1.B-2	Reactor Protection System MG Set Startup Following an RPS MG Set Trip	Revision 0
		T-221-2	Main Steam Isolation Valve Bypass	Revision 15

ATTACHMENT

Peach Bottom Atomic Power Station Unit 2

Initiating Event MSIV Closure
Detailed Risk Evaluation

Conclusion:

The increase in risk represented by this performance deficiency (PD) is the calculated Incremental Conditional Core Damage Probability (ICCDP).
ICCDP=Conditional Core Damage Probability (CCDP) – Core Damage Probability (CDP)
ICCDP= 6.1E-6 (CCDP) – 2.6E-7 (CDP) = 5.8E-6

The calculated increase in risk was estimated to be Preliminary White, or a finding of low to moderate safety significance.

Background:
This significance determination process (SDP) evaluated the closure of the main steam isolation valves (MSIVs) and the isolation of the reactor from its normal heat sink. This event results in the bulk of the reactor decay heat going into the containment (torus) rather than the normal heat sink (main condenser through the turbine bypass valves). Additional inventory will be sent to the torus through the turbine exhaust systems of the high-pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems depending on their operating status/functionality. Because the normal heat sink is lost through the closure of the MSIVs, the main turbine bypass valve pressure set function cannot be used to control the reactor pressure as well. The result is a multitude of potential core damage sequences at a higher frequency due to the MSIV closure event along with the various failure probabilities associated with mitigating equipment, potentially increasing workload and challenges to the operational staff.

This PD resulted in a significant increase in the frequency of the loss of condenser heat sink event through the closure of the MSIVs. The nominal frequency for this type of event in the NRC Standardized Plant Analysis Risk (SPAR) model is 1 in 25 per year and this PD was associated with an actual MSIV closure event occurring in the given year (i.e., set to 1/yr). For SDP evaluations, the CCDP is multiplied by one inverse year (1/year) to equate this to a change in average core damage frequency (CDF). A plot of core damage probability (CDP) for a given unit versus time would show a spike in CDP due to this initiating event. This spike, or dirac delta function, represents the impact to the plant risk profile relative to the actual event happening well over its expected frequency.

The risk assessment of operational events handbook (RASP) manual, Volume I – Internal Events, Revision 2.02 defines the guidance on how to evaluate the risk increase of initiating events. The RASP manual defines what is referred to as the “failure memory approach.” This is used to estimate the risk significance of operational events. In the failure memory approach, the basic events associated with the observed failures and other off-normal situations are configured to be failed (i.e., MSIVs closed, initiating event loss of condenser heat sink set to 1.0), while observed successes and unchallenged components are assumed capable of failing with their nominal probabilities. In summary, this is an evaluation of the impact of the actual event occurring with a multitude of potential probabilistic failures of mitigating equipment and operator actions.

In summary, the SDP does not model what actually occurred with equipment successes and operational success actions, it models the probabilistic impact of the various core damage sequences given the normal heat sink and high-pressure reactor feedwater pump injection were impacted due to the PD.

Influential Assumptions

Exposure Time

This was an initiating event (IE) evaluated SDP. Exposure time is not directly applicable due to the methodology used in evaluating the impact of the event. Inspection Manual Chapter (IMC) 0308, Attachment 3, “Technical Basis for Significance Determination Process,” Section 0308.03-08.02, “Treatment of Degraded Conditions and Initiating Events,” defines how to treat these events. The IE frequency was set to 1.0 for the loss of condenser heat sink event (i.e., because the IE actually occurred). The ICCDP estimate represents the increase in risk to the plant based on the actual change in the expected frequency of the event (i.e., 1/yr vs. nominal expected frequency over the year). The event is basically a dirac delta function or spike in core damage probability which has an overall impact to the plant risk profile in a given year.

High-Pressure Injection (HPI) Basic Event Changes

The senior reactor analyst (SRA) determined that the HPCI and RCIC system failure-to-run (FTR) basic event values should be adjusted to prevent inappropriate skewing of the risk contribution for FTR basic events. This was performed to reflect the as-built, as-operated plant and procedures. Evaluating FTR event impact with dynamic PRA models is beyond the scope of existing SPAR models and SDP analysis. Modeling of this nature regarding dynamic consideration of all possible scenarios is not consistent with the principles of scrutability, consistency and timeliness for SDPs as outlined within IMC 0308. Notwithstanding this, the SRA has incorporated a simplified surrogate method to recognize that some early success (4 to 6 hours) of HPCI or RCIC should be accounted for. PB-PRA-004, Peach Bottom PRA Human Reliability Analysis (HRA) Notebook Volume I, human error probability (HEP) event A22, references that maximization of the control rod drive system flow for reactor pressure vessel injection per plant procedures is a viable makeup method for level if there is early success of HPI systems.

The SRA noted that the SPAR model does not recognize that early success (4 to 6 hours) with subsequent failure could allow for the control rod drive (CRD) to be maximized to match decay heat inventory requirements. This is an estimate considering that decay heat removal for cooldown along with sensible heat in the reactor coolant system (RCS) are also of consideration. The SRA used a simplified method as a surrogate to support a best estimate risk assessment. As noted, the approach is consistent with the as-built as-operated plant and procedures and results in what the SRA considers to be a more accurate non-skewed risk result due to the elevated HPCI and RCIC 24-hour FTR value.

Basic Event, ADS-XHE-XM-MDEPR, Operator fails to initiate reactor depressurization

The SRA noted that the most dominant basic event with the highest risk achievement worth (RAW) was represented by the failure of operators to initiate reactor depressurization within the core damage cutsets, given the MSIV closure event (loss-of-condenser-heat-removal event). The SRA noted that Section 9.2 of the RASP handbook, Volume I, gives specific guidance on quantifying human error probabilities (HEPs). The guidance states that a key aspect of the SPAR-H method to ensure consistency, if the human failure event (HFE) is already defined and modeled, is the utilization of the SPAR-H method to quantify the HEP. Other human reliability analysis (HRA) methods used to quantify HEPs may be used as sensitivity analyses.

This dominant HEP is modeled within the Peach Bottom SPAR model as an action that is well-trained, and highly reliable. Therefore, no change to the HEP for this action was performed. Constellation's Peach Bottom model of record (MOR) had two separate HEPs related to depressurization actions, one being to depressurize to below 600 psig with its own calculated HEP in support of condensate injection. The other being the failure to depressurize completely to below the shutoff head of lower pressure systems such as low pressure coolant injection (LPCI) or core spray. The SRA noted that the SPAR model does not differentiate between levels of pressure relative to depressurization. If depressurization is not successful then high pressure systems such as HPCI, RCIC and reactor feedwater pumps, which can inject against SRV pressure, are initially required. The depressurization is either successful to allow lower pressure systems to inject (condensate, LPCI, Core Spray) or it is not. Furthermore, the SRA noted that the existing Constellation PRA MOR, approved at the time of this event, did not differentiate between levels of depressurization. If either depressurization basic event (i.e., below 650 psig or complete depressurization) failed then the function was failed (i.e., OR-gate-complete dependence).

During this SDP evaluation Constellation determined that within their model, a successful action of inhibiting the automatic depressurization system (ADS) function would break any dependency they assumed to have existed and allow for treating the events with zero dependency. The SRA noted this was not relevant to a SPAR model assessment as that assumption does not apply when using the SPAR model for this SDP. The SPAR model does not contain an inhibit ADS basic event within the applicable core damage sequences of interest. That event is only used within anticipated transient without scram scenarios and has no impact for this SDP. In summary, it was determined the nominal base case fail to depressurize HEP of 5E-4 is valid. Notwithstanding this, the SRA recognized uncertainties with this basic event value and performed adjustments to review this HEP value. This included using an independent method, IDHEAS-ECA, to further evaluate this HEP in the sensitivity analyses section within this SDP.

Joint Human Error Probabilities (JHEPs)

RASP handbook Section 9.4, Analysis of Dependencies, states an analyst should not use a minimum joint HEP (JHEP) of less than 1E-6 for SDP analyses. Therefore, an SDP analysis always assumes some level of dependence between human failure events (HFEs), even if specific reasons for that dependence cannot be identified. Therefore a few JHEPs were created for cutsets where HFEs resulted in dropping below the 1E-6 value. These were HFEs with a combined 1E-7 value and included operators failing to refill the condensate storage tank (CST), failing to recover the power conversion system (PCS) and failing to initiate residual heat removal (RHR) suppression pool cooling. The SRA noted that these events were related to containment heat removal (CHR) and late injection. The second JHEP consisted of HFEs related to failure to initiate suppression pool cooling, failure to recover the power conversion system, and failure to vent containment. These impact CHR and late injection and potential dependencies appear to exist supporting this JHEP minimum cutoff value. However, the JHEP values will be adjusted in the sensitivity analyses section. The SRA noted the overall values for these few JHEPs did not represent significant increases in the combined failure probability using this guidance but was consistent with recognizing overall dependencies within HFEs.

Standardized Plant Analysis Risk (SPAR) Model Changes for SDP evaluation

High Pressure Injection

To address the high-pressure injection potential conservatisms using a FTR value for 24 hours and subsequent failure probabilities of 0.175, the SRA applied the following formula as a surrogate for a modified fault tree. The intent was to recognize the emergency operating procedures (EOPs) which direct the ability for maximization of the CRD system to address reactor pressure vessel (RPV) level issues. The following will be applied to RCIC and is applicable to HPCI as well.

RCIC fails within an early window of an estimated 4 to 6 hours with uncertainties, OR (OR gate) RCIC would fail within its 24-hour mission time ANDed (AND gate) x failure of a modified CRD fault tree approach recognizing the need to maximize CRD flow and address the suction source from the CST. The SRA created a new basic event, CRD-XHE-XM-MAXCRD and used SPAR-H with diagnosis and action considerations. All performance shaping factors (PSFs) were set at nominal with high stress invoked. The result was a failure probability of 2E-2. However, the intent is a surrogate, and this action would be part of sequences with the failure to depressurize in core damage cutsets. Therefore, the analyst factored in dependency from a cutset level. Although control rod drive (CRD) and safety relief valves (SRVs) are diverse systems both are related to restoring RPV level with similar cues in the main control room (MCR) area and same crew. For this estimation a moderate dependency was assumed with the formula (1+ 6 x (2E-2)/7 from NUREG CR/6883 Appendix C. Therefore, a modified CRD fault tree was solved with a 0.16 HEP within it.

The RCIC late failure term, ZT-TDP-FR-L-HCI-RCI5, was created using a 5-hour mission time to capture any uncertainties. The RCIC fail to run early term was then calculated to be 4.35E-2. If RCIC fails in this early window, or a combination of failing to run for 24 hours with the failure of CRD, this would equate to an estimated FTR value for RCIC. The simplified surrogate formula for a best estimate of RCIC FTR would then be:

RCIC FTR early window + RCIC fails to run (FTR) for 24-hour mission (RCIC FTR 24hr) x (failure of modified CRD fault tree crediting ability of CRD if early success of RCIC).

Thus, 4.35E-2 + (0.175 RCIC FTR 24 hours) x 0.16 (modified CRD fault tree (FT) with dependency considerations) resulting in 4.35E-2 + 2.8E-2 = 7E-2. The SRA noted a low CRD dependency would result in a modified CRD FT of 7E-2. This would change the above calculation to 5.6E-2. Therefore, a nominal estimated value of 6E-2 will be used as a surrogate for consideration of the as-found, as-built plant and procedures in place. This would apply to HPCI as well. For clarification, the actual CRD fault tree within the SPAR was not revised from its current FT, but this was just a simplified calculation using it as a tool for a better estimate of RCIC FTR to be used in a change set. The intent was to eliminate the potential overestimation of risk or skewing of risk using the 0.175 FTR probabilities for RCIC and HPCI.

A change set within the SPAR model was created with the modified values for HPCI and RCIC fail to run and the initiating event for loss of condenser heat removal was set to 1.0:

- Revised RCI-TDP-FR-TRAIN and HCI-TDP-FR-TRAIN to 6E-2
- IE-LOCHS frequency was set to 1.0 as MSIVs closed due to PD

The standard HPCI and RCIC failure to run basic event of 0.175 for the 24-hour mission time will be used in the sensitivity analyses section, with the 6E-2 used as the best estimate SDP assumption.

P1 Fault Tree (One stuck open SRV)

The SRA noted that the P1 fault tree had been turned off for this loss-of-condenser-heat-sink (IE-LOCHS) event. The basic event for PPR-SRV-OO-1VLV, SRV fails to re-close was 'AND' gated with a House station blackout (SBO) event. Discussions with research risk personnel and

Idaho National Labs contractors resulted in determining that this was accurate when running conditional assessments for equipment failures using all of the SPAR postulated events. This is due to there being an inadvertent opening of an SRV in a postulated event. The concern was double counting and so it was 'AND' gated with the SBO event for other events. However, if an evaluation is only being done for an event like a LOCHS, it should be activated to ensure no risk contributions are missed. Therefore, the SRA deleted the 'AND' gate and SBO House event and changed to a simple OR gate.

The SRA developed post processing rules to recognize that if a stuck open SRV occurred in a cutset that:

- The failure to depressurize should use the same value as the inadvertent stuck open event or ADS-XHE-XM-MDEPR2.
- The HPCI and RCIC mission times should be changed to a much lower value (6 hours) as a stuck open SRV will reduce steam pressure which is the motive force for operation. This effectively reduces the FTR value of HPCI and RCIC to avoid conservatism.

Joint Human Error Probabilities (JHEPs)
As stated in the assumption section for an SDP evaluation the direction is to use 1E-6 as the JHEP minimum cutoff value. This assumes some level of dependence exists between human failure events.

There were two notable cutsets that contained actions relative to CHR and late injection.

- JHEP1 basic event was created through the use of a post processing rule for the failure to refill the CST, the probabilistic failure of recovering the power conversion system, and the probabilistic failure of placing torus cooling in service. The rule set the minimum cutoff value at 1E-6.
- JHEP2 was created using the post processing rules for the failure to vent containment, failure to recover the power conversion system and failure to place torus cooling in service. The rule set the minimum cutoff value at 1E-6.

The above values will be adjusted in the sensitivity analyses section to be consistent with one of the lower bounds that Constellation's PRA model of record used for JHEPs, or 5E-7.

FLEX Mitigating Strategy
FLEX credit did not have a significant impact on this evaluation. FLEX credit was invoked by setting the basic event, FLX-XHE-XE-ELAP from 1.0 to 5E-2. The FLEX-480 fault tree was revised to modify credit recognizing procedures and timing by setting FLX-XHE-XM-4802 to TRUE. FLEX-DG1 was credited with basic events modified to increase the failure to run and start terms consistent with a recent PWROG-18043-P Rev 1, analyses of failure data.

SDP Results
SPAR model version 8.80 for Peach Bottom Unit 2 was used for this SDP evaluation.
ICCDP= CCDP – CDP
CCDP= 6E-6 – the dominant cutsets consist of the initiating event LOCHS, with failures of HPCI and RCIC and failure to depressurize. There are various combinations of fail to run, HPCI injection valve failures, failure to start along with the failure to depressurize within the cutsets.
CDP = 2.6E-7 – the dominant cutsets consisted of similar events and sequences
ICCDP= 5.84E-6 or of low to moderate safety significance

Sensitivity Evaluations:

Specific review for evaluating sensitivity values for failure to depressurize

As discussed in the influential assumption section, a key assumption is the basic event, ADS-XHE-XM-MDEPR, Operator fails to initiate reactor depressurization. In accordance with RASP Handbook Volume I guidance, Section 9.2 the SPAR-H method will be used to quantify this HEP. The SRA used only the Action activity with performance shaping factors (PSFs) of extra time considering cutsets with a fail to run within them, with high/experience training, with high stress considering both HPCI/RCIC equipment would have unexpectantly failed with all other PSFs nominal. The result was a 1E-4 HEP. This will be used in a few of the following sensitivity reviews.

2nd review of failure to depressurize using an independent method from SPAR-H, IDHEAS-ECA

In accordance with RASP guidance the SRA used Integrated Human Event Analysis System for Event and Condition Assessment (IDHEAS-ECA) for another sensitivity review. NUREG-2198, The General Methodology of An Integrated Human Event Analysis System, IDHEAS-G was referenced with IDHEAS-ECA version 1.1 software used. The five IDHEAS-ECA macro cognitive functions (MCFs) have base performance influence factors (PIFs) (e.g., scenario familiarity, task complexity, etc.) that are required to be evaluated. Other PIFs can be evaluated if they are applicable within the context of the human failure event. The applicable PIFs for the four evaluated MCFs for this HFE (failure to depressurize) are:

Detection

- Scenario Familiarity-no impact
- Task Complexity-no impact
- Other PIFs not evaluated/selected

Understanding

- Scenario Familiarity-no impact
- Information Completeness and Reliability-no impact
- Task Complexity-no impact
- Other PIFs not evaluated/selected

Decision Making

- Scenario Familiarity-no impact
- Information Completeness and Reliability-this would depend on the situation as in the cutsets equipment is failed and there may be decisions which have to be made on if short term actions can recover the equipment, but information will be coming from the field- this was determined to not have an impact due to the uncertainty of the different potential scenarios
- Task Complexity-no impact
- Other PIFs not evaluated/selected

Action

- Scenario familiarity-no impact

Inter Team was determined to not apply. The time available versus time required was determined to have no impact on the results.

The IDHEAS-ECA result was 2.2E-3, however the SRA recognized this did not account for recovery. The SRA applied a recovery factor of 10 to each of the cognitive MCAs with the result being a HEP of 3.1E-4. A recovery factor of 10 was then applied to Action, with the final result, cognitive and action resulting in a HEP of 2.2E-4. This IDHEAS-ECA as with any HEP has uncertainty but was bounded by the lower SPAR-H sensitivity of 1E-4.

Lastly, the SRA maximized the full recovery credit of 20 within the IDHEAS model for all of the MCFs, including detection, understanding, decision making and action. With recovery fully maximized to 20 in all MCFs, the final HEP value was 1.1E-4. Given that added stress was not applied within the PSFs, this may be a non-conservative value and a more appropriate value would be the 2.2E-4. Notwithstanding this, the SRA used the lower failure probability value of 1E-4 when applying the applicable sensitivities to the SPAR value for failure to depressurize.

Sensitivity 1
Assume HPCI and RCIC failure to run rates for 24-hour mission time unadjusted to credit CRD for early success. Use the nominal 5E-4 SPAR-H value for ADS-XHE-XM-MDEPR and use the RASP guidance for minimum JHEPs 1 and 2 at 1E-6.
ICCDP= CCDP-CDP
ICCDP= CCDP 2.32E-5 – CDP 9.7E-7 =2.2E-5
The SRA noted that using the full 24-hour mission time without some adjustment in the HPCI and RCIC failure to run would result in what is believed to be a conservative result. CRD maximization may still be an option with recognized dependency.

Sensitivity 2
Assume HPCI and RCIC failure to run rates for 24-hour mission time unadjusted to credit CRD. Use the sensitivity value calculated above of 1E-4 for failure to depressurize, ADS-XHE-XM-MDEPR for any cutset containing a FTR event and use JHEPs 1 and 2 at 1E-6.
ICCDP= CCDP 6.6E-6 – CDP 2.7E-7 = 6.3E-6
Dominant cutsets are the LOCHS event with various failures of HPCI or RCIC and failure to depressurize.

Sensitivity 3
Assume minimal lower bound values. HPCI and RCIC failure to run set to the SDP estimate of 6E-2, failure to depressurize set to 1E-4 for any cutset containing a FTR event, and JHEP1 and JHEP2 set to 5E-7, consistent with many of the minimum cutoff JHEP values used within Constellation's model or record.
ICCDP= CCDP 2.63E-6 – CDP 1.1E-7 = 2.52E-6
Dominant cutsets include the LOCHS event with failures of HPCI and RCIC and one cutset of LOCHS with various operator failures such as failing to place torus cooling in service, failing to makeup to the CST and failing to recover PCS.

- For Sensitivity 3 setting ADS-XHE-XM-MDEPR to a minimum of 1E-4 for all cutsets (i.e., not only cutsets with a FTR event) resulted in an ICCDP of CCDP 2.34E-6 – CDP 9.7E-8 = 2.24E-6

The SDP best estimate value and the sensitivities indicate the best estimate value would suggest this PD is of low to moderate safety significance. The SRA determined Sensitivity 1 was not representative of a best estimate with skewed FTR impacts.

Contributions from External Events:
The risk associated with this issue is associated with an actual loss of the normal condenser heat sink via the closure of the MSIVs and inability to use normal high pressure reactor feedwater pumps and the main turbine bypass valves for heat removal. External events are not applicable in this SDP.

Potential Risk Contribution from LERF:
The SRA reviewed portions of the Peach Bottom's PRA summary notebook, PB-PRA-013, revision 2 relative to the analysis of large early release frequency (LERF). The evaluation incorporates a Level 2 methodology analyzing issues such as magnitude and timing of calculated radionuclide releases through level 2 containment event trees. The SRA noted that the licensee had used a LERF multiplier for class IA events (e.g., MSIV closure-HPI failure) CDF sequences of a nominal 1E-1. Class 1A events are loss of injection with reactor at high pressure. Other class events also have factors less than 1.0. Therefore, the SRA determined that this evaluation does not increase the conditional large early release probability (CLERP) importance with respect to risk over or beyond that calculated for the ICCDP increase.

PB-PRA-013, Revision 6, 2018 PRA Table 3.4-2A indicated a conditional large early release probability (CLERP) of 1E-6 for an MSIV closure event. The SRA noted that was based on a CCDP of 1E-5 for the MSIV closure event and the Licensee has since performed Application Specific Models to reflect insights not recognized in the existing model of record. These reviews support a lower modified CCDP for the MSIV closure event due to conservatisms within the model being revisited. This would support a lower CLERP value in the E-7 range for an MSIV closure event.

IMC 0609, Appendix H, Table 5.2, applies LERF factors of 1.0 and 0.6 for high pressure core damage accident sequences with the drywell dry or flooded, respectively. These Appendix H LERF factors are considered conservative bounding values. More recent insights from an NRC Office of Research sponsored study by Energy Research, Inc. (ERI/NRC-03-04), November 2003 indicates that without reactor coolant system (RCS) injection during a station blackout (SBO), there is a high probability that the RCS would subsequently depressurize as a result of either temperature-induced creep rupture of the steam lines or a stuck-open safety relief valve (SRV) (due to high temperature cycling). Subsequent State of the Art Reactor Consequence Analysis Project at Peach Bottom Nuclear Power Station (NUREG/CR-7110) have identified that improved modeling and analysis of anticipated types and sizes of reactor coolant ruptures, projected containment heating and fuel-coolant interactions, and operator actions taken to flood containment in accordance with Severe Accident Management Guidelines, significantly reduce the potential for containment breach and the likelihood of a LERF.

The SPAR model dominant core damage sequences would not significantly contribute to LERF risk due to timing considerations. These sequences involve a failure of high-pressure coolant injection/reactor core isolation cooling and a failure to depressurize the RCS, resulting in the failure of any injection to the RCS (i.e., no low-pressure injection from core spray or low-pressure coolant injection systems). These sequences are similar to the accident conditions that would be encountered during an SBO event without high pressure coolant injection availability.

Therefore, the above reports indicate a more benign containment response at the time of vessel breach, in terms of direct containment heating and fuel-coolant interaction-induced containment failure. As a result of the above considerations, the use of a LERF multiple based on a depressurized RCS and a flooded DW floor would be appropriate. The SRA concluded that the risk due to ΔLERP is consistent and bounded by the ICCDP results (i.e., White, of low to moderate safety significance).

Licensee's Risk Evaluation and Technical Analysis:
Constellation modeled this event as a transient, with subsequent failure of MSIVs, and not a loss of condenser heat sink event. They performed an SDP evaluation, PB-SDP-001, revision 0,

Peach Bottom 2022 RPS Power Supply Loss Calculations in Support of PRA Significance Determination. Constellation stated “in most cases, their PRA model of record is the best choice to assess risk significance. But, for this scenario, there are several potential conservative issues associated with their model of record that could affect the results.” The SDP stated that if the items were modeled in detail this could cause their model to match the SPAR model results more closely. The SRA did note that there were a very large number of JHEP combinations appearing in Constellation’s MOR core damage cutsets as their model is more detailed than the SPAR model. The SRA noted that this could suggest if more key basic events were added to the SPAR model there is a potential of an increase in the SPAR calculated risk for this event. Constellation made various changes within an application specific model to represent this event, including modeling changes to significantly reduce HEPs for failure to depressurize. Their application specific model results came out to an SDP minus Base case of 4.7E-6 for this event. However, Constellation believes their ASM results have multiple conservatisms and are not reflective of the relative risk significance of the event. Notwithstanding this, the SRA noted this was very similar to the final best estimate NRC performed SPAR model calculation.

Constellation stated the SPAR model would be more accurate, as long as recommended changes were applied. Constellations SPAR results for the SDP case, was summarized in Table 5-1, of PB-SDP-001, Summary of SPAR model results, with an SDP risk increase conclusion of 4.64E-7 or of very low safety significance. Constellation’s review determined that the SPAR failure to depressurize event should be decreased by a factor of 25, for an MSIV closure event with initial failure to operate of HPCI and RCIC. Constellation suggested this revised value of 2E-5 for the failure to depressurize HEP events should be used in the NRC SPAR model vice the existing 5E-4 basic event value. Additionally, for sequences where HPCI or RCIC would fail to run at any time, even after 35 minutes, the failure rate was reduced due to consideration of some amount of run time of these systems. For any HPCI or RCIC fail to run basic event in a cutset, Constellation used a post processing rule to decrease the failure to depressurize events from 5E-4 to 3E-6, or a factor of 150 reduction in the existing base SPAR HEP value. These adjustments were applied when they ran the SPAR model for this event and contributed to the development of the 4.64E-7 ICCDP determination. The SRA noted that the SPAR model does not contain two separate events for depressurizing, as you either fail to depressurize to lower pressure systems or succeed. Therefore, Constellation took revised elements (i.e., separate depressurization basic events) out of their own model and applied them to the SPAR model with respect to the failure to depressurize HEP. The SRA did note the SPAR model does credit condensate injection when depressurization is successful. The SRA determined that this suggested change for the SPAR model to invoke for this SDP would be a significant change in historical methods of evaluating LOCHS events and would be inconsistent with the principles of scrutability and repeatability outlined within the IMC 0308 bases document.

Preliminary Significance Determination

The NRC’s preliminary quantitative risk assessment concluded the ICCDP to be 5.84E-6, or of low-to-moderate safety significance (White). The dominant core damage sequences were an actual loss-of-condenser-heat sink event with failure of high-pressure injection and failure to depressurize. The SRA concluded that the risk should be based on the ICCDP results and that the CLERP would not increase the overall significance of the finding. Sensitivity and uncertainty analyses show a very high confidence in this quantitative risk estimate.

References:

Constellation PRA Application Notebook, PB-SDP-001, Revision 0, Peach Bottom 2022 RPS Power Supply Loss Calculations in Support of PRA SDP

Peach Bottom Unit 2 SPAR Model Version 8.80

The risk assessment of operational events handbook (RASP) manual, volume I – Internal Events, revision 2.02

Inspection Manual Chapter 0308, Attachment 3, Technical Basis for Significance Determination Process, Section 0308.03-08.02, Treatment of Degraded Conditions and Initiating Events defines how to treat these events

NUREG-2198, The General Methodology of An Integrated Human Event Analysis System, IDHEAS-G

PB-PRA-004, Peach Bottom PRA Human Reliability Analysis Notebook Volume I

PB-PRA-013, revision 6, 2018, Summary Notebook PB218A2 and PB318A2

Issue Report, 4500178, RPS Scram