UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
2100 RENAISSANCE BOULEVARD, SUITE 100
KING OF PRUSSIA, PENNSYLVANIA 19406-2713

February 10, 2020

Mr. Eric Carr
President and Chief Nuclear Officer
PSEG Nuclear, LLC
PO Box 236
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION: INTEGRATED INSPECTION REPORT 05000354/2019004

Dear Mr. Carr:

On December 31, 2019, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Hope Creek Generating Station. On January 22, 2020, the NRC inspectors discussed the results of this inspection with Mr. Steve Poorman, Plant Manager, and other members of your staff. The results of this inspection are documented in the enclosed report.

Two findings of very low safety significance (Green) are documented in this report. One of these findings involved a violation of NRC requirements. We are treating this violation as a non-cited violation (NCV) consistent with Section 2.3.2 of the Enforcement Policy.

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

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 Hope Creek Generating Station.

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

Sincerely,

/RA/

Brice A. Bickett, Chief
Reactor Projects Branch 3
Division of Reactor Projects

Docket No. 05000354
License No. NPF-57

Enclosure:
Inspection Report 05000354/2019004

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U.S. NUCLEAR REGULATORY COMMISSION
Inspection Report

Docket Number:	05000354
License Number:	NPF-57
Report Number:	05000354/2019004
Enterprise Identifier:	I-2019-004-0044
Licensee:	PSEG Nuclear, LLC
Facility:	Hope Creek Generating Station
Location:	Hancock's Bridge, NJ 08038
Inspection Dates:	October 01, 2019 to December 31, 2019
Inspectors:	A. Ziedonis, Senior Resident Inspector J. Patel, Resident Inspector H. Anagnostopoulos, Senior Health Physicist E. Andrews, Health Physicist J. Kulp, Senior Reactor Inspector R. Rolph, Senior Health Physicist S. Shaffer, Senior Health Physicist
Approved By:	Brice A. Bickett, Chief Reactor Projects Branch 3 Division of Reactor Projects

Enclosure

SUMMARY

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

List of Findings and Violations

Inadequate Evaluation of Emergency Diesel Generator Preventive Maintenance Change Request			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green FIN 05000354/2019004-01 Open/Closed	[P.5] - Operating Experience	71152
The inspectors determined there was a self-revealing, Green Finding because PSEG did not adequately evaluate a preventive maintenance (PM) change to the 'C' emergency diesel generator (EDG) JW pump seal, in accordance with MA-AA-716-210, "Preventive Maintenance Program," Revision 10. Specifically, PSEG did not adequately review the PM basis for the 'C' EDG JW pump seal prior to cancelling the PM. Consequently, on September 4, 2019, the JW pump seal failed due to age-related embrittlement during a surveillance run and rendered the 'C' EDG inoperable.			

Failure to Control Transient Combustible Lube Oil In Accordance With Fire Protection Program Procedure			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Initiating Events	Green NCV 05000354/2019004-02 Open/Closed	[H.12] - Avoid Complacency	71152
The inspectors identified a finding of very low safety significance (Green) and associated non-cited violation (NCV) of Hope Creek Generating Station (HCGS) Technical Specification (TS), Section 6.8, "Procedures and Programs," because PSEG did not implement requirements in the Fire Protection Program (FPP) procedure. Specifically, on September 26, 2019, the inspectors identified five non-permitted, 55-gallon drums of lube oil stored in the reactor building, in the proximity of the end-of-cycle recirculation pump trip (EOC-RPT) breaker cabinet. This was contrary to FP-AA-011, "Control of Transient Combustible Material," Revision 6, section 4.1, "Transient Combustible Control General Requirements," and exceeded transient combustible load limits established in the procedure.			

Additional Tracking Items

None.

PLANT STATUS

The Hope Creek Generating Station (Hope Creek) began the inspection period at approximately 77 percent rated thermal power (RTP) in an end-of-cycle coastdown period. On October 16, Hope Creek commenced a downpower and a manual shutdown to conduct a maintenance and refueling outage (H1R22). Following completion of maintenance and refueling, Hope Creek commenced a reactor startup on November 20 and achieved 100 percent RTP on November 24. On December 14, Hope Creek commenced a downpower to approximately 71 percent for a rod pattern adjustment, and returned to 100 percent RTP later the same day. On December 15, Hope Creek commenced a downpower to approximately 75 percent for a rod pattern adjustment, and returned to 100 percent RTP on December 16. The station remained at approximately 100 percent 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 plant status activities described in IMC 2515, Appendix D, "Plant Status," and conducted routine reviews using IP 71152, "Problem Identification and Resolution." The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel to assess PSEG's 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 03.02) (1 Sample)

(1) The inspectors evaluated readiness for seasonal extreme weather conditions prior to the onset of seasonal cold temperatures for the following systems on December 27: station service water, station auxiliary cooling system, and emergency diesel generators.

External Flooding Sample (IP Section 03.04) (1 Sample)

(1) The inspectors evaluated readiness to cope with external flooding in the emergency diesel generator rooms and service water intake structure on December 16.

71111.04Q - Equipment Alignment

Partial Walkdown Sample (IP Section 03.01) (4 Samples)

The inspectors evaluated system configurations during partial walkdowns of the following systems/trains:

(1) Core spray system during torus refill in Mode 4 on November 19
(2) HPCI system with elevated turbine casing temperature on December 20
(3) RCIC system with elevated turbine casing temperature on December 20
(4) 'B' service water following planned maintenance on December 27

71111.05Q - Fire Protection

Quarterly Inspection (IP Section 03.01) (6 Samples)

The inspectors evaluated fire protection program implementation in the following selected areas:

(1) Emergency diesel generator fuel oil storage tank rooms on October 2
(2) Reactor building personnel hatch and equipment room on October 9
(3) Cable spreading room on October 10
(4) High pressure coolant injection pump and turbine room on December 6
(5) High pressure coolant injection electrical equipment room on December 6
(6) Reactor core isolation cooling pump and turbine room on December 6

71111.06 - Flood Protection Measures

Inspection Activities - Internal Flooding (IP Section 02.02a.) (1 Sample)

The inspectors evaluated internal flooding mitigation protections in the:

(1) Reactor building motor control center 10B242 area, and control rod drive pump area on December 20.

71111.07A - Heat Sink Performance

Annual Review (IP Section 02.01) (1 Sample)

The inspectors evaluated readiness and performance of:

(1) the A1 safety auxiliaries cooling system heat exchanger on October 22

71111.08G - Inservice Inspection Activities (BWR)

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

(1) The inspectors verified that the reactor coolant system boundary, reactor vessel internals, risk-significant piping system boundaries, and containment boundary are appropriately monitored for degradation and that repairs and replacements were appropriately fabricated, examined and accepted by reviewing the following activities during the Hope Creek 1R22 refueling outage from October 21, 2019, to October 25, 2019:

03.01.a - Nondestructive Examination and Welding Activities.
1. Manual Ultrasonic Examination of Reactor Vessel Nozzle to Shell Weld N2A(H1-100195)

2. Automated Phased Array Ultrasonic Examination of Nozzle to Safe-end Dissimilar Metal Weld RPV1-N2HSE (H1-100680)
3. Enhanced Visual (EVT-1) Examination of Top Guide Grid Beam (180-9304918-000) as part of Industry Initiative BWRVIP-183
4. Manual Phased Array Ultrasonic Examination of Nozzle to Safe-end Dissimilar Metal Weld RPV1-N2ESE (H1-100215)
5. Ultrasonic Thickness Examination of Containment Air Gap (H1-820007) involving a related License Renewal commitment
6. Dye Penetrant Test of 1-inch Socket Weld HC-1-P-BB-274-45 (WO 60140862)
7. Welding Activity: 1 inch, Class 1, Reactor Instrument Line to Penetration J51, Welds HC-1-P-BB-274-45 and HC-1-P-BB-274-46. (WO 60140862)

71111.11A - Licensed Operator Requalification Program and Licensed Operator Performance

Requalification Examination Results (IP Section 03.03) (1 Sample)

(1) The inspectors reviewed and evaluated the licensed operator examination results for the requalification annual operating exam on December 3.

71111.11Q - Licensed Operator Requalification Program and Licensed Operator Performance

Licensed Operator Performance in the Actual Plant/Main Control Room (IP Section 03.01) (1 Sample)

(1) The inspectors observed and evaluated licensed operator performance in the Control Room during plant shutdown for a planned refueling outage on October 16, and during reactor and plant startup on November 20.

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

(1) The inspectors observed and evaluated a crew of licensed operators in the plant's simulator during licensed operator training that involved a neutron monitoring instrument failure, a tornado event that resulted in loss of a non-safety related 4kV bus, thermal hydraulic instabilities, and an anticipated transient without scram scenario on December 5.

71111.13 - Maintenance Risk Assessments and Emergent Work Control

Risk Assessment and Management Sample (IP Section 03.01) (4 Samples)

The inspectors evaluated the risk assessments for the following planned and emergent work activities:

(1) Emergent repairs to the 'H' safety relief valve discharge t-quencher piping on October 26
(2) Emergent testing in response to elevated control rod friction on November 26
(3) Emergent repairs to the RCIC drain pot level control valve for a packing steam leak on December 5
(4) Emergent repairs to the FLEX diesel generator on December 17

71111.15 - Operability Determinations and Functionality Assessments

Operability Determination or Functionality Assessment (IP Section 02.02) (3 Samples)

The inspectors evaluated the following operability determinations and functionality assessments:

(1) Safety relief valve functionality and local leak rate testing on October 18
(2) Residual heat removal 'B' low pressure coolant injection valve following breaker troubleshooting on November 19
(3) 'D' emergency diesel generator emergency load sequencer loss of power timer card on December 6

71111.18 - Plant Modifications

Temporary Modifications and/or Permanent Modifications (IP Section 03.01 and/or 03.02) (2 Samples)

The inspectors evaluated the following temporary or permanent modifications:

(1) Main steam safety relief valve design change and installation during refueling outage H1R22
(2) Equivalency evaluation for 'H' safety relief valve t-quencher pipe section replacements on November 11

71111.19 - Post-Maintenance Testing

Post-Maintenance Test Sample (IP Section 03.01) (6 Samples)

The inspectors evaluated the following post maintenance tests:

(1) 'H' safety relief valve t-quencher weld non-destructive examination following repairs on November 18 (Work Order 60144641).
(2) High pressure coolant injection system following planned maintenance on November 21
(3) New fuel conditioning during start up on November 22
(4) Control rod scram timing following accumulator replacement on November 25
(5) High pressure coolant injection system steam admission line pressure inspection following valve replacement on December 3
(6) Reactor coolant injection system steam admission line pressure inspection following valve replacement on December 3

71111.20 - Refueling and Other Outage Activities

Refueling/Other Outage Sample (IP Section 03.01) (1 Sample)

(1) The inspectors evaluated refueling outage H1R22 activities from October 16 to November 21.

71111.22 - Surveillance Testing

The inspectors evaluated the following surveillance tests:

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

(1) HC.OP-ST.05-0002, high pressure coolant injection functional test (low pressure) on November 22
(2) HC.IC-SC.BF-0001, scram discharge volume high water level test on December 2
(3) HC.RE-ST.BF-0001, control rod scram time surveillance on December 6

Containment Isolation Valve Testing (IP Section 03.01) (1 Sample)

(1) HC.OP-LR.AB-0001/2/3/4, main steam isolation valve local leak rate testing (as-found) on October 20

FLEX Testing (IP Section 03.02) (1 Sample)

(1) HC.OP-PT.GS-0001, torus hardened vent remote shutdown panel testing on October 28

71114.06 - Drill Evaluation

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

The inspectors evaluated:

(1) The inspectors evaluated the operators in the control room simulator during a full participation emergency preparedness drill on October 2.

RADIATION SAFETY

71124.01 - Radiological Hazard Assessment and Exposure Controls

Radiological Hazard Assessment (IP Section 02.01) (1 Sample)

The inspectors evaluated radiological hazards assessments and controls.

(1) The inspectors reviewed the following:

Radiological Surveys
- Reactor Building select areas
- 'A' Reactor Water Clean-up Pump Room
- 'B' Reactor Water Clean-up Pump Room

Risk Significant Radiological Work Activities
- Initial Drywell Entry
- Refuel Floor Reactor Disassembly Activities

Air Sample Survey Records

- Observed AMS-4's operating in the Reactor Building

Instructions to Workers (IP Section 02.02) (1 Sample)

The inspectors evaluated instructions to workers including radiation work permits used to access high radiation areas.

(1) The inspectors reviewed the following:

Radiation Work Packages

- RWP-8
- RWP-10
- RWP-12
- RWP-17
- RWP-20

Electronic Alarming Dosimeter Alarms

- No alarms occurred during the period of this inspection

Labeling of Containers

- Observed labeling of over 6 bags in the Radwaste Building

Contamination and Radioactive Material Control (IP Section 02.03) (1 Sample)

The inspectors evaluated PSEG's processes for monitoring and controlling contamination and radioactive material.

(1) The inspectors observed the controls in place for the Reactor Cavity during disassembly activities.

Radiological Hazards Control and Work Coverage (IP Section 02.04) (1 Sample)

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

(1) The inspectors also reviewed the following radiological work package for areas with airborne radioactivity:
    - No work packages were available for review during this inspection

High Radiation Area and Very High Radiation Area Controls (IP Section 02.05) (1 Sample)

(1) The inspectors evaluated risk-significant high radiation area and very high radiation area controls.

Radiation Worker Performance and Radiation Protection Technician Proficiency (IP Section 02.06) (1 Sample)

(1) The inspectors evaluated radiation worker performance and radiation protection technician proficiency.

71124.02 - Occupational ALARA Planning and Controls

Radiological Work Planning (IP Section 02.01) (1 Sample)

The inspectors evaluated the PSEG's radiological work planning.

(1) The inspectors reviewed the following activities:
- RWP-8 refuel floor activities
- RWP-10 maintenance support activities
- RWP-12 in-service inspection activities
- RWP-17 safety relief valve work
- RWP-20 maintenance balance of plant

Verification of Dose Estimates and Exposure Tracking Systems (IP Section 02.02) (1 Sample)

The inspectors evaluated dose estimates and exposure tracking.

(1) The inspectors reviewed the following as low as reasonably achievable planning documents:
- ALARA plan 27
- ALARA plan 31
- ALARA plan 36
- ALARA plan 39

Implementation of ALARA and Radiological Work Controls (IP Section 02.03) (1 Sample)

The inspectors reviewed as low as reasonably achievable practices and radiological work controls.

(1) The inspectors reviewed the work in progress and post job reviews for following activities:
- RWP-8 refuel floor activities
- RWP-10 maintenance Support activities
- RWP-12 in-service inspection activities
- RWP-17 safety relief valve work
- RWP-20 maintenance balance of plant

Radiation Worker Performance (IP Section 02.04) (1 Sample)

The inspectors evaluated radiation worker and radiation protection technician performance during:

(1) Reactor building refuel floor reactor disassembly activities and initial drywell entry

71124.04 - Occupational Dose Assessment

Source Term Categorization (IP Section 02.01) (1 Sample)

(1) The inspectors verified PSEG characterized the radiation types and energies that were monitored, and verified PSEG developed appropriate scaling factors for hard-to-detect radionuclide activity and alpha radionuclides in internal dose assessments.

External Dosimetry (IP Section 02.02) (1 Sample)

(1) The inspectors evaluated external dosimetry to include the following:
- Verification of National Voluntary Laboratory Accreditation Program accreditation for dosimetry processor
- Onsite storage of passive dosimeters
- Determine if bias and correlations for electronic alarming dosimeter are appropriate

Internal Dosimetry (IP Section 02.03) (1 Sample)

The inspectors evaluated the internal dosimetry program implementation.

(1) The inspectors evaluated the internal dosimetry program to include routine bioassay, special bioassay, and dose assessments.

Special Dosimetric Situations (IP Section 02.04) (1 Sample)

The inspectors evaluated the following special dosimetric situation:

(1) The inspectors evaluated special dosimetric situations to include declared pregnant workers, dosimeter placement and assessment, shallow dose equivalent, neutron dose assessment, and dose of legal record.

71124.05 - Radiation Monitoring Instrumentation

Walk Downs and Observations (IP Section 02.01) (1 Sample)

(1) The inspectors evaluated radiation monitoring instrumentation during plant walkdowns to include the following:
- Portable survey instruments
- Radiation area monitors and continuous air monitors
- Personnel contamination monitors, portal monitors and small article monitors

Calibration and Testing Program (IP Section 02.02) (1 Sample)

(1) The inspectors evaluated PSEG's calibration and testing program. The inspectors specifically assessed the following instruments and equipment:
- Laboratory instrumentation
- Whole body counter
- Post-accident monitoring instrumentation

- Portal monitors, personnel contamination monitors, and small article monitors
- Portable survey instruments, area radiation monitors, and air samplers/continuous air monitors
- Instrument calibrator
- Calibration and check sources
- Electronic alarming dosimeters

71124.06 - Radioactive Gaseous and Liquid Effluent Treatment

Walk Downs and Observations (IP Section 02.01) (1 Sample)

(1) The inspectors walked down the gaseous and liquid radioactive effluent monitoring and filtered ventilation systems to assess the material condition and verify proper alignment according to plant design.

Calibration and Testing Program (Process & Effluent Monitors) (IP Section 02.02) (1 Sample)

(1) The inspectors evaluated the gaseous and liquid effluent monitor instrument calibration and testing.

Sampling and Analysis (IP Section 02.03) (1 Sample)

(1) The inspectors reviewed:
- Radioactive effluent sampling activities
- Representative sampling requirements
- Compensatory measures taken during effluent discharges with inoperable effluent radiation monitoring instrumentation
- The use of compensatory radioactive effluent sampling
- The results of the inter-laboratory and intra-laboratory comparison program, including scaling of hard-to-detect isotopes

Instrumentation and Equipment (IP Section 02.04) (1 Sample)

(1) The inspectors reviewed radioactive effluent discharge system surveillance test results and reviewed the methodology used to determine the radioactive effluent stack and vent flow rates based on Technical Specifications/Off Site Dose Calculation Manual acceptance criteria.

Dose Calculations (IP Section 02.05) (1 Sample)

(1) The inspectors reviewed several liquid and gaseous discharge permits to evaluate public dose calculations (monthly, quarterly, and annual) and the annual radiological effluent release reports for 2017 and 2018.

71124.08 - Radioactive Solid Waste Processing and Radioactive Material Handling, Storage, and Transportation

Radioactive Material Storage (IP Section 02.01) (1 Sample)

The inspectors evaluated radioactive material storage.

(1) The inspectors observed radioactive waste container storage areas and verified the postings and controls and that PSEG had established a process for monitoring the impact of long-term storage of the waste.

Radioactive Waste System Walkdown (IP Section 02.02) (1 Sample)

The inspectors evaluated the following radioactive waste processing systems [and processes] during plant walkdowns:

(1) The inspectors walked down the following:
- Accessible portions of liquid and solid radioactive waste processing systems to verify current system alignment and material condition
- Abandoned in place radioactive waste processing equipment to review the controls in place to ensure protection of personnel
- Changes made to the radioactive waste processing systems since the last inspection
- Processes for mixing and transferring radioactive waste resin and/or sludge discharges into shipping/disposal containers
- Current methods and procedures for dewatering waste

Waste Characterization and Classification (IP Section 02.03) (1 Sample)

The inspectors evaluated the radioactive waste characterization and classification for the following waste streams:

(1) The inspectors identified radioactive waste streams and reviewed radiochemical sample analysis results to support radioactive waste characterization. The inspectors reviewed the use of scaling factors and calculations to account for difficult-to-measure radionuclides.

Shipment Preparation (IP Section 02.04) (1 Sample)

The inspectors evaluated [and observed] the following radioactive material shipment preparation processes:

(1) The inspectors reviewed the records of shipment packaging, surveying, labeling, marking, placarding, vehicle checks, emergency instructions, disposal manifest, shipping papers provided to the driver, and PSEG's verification of shipment readiness.

Shipping Records (IP Section 02.05) (1 Sample)

The inspectors evaluated the following non-excepted package shipment records:

(1) The inspectors reviewed selected non-excepted package shipment records.

OTHER ACTIVITIES – BASELINE

71151 - Performance Indicator Verification

The inspectors verified PSEG's performance indicators submittals listed below:

OR01: Occupational Exposure Control Effectiveness Sample (IP Section 02.15) (1 Sample)

(1) Occupational Exposure Control Effectiveness for the period from December 2018 through August 2019

PR01: Radiological Effluent Technical Specifications/Offsite Dose Calculation Manual Radiological Effluent Occurrences (RETS/ODCM) Radiological Effluent Occurrences Sample (IP Section 02.16) (1 Sample)

(1) The inspectors reviewed PSEG's submittals for the radiological effluent TS/ODCM radiological effluent occurrence PI for the fourth quarter 2018 through the third quarter of 2019.

71152 - Problem Identification and Resolution

Semiannual Trend Review (IP Section 02.02) (1 Sample)

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

Annual Follow-up of Selected Issues (IP Section 02.03) (3 Samples)

The inspectors reviewed PSEG's implementation of its corrective action program related to the following issues:

(1) Verification of PSEG's 'H' safety relief valve t-quencher piping corrective actions following identification of through-wall flaws on November 4
(2) Weaknesses identified in PSEG's control of transient combustible materials on November 19
(3) Review of evaluation and corrective actions to address a 'C' emergency diesel generator jacket water pump seal failure on November 22

INSPECTION RESULTS

Inadequate Evaluation of EDG Preventive Maintenance Change Request			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green FIN 05000354/2019004-01 Open/Closed	[P.5] - Operating Experience	71152
The inspectors determined there was a self-revealing, Green Finding because PSEG did not adequately evaluate a PM change to the 'C' EDG JW pump seal, in accordance with MA-AA-716-210, "Preventive Maintenance Program," Revision 10. Specifically, PSEG did not adequately review the PM basis for the 'C' EDG JW pump seal prior to cancelling the			

PM. Consequently, on September 4, 2019, the JW pump seal failed due to age-related embrittlement during a surveillance run and rendered the 'C' EDG inoperable.

Description: Hope Creek utilizes four EDGs to serve as the standby electric power source in case both normal and alternate off-site power supplies to the safety-related emergency 4.16 kV buses are lost. These EDGs can supply all safety-related emergency loads that are required to safely shutdown the reactor, maintain the plant in a safe shutdown condition, and mitigate the consequences of an accident. Each of the EDGs is equipped with a safety-related JW cooling subsystem. The JW system removes excess heat from the engine when running and maintains the EDG sufficiently heated in a standby condition to ensure reliable starting. The JW consists of demineralized water with chemicals additives for corrosion inhibiting (Sodium Nitrite) and anti-freezing (Ethylene Glycol). The JW is circulated by an engine shaft-driven centrifugal pump with a single mechanical seal. The mechanical seal has a stationary hard seal face made of 316 stainless steel, and a rotating sacrificial seal face made of resin impregnated carbon with a nitrile elastomer bellows in the rotating seal assembly.

At 9:56 a.m. on September 4, 2019, 'C' EDG was placed in service for a 24-hour surveillance run in accordance with TS Surveillance Requirement 4.8.1.1.2k.1. Approximately 5 hours into the test, the main control room received an overhead alarm for the 'C' EDG indicating "jacket water filling." In response to the alarm, a nuclear equipment operator observed leakage from the JW pump tell-tail and estimated the leakage at approximately 0.5 gallons per minute (gpm) using local collection and quantification. Immediately, operators in the main control room reduced load on the EDG. At 1515 hours, EDG was declared inoperable based on the leakage exceeding the operability limit of 54 ml/min (0.014 gpm), as defined in HC.OP-ST.KJ-0016, 'C' EDG 24 Hour Operability Run and Hot Restart Test, Revision 36, limitation step 3.2.13. PSEG entered this issue into their corrective action program under notification (NOTF) 20832182, performed a prompt investigation, replaced the mechanical seal to restore EDG to an operable condition, and assigned an apparent cause evaluation (ACE) in the corrective action program (CAP). The prompt investigation documented that most likely cause of the tell-tail leakage was a seal bellows failure, and concluded that failure analysis of the failed seal would be necessary to determine the exact cause.

The inspectors reviewed PSEG's ACE 70209242, which determined that the direct cause of the JW seal failure was age-related brittleness of the nitrile elastomer seal bellow, and the apparent cause was attributed to an inappropriate PM strategy (i.e., "PM") change request (PCR) 80115269-1391. The ACE determined that the failed seal had been installed for approximately six-and-a-half years. Failure analysis determined the bellows material was identified as nitrile rubber, with a temperature-based service life of approximately 4 years at the JW system stand-by temperature range of 150-155° F. The failure analysis report concluded that the degradation observed in the bellows was a time-dependent and temperature-dependent aging mechanism, and as such, recommended establishing a preventive maintenance (PM) replacement strategy.

The inspectors reviewed PCR 80115269-1391, approved on August 1, 2016, to understand the basis for the PM revision, which focused on changing the PM from a time-based frequency to a condition-based maintenance (CBM) strategy. The PCR stated that prior to July 2016, there was a time-based PM to replace the JW and IC pump mechanical seals every 4 years. The PCR stated that under the proposed CBM strategy, conditions of seal leakage would be identified by system engineering or operations during periodic walkdowns and observations of surveillance runs. Furthermore, it stated that such examples would be entered into the CAP in a timely manner and seals would be replaced during the next system

window prior to seal failure. The inspectors noted that as an outcome of this PCR, PSEG changed the 'D' EDG JW and IC pump seal PM to “on-condition,” however, PSEG deactivated the seal PMs for 'A', 'B', and 'C' EDGs, instead of modifying them to reflect CBM.

The inspectors noted the PCR discussed the basis for the 4-year PM strategy was due to previous age-related failures of JW pump seals, as discussed in 2010 ACE 70109413-0030, Revision 1, and 2001 ACE 70020296-0010. The inspectors noted that both 2010 and 2001 ACEs determined the cause of the seals failures was attributed to age-related embrittlement, and further noted that age-related failures had occurred without prior leakage history. However, the 2016 PCR did not evaluate whether CBM was appropriate mechanism to prevent sudden failures due to age-related embrittlement, where no previous evidence of seal leakage had been identified.

On July 8 and August 5 of 2016, PSEG identified a repetitive small leak on the 'C' EDG JW pumps seals during monthly surveillance runs. PSEG documented the leakage under NOTF 20735199, and assigned the NOTF to PM work order 30250778 to replace the JW and IC pump seals during the 4-year PM in March of 2017. However, on August 10, 2016, the PM work order 30250778 was deactivated, following approval of PCR 80115269-1391 on August 1, 2010 (as discussed above). Subsequently, PSEG reassigned the NOTF to corrective maintenance (CM) work order 60130195, initially with the same due date of March 2017. However, the due date to implement this CM work order was extended multiple times prior the failure of the JW pump seal on September 4, 2019. PSEG discussed the 2016 leakage in 2019 ACE 70209242, and determined that the small leakage from 2016 did not re-occur after August 2016, and also determined that the embrittled elastomer sudden failure was a different leakage mechanism than the minor leakage in 2016.

The inspectors reviewed MA-AA-716-210, “Preventive Maintenance Program,” Revision 10. Step 4.4.3 states that the purpose of the PM change process is to ensure proposed changes are accurately evaluated. In addition, MA-AA-716-210-1005, “Predefine Change Processing,” Revision 7, step 4.2.1, requires PCRs for PMs containing critical components (e.g., components associated with MSPI systems such as the EDGs) are processed in accordance with Attachment 5, which requires a description of the current PM basis in step 4. Therefore, the inspectors determined that PSEG was not in compliance with MA-AA-716-210, step 4.4.3, because PCR 80115269-1391 did not adequately evaluate the PM basis for the ‘C’ EDG JW pump seal prior to converting to CBM and de-activating the PM for 'C' EDG.

Corrective Actions: PSEG wrote NOTF 20832182 on September 4, 2019, to capture the 'C' EDG JW pump seal leakage, declared 'C' EDG inoperable, replaced the seal assembly, performed a failure analysis and a CAP evaluation, and replaced the seals on the other EDGs as an extent of condition.

Corrective Action References: 20832182, 70209242, 20833732, 20832065, 20832383, 20832393

Performance Assessment:

Performance Deficiency: The inspectors determined that not adequately evaluating a PM change to the 'C' EDG JW pump seal, in accordance with MA-AA-716-201, was a performance deficiency that was reasonably within PSEG's ability to foresee and should have been prevented. Specifically, PCR 80115269-1391 did not adequately evaluate the PM basis

for the 'C' EDG JW pump seal prior to converting to CBM and de-activating the PM for 'C' EDG.

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, not adequately evaluating a PM change prior to cancellation resulted in a seal failure that rendered the 'C' EDG inoperable, and required unavailability of the EDG to replace the failed seal and restore EDG operability.

Significance: The inspectors assessed the significance of the finding using Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." The inspectors assessed the significance of the finding using Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." The inspectors determined that the finding represented an actual loss of function of a single train ('C' EDG) for greater than its allowed TS outage time, and therefore required a detail risk evaluation (DRE).

The senior reactor analyst (SRA) used the Systems Analysis Programs for Hands-On Evaluation (SAPHIRE), Revision 8.2.1, Standardized Plant Analysis Risk (SPAR) Model, version 8.59 to perform the DRE. The SRA contacted Idaho National Labs (INL) to obtain an updated model (limited version use) which incorporated the use of the FLEX (Mitigation Strategies for Beyond Design Basis External Events) strategy to ensure consistency with the as-built plant and procedures. The SRA used the guidance in the risk assessment standardization project (RASP) handbook Volume I, Revision 2.02, Section 2.5, to determine an exposure time of a nominal 56 days, including repair time. This was based on evaluation of the proven 'C' EDG runtime prior to the degraded condition, the expected bounding pump seal leak rate, and the automatic jacket water (JW) makeup capability. For the dominant core damage scenarios, the demineralized water pumps would not have power, however there would be an estimated minimum of at least 160 gallons of water in the piping above the EDG JW expansion tank available. The design is such that a solenoid operated fill valve (powered from the EDG class 1E inverter) would cycle open and closed based on level, to make-up for the leak for many hours before operator action would be required. The SRA noted that PSEG had a Station Blackout/Loss of Offsite Power/Diesel Generator Malfunction procedure, "HC.OP-AB.ZZ-0135(Q)," which was unique in that it had a section within the procedure for actions to take to align firewater for JW leaks greater than 54 milliliters (ml/minute). The SRA noted that given an assumed leak-rate of 0.5 GPM and the make-up water volume available, the 'C' EDG would likely have run up to 11 or more hours before any operator action would have been required to align diesel driven firewater for water addition to the JW expansion tank. The SRA reviewed the internal operating experience of JW leaks at the station from these pump seals over the last 20 years and noted that there had not been a leak-rate which exceeded this one, as they were all noted to be of the same magnitude or less. The exposure time estimate factored in the expected minimum EDG run time and an 11 hour 'C' EDG proven run time 53 days earlier during testing, to determine the 24 hour mission time would have been satisfied. Therefore, a best estimate exposure time of 53 days plus repair time was used.

The SRA noted this is a bounding assumption given there is uncertainty with when this failure may have impacted the ability of the machine to perform its 24 hour mission time. The JW system is maintained at elevated temperatures in the standby condition resulting in some continuous degradation of the elastomer material. The SRA walked down the recovery

procedure in the plant and determined that it was not complex and easily performed. Additionally, there would be numerous available cues/alarms with very clear procedure guidance for JW makeup to the tank from either the demineralizer water system static head or the diesel driven firewater system. Notwithstanding this, the SRA modeled this as a failure of the 'C' EDG to run because of the eventual need for operator action to respond to the leak. The SRA noted that PSEG had performed a human error probability (HEP) recovery analysis determining an extremely low failure probability (nominal 5E-4) for the required JW makeup action.

Although recovery actions would be available, the SRA bounded the risk by assuming zero recovery for the internal event risk evaluation along with a 'C' EDG failure to run event. The following adjustments or assumptions were made for the SPAR model internal event risk review:

- Given the run time of the EDG, ACP-XHE-XL-INVCOOL, “operators fail to provide alternate cooling” was adjusted to 2E-3 to account for extra time available for inverter cooling;
- The Reactor Core Isolation Cooling (RCIC) and High Pressure Coolant Injection (HPCI) room steam leak failure probabilities were decreased an order of magnitude to 0.01 consistent with the lower failure probabilities in the Hope Creek model;
- Plant procedures perform containment venting using the hardened vent system upgrades and extra time with high stress was assumed in SPAR-H, adjusting CVS-XHE-XM-VENTLT to 6E-3, this was not a dominant driver of risk;
- The FLEX event tree was invoked (credited) by revising FLX-XHE-ELAP from TRUE to 5E-2 for the failure to declare an ELAP within 1 hour;
- The FLEX event tree was revised to fail the second FLEX generator and rely on only one FLEX generator due to timing considerations with deployment hookup and battery life;
- FLEX equipment failure rates were increased for a bounding assessment;
- The ‘C’ EDG failure to run was set to TRUE with no recovery (very conservative);
- Although this failure was recoverable the basic event was not set to its nonrecovery probability consistent with section 3.2 of the RASP, which would result in a bounding risk estimate because full common cause failure (CCF) would be invoked;
- The EDG CCF for all four diesels was conditionally increased by a factor of 40 (2E-3) from its base case failure probability (7E-5) due to setting the failure to TRUE;
- For the conditional case (failure of the EDG to run), the station blackout event trees were adjusted to reflect up to 10 hours for offsite power recovery in place of the 4 hour recovery (to account for the extended run time before degradation was noted). This was consistent with the base case model when there would be success of only the 'C' EDG. The SRA noted this had very little effect on the calculation of increased risk;
- For the conditional case, the offsite power 30 minute sequences were changed to 4 hours to recognize the availability of the 'C' EDG (i.e. low pressure core spray and low pressure coolant injection) and safety relief valves to provide adequate core coverage for events where both RCIC and HPCI injection fails;
- The HPCI and RCIC failure to run mission times and failure rates were adjusted to 6 hours for sequences of station blackout (SBO) with a failed open safety relief valve (SRV).

The base case and condition case cutsets were run and gathered to a core damage end-state. The increase in core damage frequency (CDF) per year given the 56 day exposure was calculated to be 7.5E-8/yr for internal events. The dominant cutsets were weather

related LOOP events, with failure of the Salem Unit 3 gas turbine, common cause failure of all four EDGs, failure to recover an EDG, failure to recover offsite power along with a convolution factor, and a failed open SRV. Other dominant cutsets included weather related LOOPs, failure of the Salem Unit 3 gas turbine, common cause failure of all four EDGs, failure to recover offsite power and failure of FLEX equipment.

The SRA noted that PSEG performed a thorough internal and external event risk review with several sensitivities. PSEG performed the risk evaluation using the assumption of a recoverable degraded failure, setting the EDG failure to run event to its nonrecoverable probability with various sensitivities for HEPs (E-4 to E-1). The CCF adjustment was then made using this higher failure rate (i.e. E-2 recovery sensitivity HEP added to nominal failure probability of EDG) in accordance with Table 5-1 Category 4 of the RASP manual. It is noted this would not invoke a full CCF adjustment such as setting the failure to run as TRUE. The total risk increase using this method was on the order of 2E-7/yr given the credit for recovery and treatment as a degraded condition. PSEG also performed the evaluation setting the failure to run as TRUE, invoking full common cause adjustment in accordance with section 3.5 of the RASP manual for modeling a support system failure. This was consistent with how the SRA performed the internal events analysis using the SPAR model. The SRA considered this to be appropriate and a bounding case for all the reasons noted above. PSEG considered this to be very conservative because the EDG did not fail and was easily recoverable. There was a small recovery factor applied consistent with the NRC RASP manual guidance. The increase in risk was dominated by external event or fire risk contribution. The total increase in CDF/yr for the entire exposure time was calculated to be 8E-7/yr using the full effect of CCF increase. The dominant core damage cutsets consisted of various postulated fire area events, which resulted in conditional LOOPs, with common cause failure of all four EDGs and failure to provide core cooling using alternate external low-pressure injection. The SRA noted that the range of PSEG's total calculated increase in risk (internal and external risk) using the two methods was between 2E-7 and 8E-7/yr. The SRA reviewed PSEG's risk evaluation and determined it to be reasonable and noted that the calculated internal events risk increase was very close to the SPAR model output. The SRA noted the conservatisms in the above analysis, including the exposure time assumption and determined that the impact on the increase to large early release frequency (LERF) would not change the conclusion of a very low safety significance issue (Green) for the increase in CDF/yr due to the degradation associated with the 'C' EDG.

Cross-Cutting Aspect: P.5 - Operating Experience: The organization systematically and effectively collects, evaluates, and implements relevant internal and external operating experience in a timely manner. Specifically, PSEG did not effectively evaluate internal age-related seal failure history, as documented in previous station CAP evaluations, prior to cancelling seal PM activities.

Enforcement: Inspectors did not identify a violation of regulatory requirements associated with this finding.

Failure to Control Transient Combustible Lube Oil In Accordance With Fire Protection Program Procedure			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Initiating Events	Green NCV 05000354/2019004-02 Open/Closed	[H.12] - Avoid Complacency	71152

The inspectors identified a finding of very low safety significance (Green) and associated non-cited violation (NCV) of Hope Creek Generating Station (HCGS) TS, Section 6.8, "Procedures and Programs," because PSEG did not implement requirements in the Fire Protection Program (FPP) procedure. Specifically, on September 26, 2019, the inspectors identified five non-permitted, 55-gallon drums of lube oil stored in the reactor building, in the proximity of the end-of-cycle recirculation pump trip (EOC-RPT) breaker cabinet. This was contrary to FP-AA-011, "Control of Transient Combustible Material," Revision 6, Section 4.1, "Transient Combustible Control General Requirements," and exceeded transient combustible load limits established in the procedure.

Description: On September 26, 2019, during the walkdown of fire area RB1, room 4331, reactor building, the inspectors identified five 55-gallon drums of lube oil unattended, and without a transient combustible permit (TCP) staged in proximity of the EOC-RPT breaker cabinet, which was contrary to requirements in procedure FP-AA-011. Section 4.1, "Transient Combustible Control General Requirements," step 4.1.3, states that "a TCP is required prior to staging any flammable liquid or combustible liquid inside a Safety Related or Critical building." HCGS's reactor building is designated as a Critical building in HCGS FPP documents. Upon further review, the inspectors noted that procedure FP-AA-011 establishes a transient combustible load limits of 4,480,000 BTU per room in any area of the plant. FP-AA-011, Attachment 6, provides estimated heat content of common transient combustibles. For flammable liquid, the estimated heat content is 90,000 BTU per gallon. The inspectors calculated that five 55-gallon drums of lube oil equate to 24,750,000 BTU of heat load, which exceeded the combustible load limits established in the transient combustible control procedure. Additionally, the inspectors noted that the procedure allows transient combustible load limits to be exceeded with prior Engineering approval. In this case, Engineering had not reviewed and approved the storage of these combustible loads in the reactor building. The inspectors noted that neither the site fire marshal nor fire protection supervisor had approved these transient combustibles. The inspectors made operations aware, on the day of identification, of the oil drums staged without an approved TCP, and PSEG removed the oil drums later the same day.

The inspectors also identified transient combustible materials stored in the transient combustible free zone. On September 11, 2019, during a walkdown of fire area IS3, service water traveling water screen upper room, the inspectors identified cables, plastic buckets, wood, cardboard boxes, rags and insulation materials, and other transient combustible materials left unattended in the area without a TCP. For all of the additional examples of transient combustible materials, PSEG immediately removed the materials and entered the issue in their corrective action program.

Corrective Actions: Immediately following identification by the NRC, PSEG promptly removed the lube oil from the area, and entered the issue in their corrective action program.

Corrective Action References: 20834837

Performance Assessment:

Performance Deficiency: The inspectors determined that PSEG's failure to control transient combustible materials in the reactor building in accordance with procedure FP-AA-011 was a performance deficiency that was reasonably within the PSEG's ability to foresee and should have been prevented. Specifically, on September 26, 2019, the inspectors identified transient combustible materials left unattended and without a TCP in room 4331 of the reactor building, in proximity to the EOC-RPT breaker cabinet.

Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the Protection Against External Factors 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. This finding is also similar to example described in IMC 0612, "Power Reactor Inspection Reports," Appendix E, "Examples of Minor Issues," Section 4k. Specifically, HCGS did not account for the amount of transient combustibles present in the area and the amount of combustible loading in the fire area exceeded the maximum load limit allowed by the FPP procedure.

Significance: The inspectors assessed the significance of the finding using Appendix F, "Fire Protection and Post - Fire Safe Shutdown SDP." Using Appendix F, Attachment 1, "Fire Protection SDP Phase 1 Worksheet," the inspectors assigned the category to fire prevention and administrative controls. The inspectors determined that the safety significance of the finding was very low because based on the SDP qualitative screening question (step 1.4.1) related to fire prevention and administrative controls, the finding does not increase the likelihood of a fire, delay detection of a fire, or result in a more significant fire than previously analyzed such that the credited safe shutdown strategy was adversely impacted.

Cross-Cutting Aspect: H.12 - Avoid Complacency: Individuals recognize and plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful outcomes. Individuals implement appropriate error reduction tools. Specifically, individuals involved with staging equipment associated with outage work activities did not consider undesired consequences of their actions to leave the oil unattended without an approved TCP.

Enforcement:

Violation: HCGS TS, Section 6.8.1, states, in part, written procedures shall be established, implemented, and maintained for implementing the Fire Protection Program. HCGS's FPP implementing procedure CC-AA-211, "Fire Protection Program," Revision 5, states that operational aspects of the FPP are controlled by implementing the procedure FP-AA-011, "Control of Transient Combustible Materials." Section 4.1, "Transient Combustible Control General Requirements," step 4.1.3, of procedure FP-AA-011 requires a TCP for staging any flammable liquid or combustible liquid inside a Safety Related or Critical building. Attachment 4 of procedure FP-AA-011 establishes a combustible load limit of 4,480,000 BTU per room in any area of the plant and allows exceeding this limit with prior Engineering approval.

Contrary to the above, on September 26, 2019, PSEG failed to implement FPP requirements as stated in FP-AA-011. Specifically, PSEG stored five 55-gallon drums of lube oil in a critical building, unattended and without a TCP. This exceeded the combustible loading limits of 4,480,000 BTU, and prior Engineering approval was not obtained.

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

Observation: Semiannual Trend Observations	71152
The inspectors evaluated a sample of condition reports generated over the course of the past two quarters, and determined that, in general, PSEG was appropriately identifying and resolving trends. Most notably, PSEG identified a negative trend associated with human performance errors. Some of the specific examples include: · On November 27, an instrument and control technician error resulted in an unplanned half scram condition (20840287); · On November 19, an operations department status control error resulted in an unplanned trip of the 'A' control rod drive pump during start-up preparations from a refueling outage (20839590); · On October 11, an electrical maintenance technician bumped a breaker and therefore removed power to the 'C' core spray room cooler (20835563); · On October 10, a chemistry department status control event resulted in an overflow condition of a sampling sink spilling water on the floor of the plant (20835463); · On September 27, an instrument and control technician error resulted in an unplanned half scram condition (20835030); and · On September 15, a chemistry department status control event resulted in an inadvertent isolation of hypo-chlorination to the 'C' service water train (20835463). In response, PSEG addressed each of the issues individually, including the performance of prompt investigations, human performance review boards, and human performance stand-downs. In response to the collective negative trend, PSEG documented NOTF 20840930, and performed a Human Performance Investigation, which generated multiple assigned actions across several departments. The inspectors determined that none of the above issues were of greater than minor safety significance, in accordance with IMC 0612, Appendix B. The inspectors determined that PSEG's actions in response to the human performance issues were appropriate to the circumstances.	

EXIT MEETINGS AND DEBRIEFS

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

- On January 22, 2020, the inspectors presented the integrated inspection results to Mr. Steve Poorman, Plant Manager and other members of PSEG's staff.
- On October 18, 2019, the inspectors presented a debrief of the Radiological Safety Inspection inspection results to Mr. Ed Casulli, Site Vice President and other members of PSEG's staff.

DOCUMENTS REVIEWED

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
71124.01	Calibration Records	20834515	Required radioactive material storage inspections	
	Corrective Action Documents	20818215	New High Radiation Area	
		20818745	Radiation Protection Underwater Instrument Failure	
		20834934	Failure to complete follow-up actions	
	Procedures	RP-AA-301	Radiological Air Sampling Program	8
		RP-AA-376	Radiological Posting, Labeling, and Markings	10
		RP-AA-503	Unconditional Release Survey Method	8
	Radiation Surveys	HC-M-201904403	Reactor Water Clean-up Pump Room	03/09/2019
		HC-M-20190710-13	'A' Reactor Water Clean-up Pump Room	07/10/2019
		HC-M-20190724-14	201' Reactor Building	07/03/2019
		HC-M-20190903-8	201' Reactor Building	09/03/2019
		HC-M-20191016-27	100' Drywell	10/16/2019
		HC-M-20191016-28	87' Drywell	10/16/2019
		HC-M-20191017-4	100' Drywell	10/17/2019
		HC-M-20191017-5	144' Drywell	10/17/2019
		HC-M-20191017-6	153' Drywell	10/17/2019
		HC-M-20191017-7	87' Drywell Under Vessel	10/16/2019
		HC-M-20191017-8	87' Drywell	10/17/2019
		HC-M-2019117-9	87' Drywell Subpile	10/17/2019
71124.02	Corrective Action Documents	20829254	Self Reading Dosimeter Dose Rate Alarm	

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
	Procedures	RP-AA-400	ALARA Program	7
		RP-AA-401	Occupational ALARA Planning and Controls	14
		RP-HC-400-1004	Emergent Dose Control and Authorization	2