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

February 11, 2014

EA-14-021

Mr. Thomas P. Joyce
President and Chief Nuclear Officer
PSEG Nuclear LLC - N09
P.O. Box 236
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION UNIT 1 – NRC INTEGRATED INSPECTION REPORT AND EXERCISE OF ENFORCEMENT DISCRETION 05000354/2013005

Dear Mr. Joyce:

On December 31, 2013, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Hope Creek Generating Station (HCGS). The enclosed inspection report documents the inspection results, which were discussed on January 16, 2014, with Mr. P. Davison, Site Vice President of Hope Creek, and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents two NRC-identified and two self-revealing findings of very low safety significance (Green). Two of these findings were determined to involve a violation of NRC requirements. Additionally, a licensee-identified violation, which was determined to be of very low safety significance, is listed in this report. However, because of the very low safety significance, and because they are entered into your corrective action program (CAP), the NRC is treating the findings as non-cited violations (NCVs) consistent with Section 2.3.2 of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident

Inspector at HCGS. In addition, if you disagree with the cross-cutting aspect assigned any of the findings 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 Regional Administrator, Region I, and the NRC Resident Inspector at Hope Creek Generating Station.

In addition to the issues identified above, a violation involving a failure to set secondary containment during operations with the potential to drain the reactor vessel (OPDRVs) was identified during the Hope Creek refueling outage. Specifically, on October 15, October 20, and October 23, 2013, while all other Technical Specifications were met, Public Service Enterprise Group (PSEG) conducted several OPDRVs without establishing secondary containment operability, which is a violation of Technical Specification (TS) 3.6.5.1, "Secondary Containment." NRC issued EGM 11- 003, "Enforcement Guidance Memorandum on Dispositioning Boiling Water Reactor (BWR) Licensee Noncompliance with TS Containment Requirements During Operations with a Potential for Draining the Reactor Vessel," on October 4, 2011, allowing for the exercise of enforcement discretion for such OPDRV-related TS violations, when certain criteria are met. The EGM, which was revised on December 20, 2012, also requires that, to be eligible for discretion, a licensee must submit a license amendment request (LAR) to accept the NRC's generic change to the Standard Technical Specifications (STS) that will allow a graded approach to OPDRV requirements. The LAR must be submitted within four months of NRC publication of the STS in the Federal Register. The EGM, which was revised additionally on December 13, 2013, extends the time period of enforcement discretion to December 31, 2015, to permit refueling outage planning while the NRC staff and the Boiling Water Reactor Owners Group (BWROG) finalize a generic solution for TS changes and allows up to 12 months to submit a TS change.

The NRC concluded that, for the specified periods, PSEG met the EGM criteria and has committed to submit the LAR, as required. Therefore, I have been authorized, after consultation with the Director, Office of Enforcement, and the Regional Administrator, to exercise enforcement discretion and refrain from issuing enforcement for the violation, subject to a timely LAR being submitted.

As a result of the Safety Culture Common Language Initiative, the terminology and coding of cross-cutting aspects were revised beginning in calendar year (CY) 2014. New cross-cutting aspects identified in CY 2014 will be coded under the latest revision to IMC 0310. Cross-cutting aspects identified in the last six months of 2013 using the previous terminology will be converted to the latest revision in accordance with the cross-reference in IMC 0310. The revised cross-cutting aspects will be evaluated for cross-cutting themes and potential substantive cross-cutting issues in accordance with IMC 0305 starting with the CY 2014 mid-cycle assessment review.

In accordance with 10 Code of Federal Regulations 2.390, "Public Inspections, Exemptions, Requests for Withholding," of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's

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Sincerely,

/RA/

Michael L. Scott, Acting Director
Division of Reactor Projects

Docket No.: 50-354
License No.: NPF-57

Enclosure: Inspection Report 05000354/2013005
w/Attachment: Supplementary Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-354
License No.:	NPF-57
Report No.:	05000354/2013005
Licensee:	Public Service Enterprise Group (PSEG) Nuclear LLC
Facility:	Hope Creek Generating Station (HCGS)
Location:	P.O. Box 236 Hancocks Bridge, NJ 08038
Dates:	October 1, 2013, through December 31, 2013
Inspectors:	J. Hawkins, Senior Resident Inspector F. Ramirez, Acting Senior Resident Inspector S. Ibarrola, Resident Inspector R. Barkley, Senior Project Engineer T. Burns, Reactor Inspector M. Draxton, Project Engineer N. Floyd, Reactor Inspector J. Furia, Senior Health Physicist R. Nimitz, Senior Health Physicist T. Hedigan, Operations Engineer J. Laughlin, Emergency Preparedness Inspector, NSIR M. Orr, Reactor Inspector M. Patel, Operations Engineer J. Schoppy, Senior Reactor Inspector
Approved By:	Glenn T. Dentel, Chief Reactor Projects Branch 3 Division of Reactor Projects

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SUMMARY

IR 05000354/2013005; 10/01/2013 – 12/31/2013; Hope Creek Generating Station; Operability Determinations and Functionality Assessments, Refueling and Other Outage Activities, Problem Identification and Resolution, and Follow-Up of Events and Notices of Enforcement Discretion.

This report covered a three-month period of inspection by resident inspectors and announced inspections performed by regional inspectors. Two non-cited violations (NCVs), two findings of very low safety significance (Green), and one licensee-identified violation of very low safety significance (Green) were identified. The significance of most findings is indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using Inspection Manual Chapter (IMC) 0609, “Significance Determination Process” (SDP), dated June 2, 2011. All violations of Nuclear Regulatory Commission (NRC) requirements are dispositioned in accordance with the NRC’s Enforcement Policy, dated January 28, 2013. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 4.

Cornerstone: Initiating Events

- Green. A Green self-revealing NCV of TS 6.8.1, “Procedures and Programs,” was identified regarding PSEG’s conduct of maintenance and component configuration control during system restoration from an operation with a potential for draining the reactor vessel (OPDRV) activity. Specifically, PSEG did not close a reactor water cleanup (RWCU) valve in accordance with the maintenance procedure during the refueling outage. This resulted in increased RCS UIL in the reactor drywell area following startup. PSEG restored the mispositioned valves, conducted an extent of condition on other valves in the drywell, completed a prompt investigation concerning the valve mispositioning, and is in the process of conducting an Apparent Cause Evaluation (ACE) on the configuration control event under Order 70161461. PSEG has also placed this issue into CAP as notification 20632003.

  The performance deficiency was more than minor because it was associated with the configuration control 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 evaluated the finding using IMC 0609, Attachment 4, Initial Screening and Characterization of Findings, which required an analysis using Exhibit 1 of IMC 0609, Appendix A,” The Significance Determination Process for Findings At-Power,” dated June 19, 2012. The finding was determined to be of very low safety significance (Green) because the finding could not result in exceeding the RCS leak rate for a small loss of coolant accident (LOCA) or have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. This finding had a cross-cutting aspect in the area of Human Performance, Work Practices, because PSEG’s communication of human error prevention techniques did not support human performance and proper personnel work practices. Specifically, PSEG did not use adequate human performance tools and valve position verification techniques when controlling valve position for components associated with an OPDRV activity. [H.4(a)] (Section 1R15)

- Green. A Green self-revealing finding was identified for PSEG’s failure to identify and correct an adverse trend regarding 48 Bailey module failures across multiple systems since 2005, including six Bailey module failures in the circulating water (CW) system. As a result of continued problems associated with this previously unidentified adverse trend, on June 12, 2013, the ‘B’ CW pump tripped resulting in a manual scram of the reactor due to degrading condenser vacuum. PSEG corrective actions include addressing the programmatic weakness identified regarding the performance monitoring and trending program for circuit card failures by amending the Bailey Module Reliability Program to include fuse module and auxiliary card failures.

The finding was more than minor because it was associated with the Initiating Events cornerstone and affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, PSEG’s failure to identify and correct the adverse trend regarding Bailey module failures resulted in a manual scram from 100 percent power due to the trip of the ‘B’ CW pump concurrent with the ‘B’ CW discharge valve being gagged in the open position. The finding was determined to be of very low safety significance (Green) in accordance with Appendix A of IMC 0609, “The Significance Determination Process (SDP) for Findings At-Power,” dated June 19, 2012, because the finding did not contribute to 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. The inspectors determined that this finding had a cross-cutting aspect in the area of Problem Identification and Resolution, Corrective Action Program, because PSEG did not trend and assess information from the CAP and other assessments in the aggregate to identify programmatic and common cause problems. Specifically, PSEG failed to trend or perform an aggregate assessment of Bailey module and auxiliary card failures. [P.1(b)] (Section 4OA3.2)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance (Green) and associated NCV of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix B, Criterion V, “Instructions, Procedures and Drawings,” for PSEG’s failure to conduct primary containment (drywell) close-out activities in accordance with site procedures. Specifically, during the NRC’s drywell closeout inspection, the inspectors identified several outage-related items that were not removed from the various elevations of the drywell. As a result, PSEG did not properly inspect the drywell in preparation for power operation. PSEG corrective actions included removing the items identified during the NRC drywell closeout inspection and placing the issue in the corrective action program (CAP).

  The performance deficiency was determined to be more than minor because it is associated with the equipment performance attribute of the Mitigating Systems cornerstone, and affected the cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using NRC IMC 0609, Appendix G, “Shutdown Operations Significance Determination Process,” dated February 28, 2005, the finding was determined to be of very low safety significance (Green) because the inspectors qualitatively determined that the finding involved adequate mitigation capability and was not an event that could be characterized as a loss of control. This finding had a cross-cutting aspect in the area of Human Performance, Work Practices, because PSEG did not define and effectively communicate expectations regarding procedural compliance and personnel did not follow procedures. Specifically, PSEG personnel did not ensure that the drywell was ready for power operations as required by site procedures. [H.4(b)] (Section 1R20)

- Green. The inspectors identified a finding of very low safety significance (Green) for PSEG's failure to ensure evaluations addressed identified issues in accordance with PSEG procedure LS-AA-125, "Corrective Action Program." Specifically, PSEG failed to adequately assess the functionality of the containment vent following NRC identification of inadequate maintenance practices for an instrument air check valve (1KBV-300) and that design calculation H-1-KB-MDC-1007, "Backup Pneumatic Supply for 1GSHV-4964 and 1GSHV-11541 Valves," did not account for leakage through the valve. PSEG's corrective actions included installation of a design change to modify instrument air piping to support leak rate testing of 1KBV-300 and addition of 1KBV-300 to its check valve monitoring and preventive maintenance program. PSEG also completed a revision to design calculation H-1-KB-MDC-1007 to credit up to 500 standard cubic centimeter per minute (sccm) of leakage through 1KBV-300.

  This issue was more than minor because it was associated with the design control attribute of the mitigating systems cornerstone, and affected the cornerstone's objective to ensure the availability and reliability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined the finding to be of very low safety significance (Green) in accordance with Exhibit 2 of IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, because: it was not a deficiency affecting the design or qualification of the containment vent; it did not represent a loss of system or function; it did not represent the loss of function for any technical specification (TS) system, train, or component beyond the allowed TS outage time; and it did not represent an actual loss of function of any non TS trains of equipment designated as highly safety-significant in accordance with PSEG's maintenance rule program. The inspectors determined that the finding had a cross cutting aspect in the Human Performance area associated with Resources, because PSEG did not ensure that personnel, equipment, procedures, and other resources are available and adequate to assure nuclear safety, specifically, those necessary for maintaining long term plant safety by maintenance of design margins. Specifically, PSEG did not ensure maintenance of design margin for the containment vent system when concerns were identified regarding its functionality. This included PSEG relying upon operation of the containment vents with hydraulic jacks that have not been operated since 1992 following their installation [H.2(a)]. (Section 4OA2.5)

Other Findings

A finding of very low safety significance that was identified by PSEG was reviewed by the inspectors. Corrective actions taken or planned by PSEG have been entered into PSEG's CAP. This violation and corrective action tracking number are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Hope Creek began the inspection period at full rated thermal power (RTP). Operators commenced a manual shutdown on October 11, 2013, to start Hope Creek's planned 18th refueling outage (H1R18). On November 12, 2013, the Hope Creek was returned to full RTP. On November 22, 2013, Hope Creek conducted a planned down power to 75 percent of RTP for main condenser water box tube leak investigation and a post H1R18 rod pattern adjustment. Hope Creek was returned to 100 percent power on November 24, 2013. On December 1, 2013, the main turbine tripped followed by a reactor scram on high moisture separator level due to normal and emergency level control failures. Following corrective maintenance on the moisture separator level controllers, Hope Creek commenced a reactor startup on December 3, 2013. On December 5, 2013, during power ascension, moisture separator level control tuning was being conducted at 76 percent power and the main turbine tripped again followed by a reactor scram due to high moisture separator level. Following corrective maintenance and completion of a design change to the moisture separator level control system, Hope Creek commenced a reactor startup on December 9, 2013. On December 12, 2013, the unit was returned to full RTP and remained at or near full RTP for the duration of the inspection period except for brief periods to support planned testing and rod pattern adjustments.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01 – 1 sample)

Readiness for Seasonal Extreme Weather Conditions

a. Inspection Scope

The inspectors performed a review of PSEG's readiness for the onset of seasonal low temperatures. The review focused on the service water intake structure ventilation system, fire pump house ventilation system, condensate storage tank, and the emergency diesel generators (EDGs). The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR), technical specifications, control room logs, and the CAP to determine what temperatures or other seasonal weather could challenge these systems, and to ensure PSEG personnel had adequately prepared for these challenges. The inspectors reviewed station procedures, including PSEG's seasonal weather preparation procedure and applicable operating procedures. The inspectors performed walkdowns of the selected systems to ensure station personnel identified issues that could challenge the operability of the systems during cold weather conditions. Documents reviewed for each section of this inspection report are listed in the Attachment.

b. Findings

No findings were identified.

1R04 Equipment Alignment

Partial System Walkdowns (71111.04 – 3 samples)

a. Inspection Scope

The inspectors performed partial walkdowns of the following systems:

- ‘B’ station service water (SSW) loop during maintenance on the ‘C’ SSW discharge valve on October 17, 2013
- ‘A’ residual heat removal (RHR) pump following motor replacement on October 28, 2013
- ‘D’ 1E 4 kilovolt (kV) vital bus following a loss of the bus and subsequent repair and retest of the bus in-feed breakers on December 19, 2013

The inspectors selected these systems based on their risk-significance relative to the reactor safety cornerstones at the time they were inspected. The inspectors reviewed applicable operating procedures, system diagrams, the UFSAR, technical specifications, work orders, condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have impacted system performance of their intended safety functions. The inspectors also performed field walkdowns of accessible portions of the systems to verify system components and support equipment were aligned correctly and were operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no deficiencies. The inspectors also reviewed whether PSEG staff had properly identified equipment issues and entered them into the corrective action program for resolution with the appropriate significance characterization.

b. Findings

No findings were identified.

1R05 Fire Protection

Resident Inspector Quarterly Walkdowns (71111.05Q - 5 samples)

a. Inspection Scope

The inspectors conducted tours of the areas listed below to assess the material condition and operational status of fire protection features. The inspectors verified that PSEG controlled combustible materials and ignition sources were in accordance with administrative procedures. The inspectors verified that fire protection and suppression equipment was available for use as specified in the area pre-fire plan, and passive fire barriers were maintained in good material condition. The inspectors also verified that station personnel implemented compensatory measures for out of service, degraded, or inoperable fire protection equipment, as applicable, were in accordance with procedures.

- FRH-II-471, refuel floor, elevation 201’ on October 18, 2013
- FRH-II-435, steam tunnel, elevation 102’ on October 20, 2013

- FRH-II-435, drywell, elevation 127’ on October 23, 2013
- FRH-II-132, feedwater heater rooms, elevation 102’ on October 29, 2013
- FRH-II-563, ‘A’ and ‘B’ control area heating, ventilation, and air conditioning rooms, elevation 155’ on December 11, 2013

b. Findings

No findings were identified.

1R07 Heat Sink Performance (711111.07A – 2 samples)

a. Inspection Scope

The inspectors reviewed the A1 and A2 safety auxiliaries cooling system (SACS) heat exchangers to determine their readiness and availability to perform their safety functions. The inspectors reviewed the design basis for the components and verified PSEG’s commitments to NRC Generic Letter 89-13. The inspectors reviewed the results of performance tests for the heat exchangers and observed portions of the as-found inspection of the A1 heat exchanger (HX). The inspectors discussed the results of the most recent inspection with engineering staff and reviewed pictures of the as-found and as-left conditions. The inspectors verified that PSEG initiated appropriate corrective actions for identified deficiencies. The inspectors also verified that the number of tubes plugged within the heat exchanger did not exceed the maximum amount allowed.

b. Findings

No findings were identified.

1R08 In-service Inspection (71111.08 - 1 sample)

a. Inspection Scope

From October 21, 2013 to October 25, 2013, the inspectors conducted a review of PSEG implementation of in-service inspection (ISI) program activities for monitoring degradation of the reactor coolant system boundary, risk significant piping and components, and containment systems during the Hope Creek Generating Station Unit 1 18th refueling outage (H1R18). The sample selection for this inspection was based on the inspection procedure objectives and risk priority of those pressure retaining components in systems where degradation would result in a significant increase in risk. The inspectors observed in-process non-destructive examinations (NDE), reviewed documentation, and interviewed PSEG personnel to verify that the NDE activities performed as part of the third interval, second period, of the Hope Creek ISI program were conducted in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, 2001 Edition with Addenda through 2003.

<u>Non-destructive Examination and Welding Activities (IMC Section 02.01)</u>

The inspectors performed direct observation of NDE activities in process and reviewed documentation of non-destructive examinations listed below. Activities included review of ultrasonic testing (UT), magnetic particle testing (MT), liquid penetrant test (PT), and visual testing (VT).

<u>ASME Code Required Examinations</u>

The inspectors performed a documentation review of an automated phased array UT of the weld RPV1-N5ASE in the core spray system. The weld was a dissimilar metal weld between the safe end and core spray nozzle N5A. The inspectors reviewed the test procedure, weld volume coverage, examination report, test results, and noted that 100 percent code coverage was achieved. Four previous relevant indications were identified, evaluated, and found to be acceptable per the ASME code acceptability requirement. The test analysis and results were discussed with the respective Level III examiner. The inspectors verified the UT procedure and examiner were qualified in accordance with ASME Section XI and Performance Demonstrative Initiative requirements.

The inspectors performed a documentation review of a MT of the reactor pressure vessel head-to-flange surface RPV1-W20. The inspectors verified the MT procedure and examiner were appropriately qualified to the requirements of ASME Section XI. The inspectors also reviewed the contents of the examination report to determine that variables were applied and results were recorded and evaluated as specified by the procedure. No recordable indications were identified.

The inspectors performed a documentation review PT of reactor pressure vessel stabilizer brackets just below the N12 nozzles. The brackets are identified as component RPV1-WSB (1-8). The inspectors verified the procedure and examiner were appropriately qualified to the requirements of ASME Section XI. The inspectors reviewed the test parameters used in the test were in accordance with the limitations, precautions and prerequisites specified in the test procedure. The inspectors also reviewed the test data acquired to determine that variables were applied and results were recorded and evaluated as specified by the procedure. No recordable indications were identified.

The inspectors visually examined the condition of the primary containment liner surfaces on the 102' and 127' elevations. Limited portions of the containment surfaces above and below the listed elevations were accessible for examination. The inspectors noted that the liner coating was being well maintained in serviceable condition. The inspectors also performed a document review of the containment VT records, which had no recordable indications.

<u>Other Augmented or Industry Initiative Examinations</u>

The inspectors sampled the remote enhanced VT records of reactor vessel internals as done under water inside the reactor vessel during in-vessel visual inspection (IVVI) activities. The inspection scope included portions of the core spray piping, jet pump bracing components, and vessel cladding. The inspectors reviewed the applicable parts of the IVVI procedure, observation of a sample of digital video records, the analysis process for the observations, and documentation of indications. The inspectors verified that the activities were performed in accordance with applicable examination procedures and industry guidance.

Repair/Replacement Activities Including Welding Activities

The inspectors performed a record review of the replacement activities associated with the third-stage low-pressure 'A' feedwater heater to verify that welding and applicable NDE activities were performed in accordance with ASME code requirements. The inspectors reviewed the weld procedures and welder qualifications. Specifically, the inspectors reviewed the welding documents for the feedwater inlet/outlet nozzle, steam inlet nozzle, and drain outlet nozzle. The replacement work was performed under job number 904823.

Identification and Resolution of Problems (IMC Section 02.05)

The inspectors reviewed a sample of Hope Creek corrective action reports, which identified NDE indications, deficiencies, and other non-conforming conditions since the previous refueling outage and during the current outage. The inspectors verified that non-conforming conditions were properly identified, characterized, evaluated, and that corrective actions were identified and entered into the corrective action program for resolution.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program and Licensed Operator Performance (71111.11Q – 2 samples; 71111.11A – 1 sample)

.1 Quarterly Review of Licensed Operator Requalification Testing and Training

a. Inspection Scope

The inspectors observed licensed operator simulator training on December 19, 2013 that included loss of a 4kV vital bus, a loss of offsite power coincident with an emergency diesel generator malfunction, and a loss of coolant accident with a high pressure coolant injection failure. The inspectors evaluated operator performance during the simulated event and verified completion of critical tasks, risk significant operator actions, including the use of abnormal and emergency operating procedures. The inspectors assessed the clarity and effectiveness of communications, implementation of actions in response to alarms and degrading plant conditions, and the oversight and direction provided by the control room supervisor. The inspectors verified the accuracy and timeliness of the emergency classification made by the shift manager. Additionally, the inspectors assessed the ability of the training staff to identify and document crew performance problems.

b. Findings

No findings were identified.

.2 Quarterly Review of Licensed Operator Performance in the Main Control Room

a. Inspection Scope

The inspectors observed plant shutdown and restart activities for planned refueling outage, H1R18, on October 11, 2013 and November 9, 2013. The inspectors observed reactivity control briefings to verify that the briefings met the criteria specified in OP-AA-101-111-1004 "Operations Standards," Revision 4 and HU-AA-1211, "Pre-Job Briefings," Revision 10. Additionally, the inspectors observed licensed operator performance to verify that procedure use, crew communications, and coordination of activities between work groups similarly met established expectations and standards.

b. Findings

No findings were identified.

.3 Licensed Operator Requalification Program

a. Inspection Scope

On October 1, 2013, one NRC region-based inspector conducted an in-office review of results of licensee-administered annual operating tests for 2013. Comprehensive written exams were administered in the last quarter of 2012. The inspection assessed whether pass rates were consistent with the guidance of NRC Manual Chapter 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process (SDP)." The inspector verified that:

- Individual pass rate on the dynamic simulator test was greater than 80 percent. (Pass rate was 100 percent.)
- Individual pass rate on the job performance measures of the operating exam was greater than 80 percent. (Pass rate was 100 percent.)
- More than 80 percent of the individuals passed all portions of the requalification exam. (Pass rate was 100 percent.)
- Crew pass rate was greater than 80 percent. (Pass rate was 100 percent.)

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12Q – 3 samples)

a. Inspection Scope

The inspectors reviewed the samples listed below to assess the effectiveness of maintenance activities on structure, system, or component (SSC) performance and reliability. The inspectors reviewed corrective action program documents (notifications), maintenance work orders (orders), and maintenance rule basis documents to ensure that PSEG was identifying and properly evaluating performance problems within the scope of the maintenance rule. As applicable, the inspectors verified that the SSC was properly scoped into the maintenance rule in accordance with 10 CFR 50.65 and verified that the (a)(2) performance criteria established by PSEG staff was reasonable; for SSCs classified as (a)(1), the inspectors assessed the adequacy of goals and corrective actions to return these SSCs to (a)(2); and, the inspectors independently verified that appropriate work practices were followed for the SSCs reviewed. Additionally, the inspectors ensured that PSEG staff was identifying and addressing common cause failures that occurred within and across maintenance rule system boundaries.

- High pressure coolant injection (HPCI) steam admission valve, 1FD-HV-F001 failed to open on September 4, 2012 (Order 70143140)
- Emergency and standby lighting system (a)(1) goals and corrective actions (Order 70152885)
- RHR inboard shutdown cooling (SDC) motor operated valve F009 troubleshooting and repair after failing to stroke open (Notification 20632581)

b. Findings

No findings were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 – 5 samples)

a. Inspection Scope

The inspectors reviewed station evaluation and management of plant risk for the maintenance and emergent work activities listed below to verify that PSEG performed the appropriate risk assessments prior to removing equipment for work. The inspectors selected these activities based on potential risk significance relative to the reactor safety cornerstones. As applicable for each activity, the inspectors verified that PSEG personnel performed risk assessments as required by 10 CFR 50.65(a)(4) and that the assessments were accurate and complete. When PSEG performed emergent work, the inspectors verified that operations personnel promptly assessed and managed plant risk. The inspectors reviewed the scope of maintenance work and discussed the results of the assessment with the station’s probabilistic risk analyst to verify plant conditions were consistent with the risk assessment. The inspectors also reviewed the TS requirements and inspected portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met.

- Planned Yellow risk during refueling outage during ‘A’ vital bus work window on October 15, 2013
- Unplanned yellow risk during planned reactor water cleanup system unavailability and emergent ‘B’ RHR inoperability on October 20, 2013
- Emergent maintenance on the ‘C’ EDG during planned maintenance on the ‘B’ EDG on October 28, 2013
- Planned Yellow risk during refueling outage during reactor pressure vessel (RPV) leak test on November 3, 2013
- Planned Yellow risk during approved troubleshooting for increased drywell unidentified leakage on November 25, 2013

b. Findings

No findings were identified.

1R15 <u>Operability Determinations and Functionality Assessments</u> (71111.15 – 6 samples)

a. <u>Inspection Scope</u>

The inspectors reviewed operability determinations for the following degraded or non-conforming conditions:

- High wear particulate in HPCI booster pump oil on May 13, 2013 (Order 70159288)
- 'D' RHR pump alternate decay heat removal operability following an unplanned 'B' RHR pump trip while providing SDC on October 17, 2013 (Notification 20625727)
- Secondary containment operability during an OPDRV on October 31, 2013 (Notification 20631218)
- 'B' SDC operability with B EDG inoperable on November 4, 2013 (Notification 20628456)
- Evaluate lifting of 'M' safety relief valve (SRV) due to transient on December 1, 2013 (Order 80110848)
- HPCI operability with RPV level above Level 8 during post-scram recovery in Mode 3 on December 1, 2013 (Notification 20631904)

The inspectors selected these issues based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the operability determinations to assess whether technical specification operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the technical specifications and UFSAR to PSEG's evaluations to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled by PSEG. The inspectors determined, where appropriate, compliance with assumptions in the evaluations.

b. <u>Findings</u>

1) A Licensee-Identified Violation associated with a condition prohibited by Technical Specifications is documented in Section 4OA7 of this report for secondary containment inoperability during an OPDRV condition October 31, 2013.

2) <u>Introduction</u>: A Green self-revealing NCV of TS 6.8.1, "Procedures and Programs," was identified regarding PSEG's conduct of maintenance and component configuration control during system restoration from an OPDRV activity. Specifically, PSEG did not close a reactor water cleanup (RWCU) valve in accordance with the maintenance procedure during the refueling outage. This resulted in increased RCS UIL in the reactor drywell area following startup.

<u>Description</u>: On December 3, 2013, PSEG was conducting investigations during a forced outage for the cause of increased UIL in the drywell seen since completing the most recent refueling outage in November 2013. During the drywell walk downs, RWCU valves, 1-BG-V167 (V167) and 1-BG-V168 (V168), were found partially open, contributing approximately 0.2 gpm unidentified drywell leakage. These in-series RWCU valves are used to provide a drainage path from the 'A' reactor recirculation pump (RRP)

to the drywell floor drain sump during maintenance. During the most recent refueling outage, PSEG conducted multiple OPDRVs, including the replacement of the ‘A’ RRP seal from 5:00 p.m. on October 15, 2013, through 11:43 a.m. on October 30, 2013 (Reference Section 4OA3 of this report for OPDRV LER reviews). This OPDRV was completed in accordance with PSEG procedure OP-HC-108-102, "Management of Operations with the Potential to Drain the Reactor Vessel."

During the conduct of this OPDRV, in an effort to properly isolate the work area, the ‘A’ RRP suction and discharge isolation valves were tagged closed. Due to seat leakage past the isolation valves, the V167 and V168 valves were throttled open per work control document (WCD) 4338874 as necessary to maintain the pump body partially empty to prevent water from impacting the work area while the pump seal was removed. During restoration from the ‘A’ RRP seal work on October 30, 2013, both V167 and V168 valves were verified closed per WCD 4338874 and in accordance with guidance from PSEG procedure, OP-AA-108-101-1002, “Component Configuration Control.”

During reactor startup on November 9, 2013, UIL in the drywell was approximately 0.12 gpm. This value increased to approximately 0.29 gpm during the week of December 1, 2013. PSEG planned for a forced outage the weekend of December 13, 2013, to investigate the source of the increased UIL in the drywell. On December 1, 2013, the main turbine tripped due to a moisture separator re-heater level control issue, followed by a reactor scram. During the forced shutdown due to the reactor scram, PSEG conducted UIL investigations in the drywell and operators found V167 and V168 leaking by at approximately 0.2 gpm. Operators used mechanical assistance to fully close V167, found to be 11 turns open, and to ensure V168 was fully closed. PSEG determined that V168 was closed because the valve required less than 1/8 of a turn of movement in the closed direction. Fully closing the V167 valve stopped leakage from the drain line.

PSEG determined in notification 20632003, “Prompt Investigation Report for 1-BG-V167 and 1-BG-V168 Found Partially Open,” that the suspected cause of the component mispositioning was inadequate application of human performance tools and valve position verification techniques by the operators in accordance with PSEG procedure HU-AA-101, Human Performance Tools and Verification Practices.

PSEG restored the mispositioned valves, conducted an extent of condition on other valves in the drywell, completed a prompt investigation concerning the valve mispositioning, and is in the process of conducting an Apparent Cause Evaluation (ACE) on the configuration control event under Order 70161461. PSEG has also placed this issue into CAP as notification 20632003.

<u>Analysis</u>: PSEG’s improper conduct of maintenance and component configuration control during the most recent refueling outage for system restoration from an OPDRV activity was a performance deficiency. This performance deficiency adversely affected PSEG’s RCS UIL in the reactor drywell area. The performance deficiency was more than minor because it was associated with the configuration control 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 evaluated the finding using IMC 0609, Attachment 4, ”Initial Screening and Characterization of Findings,” which required an analysis using IMC 0609 Appendix A Exhibit 1, Initiating Events

Screening Questions. The finding was determined to be of very low safety significance (Green) because the finding could not result in exceeding the RCS leak rate for a small LOCA or have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. Specifically, the inspectors determined that because PSEG's observed increasing trend in UIL remained well below TS limits for RCS leakage, the line diameter was 1 inch, and because PSEG planned a forced outage to investigate the source of the increased UIL, the RCS leak rate for a small LOCA would not be exceeded.

This finding had a cross-cutting aspect in the area of Human Performance, Work Practices, because PSEG's communication of human error prevention techniques did not support human performance and proper personnel work practices. Specifically, PSEG did not use adequate human performance tools and valve position verification techniques when controlling valve position for components associated with an OPDRV activity. [H.4(a)]

Enforcement: TS 6.8.1, "Procedures and Programs," requires, in part, that "written procedures shall be established, implemented, and maintained covering the activities referenced below: a. the applicable procedures recommended in Appendix A of Regulatory Guide (RG) 1.33, Revision 2." RG 1.33, Revision 2, Appendix A, Section 9.a, "Procedures for Performing Maintenance," states in part, that "maintenance that can affect performance of safety-related equipment should be properly pre-planned and performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstances." PSEG's WCD (4338874) in accordance with their Component Configuration Control procedure, OP-AA-108-101-1002, directed the closure of valves V167 and V168. Contrary to this, on October 30, 2013, PSEG improperly conducted component configuration control following maintenance during the most recent refueling outage for an OPDRV activity and valve V168 was not properly closed, which adversely affected PSEG's UIL in the reactor drywell area. As part of PSEG's corrective actions, the mispositioned valves have been restored, an extent of condition on other valves in the drywell was conducted, PSEG completed a prompt investigation concerning the valve mispositioning, and is in the process of conducting an Apparent Cause Evaluation (ACE) on the configuration control event under Order 70161461. Because PSEG entered the issue into its CAP as notification 20632003, it is being treated as an NCV, consistent with Section 2.3.2 of the NRC's Enforcement Policy: **NCV 05000354/2013005-01, Failure to Follow Procedure for Configuration Control Adversely Affected Unidentified Leakage in the Drywell.**

1R19 Post-Maintenance Testing (71111.19 – 5 samples)

a. Inspection Scope

The inspectors reviewed the post-maintenance tests for the maintenance activities listed below to verify that procedures and test activities ensured system operability and functional capability. The inspectors reviewed the test procedure to verify that the procedure adequately tested the safety functions that may have been affected by the maintenance activity, that the acceptance criteria in the procedure was consistent with the information in the applicable licensing basis and/or design basis documents, and that the procedure had been properly reviewed and approved. The inspectors also witnessed the test or reviewed test data to verify that the test results adequately demonstrated restoration of the affected safety functions.

- 'A' EDG overhaul and governor replacement on October 25, 2013 (Order 60104674)
- ‘D’, ‘L’, and ‘M’ main steam SRV replacement on October 30, 2013 (Orders 50123107, 50123145, and 50136953)
- 'A' RRP following RRP speed controller troubleshooting and replacement on November 21, 2013 (Order 60114166)
- 'A' moisture separator drain controller replacement on December 3, 2013 (Order 60114285)
- ‘A’ and ‘B’ moisture separator dump valve controller design change and at-power testing from December 11 – 13, 2013 (Order 60114373)

b. Findings

No findings were identified.

1R20 Refueling and Other Outage Activities (71111.20 – 3 samples)

.1 Refueling Outage (H1R18)

a. Inspection Scope

The inspectors reviewed the station’s work schedule and outage risk plan for the Hope Creek’s 18th refueling outage (H1R18), which was conducted October 11 through November 9, 2013. The inspectors reviewed PSEG’s development and implementation of outage plans and schedules to verify that risk, industry experience, previous site-specific problems, and defense-in-depth were considered. During the outage, the inspectors observed portions of the shutdown and cooldown processes and monitored controls associated with the following outage activities:

- Configuration management, including maintenance of defense-in-depth, commensurate with the outage plan for the key safety functions and compliance with the applicable technical specifications when taking equipment out of service
- Implementation of clearance activities and confirmation that tags were properly hung and that equipment was appropriately configured to safely support the associated work or testing
- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication and instrument error accounting
- Status and configuration of electrical systems and switchyard activities to ensure that technical specifications were met
- Monitoring of decay heat removal operations
- Impact of outage work on the ability of the operators to operate the spent fuel pool cooling system
- Reactor water inventory controls, including flow paths, configurations, alternative means for inventory additions, and controls to prevent inventory loss
- Activities that could affect reactivity
- Maintenance of secondary containment as required by technical specifications
- Refueling activities, including fuel handling and fuel receipt inspections
- Fatigue management
- Tracking of startup prerequisites, walkdown of the drywell (primary containment) to verify that debris had not been left which could block the emergency core cooling system suction strainers, and startup and ascension to full power operation
- Identification and resolution of problems related to refueling outage activities

PSEG reported the use of EGM 11-003 in LER 05000354/2013-004-00. This LER and PSEG's use of EGM 11-003 is reviewed and dispositioned in Section 4OA3.

b. Findings

Introduction. The inspectors identified a finding of very low safety significance (Green) and associated NCV of 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for PSEG's failure to conduct primary containment (drywell) close-out activities in accordance with site procedures. Specifically, PSEG did not properly inspect the drywell in preparation for power operation.

Description. On November 7, 2013, Hope Creek was in Mode 4, Cold Shutdown. Temperature in the RCS was 135 degrees Fahrenheit and time to boil was 2.5 hours. Site personnel were finalizing Hope Creek refueling outage H1R18 activities and were in the process of restoring plant systems to service in preparation for power operation. As part of the NRC's outage inspections, and following the licensee's completion of primary containment (drywell) close-out activities, the inspectors conducted an independent inspection of the drywell prior to its closure. The inspection consisted of a thorough walkdown to verify, among other things, that tags were cleared, that there was no evidence of leakage and that there was no debris that might contribute to emergency core cooling systems (ECCS) equipment blockage.

During the drywell closeout inspection, the inspectors identified several outage-related items that were not removed from the various elevations of the drywell. The items included foreign material covers, tools, scaffolding that should have been removed following work, High Radiation Area posting signs and ropes, a hardhat, and loose piping insulation. In addition, the inspectors found a notable amount of trash that had to be removed from the drywell such as tie wraps, an old bump cap, hard hat covers, a number of procedures, and loose papers (mostly consisting of radiation protection documentation). Of particular concern to the inspectors, was the identification of a large plastic tarp that was being used as a water collection system around the 'B' reactor recirculation pump. This plastic tarp was found in the 87 foot elevation of the drywell, which connects with the torus through the down-comers. The torus in turn contains the suction strainers for ECCS equipment.

Licensee procedure HC.OP-GP.ZZ-0002, "Primary Containment Closeout," outlines the steps for inspecting the drywell in preparation for power operation. Attachment 1 to this procedure provides a checklist of items that need to be completed as part of PSEG's drywell close-out activities. This checklist instructs PSEG personnel to verify, among other things, that no debris or trash is in the drywell, and temporary equipment has been removed from the drywell. In addition, the licensee procedure states that drywell closeout should not be permitted with unapproved equipment, material, or unauthorized debris sources unless evaluated by station engineering personnel. The inspectors noted that procedure HC.OP-GP.ZZ-0002 had been completed and signed off by PSEG Operations management prior to the inspectors conducting their drywell closeout inspection. Based on procedure completion, PSEG personnel determined the drywell was ready for power operation. As a result, the items identified during the inspectors' walkdown would have otherwise been left inside the drywell and would have presented the potential for ECCS equipment blockage. PSEG's corrective actions include procedure revisions to support a coordinated drywell closeout to include scheduling of a drywell cleanout work window using a multi-discipline team.

Analysis. The inspectors determined that the failure to perform an adequate drywell close-out and clean-up prior to power operation in accordance with procedure HC.OP-GP.ZZ-0002, "Primary Containment Closeout," was a performance deficiency that was within PSEG's ability to foresee and correct, and should have been prevented. Using the guidance in IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated September 7, 2012, the inspectors determined that the performance deficiency is greater than minor, and therefore a finding, because it is associated with the Equipment Performance attribute of the Mitigating Systems cornerstone, and affected the cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, ECCS systems could have been impacted by the amount of loose debris and trash found in the drywell, especially the plastic tarp that could have been transported to the torus and blocked a portion of the ECCS suction strainers.

Because this finding occurred while the plant was shut down, the inspectors used IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process," dated February 28, 2005. Using Attachment 1, "Phase 1 Operational Checklists for Both Pressurized Water Reactors and Boiling Water Reactors," and specifically Checklist 8, "Boiling Water Reactor Cold Shutdown or Refueling Operation, Time to Boil greater than 2 Hours: RCS Level less than 23 feet," the inspectors qualitatively determined that the finding involved adequate mitigation capability and was not an event that could be characterized as a loss of control. As a result, the inspectors concluded that the finding was of very low safety significance (Green). This finding had a cross-cutting aspect in the area of Human Performance, Work Practices, because PSEG did not define and effectively communicate expectations regarding procedural compliance and personnel did not follow procedures. Specifically, PSEG personnel did not ensure that the drywell was ready for power operations as required by site procedures. [H.4(b)]

Enforcement. 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, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. PSEG procedure HC.OP-GP.ZZ-0002, "Primary Containment Closeout," outlines the steps for inspecting the drywell (primary containment) in preparation for power operation. HC.OP-GP.ZZ-0002 states in part, to verify no debris or trash is in the drywell and to verify no temporary equipment is in the drywell. Contrary to this requirement, on November 7, 2013, PSEG signed off procedure HC.OP-GP.ZZ-0002 as completed without ensuring that the drywell was entirely ready for power operation. As a result, the items identified during the inspectors' walkdown would have been left inside the drywell and would have presented the potential for ECCS equipment blockage. Corrective actions included removing the items identified during the NRC drywell closeout inspection and placing the issue in CAP. Since this issue was entered into PSEG's CAP as notification 20629147, this violation of 10 CFR Part 50, Appendix B, Criterion V, is being treated as an NCV consistent with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000354/2013005-02, Failure to Follow the Primary Containment Closeout Procedure when Declaring the Drywell Ready for Power Operation)**

.2 Forced Outage 132 and 133

a. Inspection Scope

The inspectors reviewed the station's work schedule and outage risk plan for two forced outages (F132 and F133) following turbine trips due to high level in the 'A' moisture separator reheater, which were conducted December 1 through December 9. The inspectors reviewed PSEG's development and implementation of outage plans and schedules to verify that risk, industry experience, previous site-specific problems, and defense-in-depth were considered. During the outage, the inspectors observed portions of the shutdown and cooldown processes and monitored controls associated with the following outage activities:

- Configuration management, including maintenance of defense-in-depth, commensurate with the outage plan for the key safety functions and compliance with the applicable technical specifications when taking equipment out of service
- Implementation of clearance activities and confirmation that tags were properly hung and that equipment was appropriately configured to safely support the associated work or testing
- Status and configuration of electrical systems and switchyard activities to ensure that technical specifications were met
- Monitoring of decay heat removal operations
- Impact of outage work on the ability of the operators to operate the spent fuel pool cooling system
- Reactor water inventory controls, including flow paths, configurations, alternative means for inventory additions, and controls to prevent inventory loss
- Activities that could affect reactivity
- Maintenance of secondary containment as required by technical specifications
- Fatigue management
- Tracking of startup prerequisites and startup and ascension to full power operation
- Identification and resolution of problems related to forced outage activities

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22 – 8 samples)

a. Inspection Scope

The inspectors observed performance of surveillance tests and/or reviewed test data of selected risk-significant SSCs to assess whether test results satisfied technical specifications, the UFSAR, and PSEG procedure requirements. The inspectors verified that test acceptance criteria were clear, tests demonstrated operational readiness and were consistent with design documentation, test instrumentation had current calibrations and the range and accuracy for the application, tests were performed as written, and applicable test prerequisites were satisfied. Upon test completion, the inspectors considered whether the test results supported that equipment was capable of performing the required safety functions. The inspectors reviewed the following surveillance tests:

- HC.OP-ST.KJ-0004, 'D' Emergency Diesel Generator Operability Test on August 19, 2013
- HC.OP-IS.AB-0102, Main Steam System Valves Cold Shutdown In-Service Test (IST) on October 12, 2013 (in-service test)

- HC.OP-LR.FC-0002, Containment Isolation Valve Type C Leak Rate Test 1FCHV-F084 – Reactor Core Isolation Cooling (RCIC) Steamline Vacuum Relief Penetration on October 17, 2013 (isolation valve)
- HC.OP-IS.KJ-0007, Integrated Emergency Diesel Generator 1CG400 Test – 18 months on October 24, 2013
- HC.OP-ST.BH-0002, Standby Liquid Control System 18-Month Flow Test on November 2, 2013
- HC.OP-IS.ZZ-0001, Inservice System Leakage Test of the Reactor Coolant Pressure Boundary on November 3, 2013
- HC.OP-IS.EA-0003, C Service Water Pump – CP502 – Inservice Test on December 21, 2013
- HC.OP-IS.BC-0002, CP202, C Residual Heat Removal Pump In-Service Test on December 23, 2013

b. Findings

No findings were identified.

1EP4 Emergency Action Level and Emergency Plan Changes (IP 71114.04 – 1 Sample)

a. Inspection Scope

The NSIR headquarters staff performed an in-office review of the latest revisions of various Emergency Plan Implementing Procedures (EPIPs) and the Emergency Plan located under ADAMS accession number ML121320593 as listed in the Attachment.

The licensee determined that in accordance with 10 CFR 50.54(q), the changes made in the revisions resulted in no reduction in the effectiveness of the Plan, and that the revised Plan continued to meet the requirements of 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50. The NRC review was not documented in a safety evaluation report and did not constitute approval of licensee-generated changes; therefore, this revision is subject to future inspection. The specific documents reviewed during this inspection are listed in the Attachment.

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstone: Occupational and Public Radiation Safety (OS, PS)

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01 - 1 sample)

a. Inspection Scope

During the period December 16-19, 2013, the inspector reviewed and assessed PSEG's performance in assessing and controlling radiological hazards in the workplace. The review was against criteria contained in 10 CFR Part 20, TSs, applicable Regulatory Guides, and PSEG procedures for determining compliance.

Inspection Planning

The inspector reviewed 2013 performance indicators for the occupational exposure cornerstone, RP program audits, corrective action documents, and reports of operational occurrences in occupational radiation safety since the last inspection.

Radiological Hazard Assessment

The inspector reviewed the following:

- changes in radiological hazards for occupational workers or members of the public and potential impact of the changes
- conducted walk-downs and made independent radiation measurements and reviewed survey documentation to determine thoroughness and frequency of the surveys
- risk-significant work activities including radiological surveys performed to identify and quantify the radiological hazard and to establish adequate protective measures.

Instructions to Workers

The inspector reviewed labeling of non-exempt licensed radioactive materials containers.

Contamination and Radioactive Material Control

The inspector reviewed the following:

- observed various locations where potentially contaminated material was monitored and released from the radiological control area and inspected methods used for control, survey, and release
- observed the performance of personnel surveying and releasing material for unrestricted use and evaluated whether the work was performed in accordance with plant procedures
- assessed whether the radiation monitoring instrumentation used for equipment release and personnel contamination surveys had appropriate sensitivity
- reviewed sealed source inventory audits and assessed whether the sources were accounted for and were tested for loose surface contamination
- reviewed recent transactions involving nationally tracked sources

Radiological Hazards Control and Work Coverage

The inspector reviewed the following:

- evaluation of radiological conditions and performed independent radiation measurements during walk-downs of the facility
- the application of dosimetry to monitor personnel working in significant dose rate gradients
- postings and physical controls for high radiation areas (HRAs), locked high radiation areas (LHRAs) and very high radiation areas (VHRA)

Risk-Significant HRA and VHRA Controls

The inspector reviewed and discussed with the Radiation Protection Manager (RPM) and supervisors, the controls and procedures for high-risk HRAs and VHRAs including any changes to relevant procedures.

Radiation Worker Performance and RP Technician Proficiency

The inspector reviewed the following:

- the performance of radiation workers and RP technicians with respect to procedure requirements and awareness of radiological conditions
- available radiological problem reports since the last inspection

Problem Identification and Resolution

The inspector evaluated whether problems associated with radiation monitoring and exposure control were being identified at an appropriate threshold and placed in the corrective action program.

b. Findings

No findings were identified.

2RS2 Occupational ALARA Planning and Controls (71124.02 - 1 sample)

a. Inspection Scope

During the period December 16-19, 2013, the inspector assessed performance with respect to maintaining occupational individual and collective radiation exposures as low as is reasonably achievable (ALARA). The inspector used the criteria in 10 CFR 20, applicable Regulatory Guides, TSs, and PSEG procedures for determining compliance.

Inspection Planning

The inspector reviewed the following:

- information regarding collective dose history, current exposure trends, ongoing and planned activities, and the plant’s three year rolling average collective exposure
- changes in the radioactive source term, and reviewed site-specific procedures associated with maintaining occupational exposures ALARA.

Radiological Work Planning

The inspector reviewed the following:

- the results achieved for completed work compared with the intended dose in ALARA planning for dose significant work activities; reviewed work-in-progress and post job reviews and compared the planned person-hour estimates versus actual person-hours; evaluated the accuracy of these estimates; and assessed the reasons for any inconsistencies
- determination of whether post-job reviews were conducted to identify lessons learned

Source Term Reduction and Control

The inspector reviewed the following:

- source term reduction activities, historical trends and current status of plant source term
- current 10 CFR 61 waste stream source term data

Problem Identification and Resolution

The inspector evaluated whether problems associated with ALARA planning and controls are being identified at an appropriate threshold and were placed in the corrective action program

b. Findings

No findings were identified.

2RS3 In-Plant Airborne Radioactivity Control and Mitigation (71124.03 - 1 sample)

a. Inspection Scope

During the period December 16-19, 2013, the inspector reviewed controls for work in airborne radioactivity areas and the use of respiratory protection devices. The inspector used the criteria in 10 CFR Part 20, the guidance in applicable Regulatory Guides, TSs, and PSEG procedures for determining compliance.

Inspection Planning

The inspector reviewed the following:

- use of the respiratory protection program and a description of the types of devices used including location, adequacy of storage facility and quantity of respiratory protection devices stored
- procedures for maintenance, inspection, and use of respiratory protection equipment including self-contained breathing apparatus (SCBA)
- reported performance indicators to identify any related to unintended dose resulting from intakes of radioactive material including during use of respiratory protective devices

Engineering Controls

The inspector reviewed the following:

- assessment of whether PSEG had established threshold criteria for evaluating levels of airborne beta-emitting and alpha-emitting radionuclides

Use of Respiratory Protection Devices

The inspector reviewed the following:

- observation of respiratory protection devices staged and ready for use in the plant, the storage and physical condition of the device components and records of equipment inspection for each type
- equipment storage, maintenance, and quality assurance including training of onsite personnel conducting maintenance and repair of such equipment

SCBA for Emergency Use

The inspector reviewed procedures for surveillance of SCBAs staged in-plant for use during emergencies.

Problem Identification and Resolution

The inspector evaluated whether problems associated with the control and mitigation of in-plant airborne radioactivity were being identified at an appropriate threshold and were placed in the corrective action program.

b. Findings

No findings were identified.

2RS4 Occupational Dose Assessment (71124.04 - 1 sample)

a. Inspection Scope

During the period December 16-19, 2013, the inspector reviewed the monitoring, assessment, and reporting of occupational dose. The inspector used the criteria in 10 CFR 20, applicable Regulatory Guides, TSs, and PSEG procedures for determining compliance.

Inspection Planning

The inspector reviewed the following:

- radiation protection program audits
- procedures associated with dosimetry operations, including issuance/use of external dosimetry, and assessments of dose for radiological incidents
- dosimetry occurrence reports and corrective action program documents for adverse trends related to electronic personal dosimeters (EPDs)

Internal Dosimetry

Routine Bioassay (In-Vivo)

The inspector reviewed the following:

- procedures to assess dose from internally deposited radionuclides, including the release of contaminated individuals
- occupational worker dose assessments

Internal Dose Assessment – Whole Body Count (WBC) Analyses

The inspector reviewed dose assessments performed using the results of WBC analyses.

Special Dosimetric Situations

The inspector reviewed training on the risks of radiation exposure, regulatory aspects of declaring a pregnancy, exposure controls, and the specific process to be used for voluntarily declaring a pregnancy.

Shallow Dose Equivalent

The inspector reviewed dose assessments for shallow dose equivalent, including associated documentation.

Problem Identification and Resolution

The inspector assessed whether problems associated with occupational dose assessment were being identified an appropriate threshold and were placed in the corrective action program.

b. Findings

No findings were identified.

2RS5 Radiation Monitoring Instrumentation (71124.05 - 1 sample)

a. Inspection Scope

During the period December 16-19, 2013, the inspector reviewed the accuracy and operability of radiation monitoring instruments that were used to protect occupational workers and members of the public. The review was against criteria contained in 10 CFR Part 20, 10 CFR Part 50, 40 CFR 190, applicable Regulatory Guides and industry standards, TSs/Offsite Dose Calculation Manual (ODCM), and PSEG station procedures for determining compliance.

Inspection Planning

The inspector reviewed the following:

- procedures that govern instrument source checks and calibrations
- effluent monitor alarm set-points and the calculation methods provided in the ODCM

Walkdowns and Observations

The inspector reviewed the following:

- observation of various portable survey instruments in use and assessed calibration and source check stickers for currency, as well as, instrument material condition and operability
- comparison of monitor response (via local readout) with actual area radiological conditions for consistency
- personnel contamination monitors, portal monitors, SAMs, and bag monitor to evaluate whether the periodic source checks and calibrations were performed in accordance with requirements

Calibration and Testing Program

The inspector reviewed the following:

- observation of portal monitors, personnel contamination monitors, and small article monitors in use and verified that the alarm set-point values were reasonable to ensure that licensed material is not released from the site
- calibration documentation for each instrument selected and reviewed the calibration methods with respect to requirements

Calibration and Check Sources

The inspector reviewed PSEG's source term or waste stream characterization per 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," to assess whether calibration sources used were representative of the types and energies of radiation encountered in the plant.

Problem Identification and Resolution

The inspector evaluated whether problems associated with radiation monitoring instrumentation were being identified by PSEG at an appropriate threshold and were placed in the corrective action program.

b. Findings

No findings were identified.

2RS6 Radioactive Gaseous and Liquid Effluent Treatment (71124.06 - 1 sample)

a. Inspection Scope

During the period December 16-19, 2013, the inspector reviewed monitoring and evaluation of gaseous and liquid effluents. The review was against criteria contained in 10 CFR Part 20, 10 CFR Part 50, 40 CFR 190, applicable Regulatory Guides and industry standards, TSs/Offsite Dose Calculation Manual (ODCM), and PSEG station procedures for determining compliance.

Event Report and Effluent Report Reviews

The inspector reviewed the following:

- 2012Annual Radioactive Effluent Release Report to determine if the report was submitted as required including anomalous results, unexpected trends, and abnormal releases that were identified
- determine if abnormal effluent results were evaluated, were entered in the corrective action program, and were adequately resolved

Offsite Dose Calculation Manual (ODCM) and Final Safety Analysis Report (FSAR) Review

The inspector reviewed the following:

- changes to the ODCM made since the last inspection
- technical bases or evaluations of any ODCM changes

Walk-downs and Observations

The inspector walked-down portions of the facility to identify potential unmonitored release points or whether changes were made to release points.

Procedures, Special Reports, and Other Documents

The inspector reviewed PSEG event reports and/or special reports related to the effluent program issued since the previous inspection.

Sampling and Analyses

The inspector reviewed the inter-laboratory and intra-laboratory comparison program to verify the quality of the radioactive effluent sample analyses. The inspector also discussed PSEG plans associated with any changes or program enhancements.

Dose Calculations

The inspector reviewed the following:

- significant changes in reported dose values compared to the previous radioactive effluent release report

- changes in methodology for offsite dose calculations since the last inspection. The inspector reviewed and discussed meteorological dispersion and deposition factors used in the ODCM and effluent dose calculations.
- the latest Land Use Census to verify changes have been incorporated into the effluent release and environmental programs.

GPI Implementation

The inspector reviewed PSEG implementation of the Ground water Protection Initiative (GPI) including monitoring results, changes to the program, and efforts to identify and control contaminated spills/leaks to ground water

Problem Identification and Resolution

Inspector assessed whether problems associated with the effluent monitoring and control program are being identified by PSEG at an appropriate threshold and placed in the corrective action program.

b. Findings

No findings were identified.

2RS7 Radiological Environmental Monitoring Program (REMP) (71124.07 - 1 sample)

a. Inspection Scope

This area was inspected during the week of December 16-20, 2013, to verify that the REMP appropriately quantifies the impact of radioactive effluent releases to the environment and sufficiently validates the integrity of the radioactive gaseous and liquid effluent release program. The inspector used the requirements in 10 CFR Part 20; 40 CFR Part 190; 10 CFR 50, Appendix I, and the site's TSs, ODCM, and station program procedures to determine acceptability.

Inspection Planning

The inspector reviewed the 2012 Annual Radiological Environmental and Effluent Operating Report to verify that the REMP was implemented in accordance with the TS and ODCM. The inspector reviewed the ODCM to identify environmental monitoring and sampling locations stations.

b. Findings

No findings were identified

2RS8 Radioactive Solid Waste Processing and Radioactive Material Handling, Storage, and Transportation (71124.08 - 1 sample)

a. Inspection Scope

During the week of December 16-20, 2013, the inspectors verified the effectiveness of PSEG's programs for processing, handling, storage, and transportation of radioactive material. The inspectors used the requirements of 10 CFR Parts 20, 61, and 71, and 10 CFR Part 50, Appendix A - Criterion 63 - Monitoring Fuel and Waste Storage, and licensee procedures required by the Technical Specifications/Process Control Program, as criteria for determining compliance.

The inspectors reviewed the solid radioactive waste system description in the UFSAR, the Process Control Program (PCP), and the recent radiological effluent release report for information on the types, amounts, and processing of radioactive waste disposed.

The inspectors reviewed the scope, the results, and the adequacy of the licensee's corrective actions of quality assurance (QA) audits performed for this area since the last inspection.

<u>Radioactive Material Storage</u>

The inspectors inspected areas where containers of radioactive waste were stored. The inspectors verified that the radioactive materials storage areas were controlled and posted as appropriate.

The inspectors verified that the licensee had established a process for monitoring the impact of long-term storage (e.g., buildup of any gases produced by waste decomposition, chemical reactions, container deformation, loss of container integrity, or re-release of free-flowing water). The inspectors verified that there were no signs of swelling, leakage, or deformation.

<u>Radioactive Waste System Walkdown</u>

The inspectors walked down accessible portions of liquid and solid radioactive waste processing systems to verify and assess that the current system configuration and operation agree with the descriptions in the UFSAR, offsite dose calculation manual, and PCP.

The inspectors identified radioactive waste processing equipment that was not operational and/or was abandoned in place, and verified that the licensee had established administrative and/or physical controls for the protection of personnel from unnecessary personnel exposure.

The inspectors reviewed the adequacy of any changes made to the radioactive waste processing systems since the last inspection. The inspectors verified that changes from what was described in the UFSAR were reviewed and documented.

The inspectors identified processes for transferring radioactive waste resin and/or sludge discharges into shipping/disposal containers. The inspectors verified that the waste stream mixing, sampling procedures, and methodology for waste concentration averaging were consistent with the PCP, and provided representative samples of the waste product for the purposes of waste classification.

For those systems that provide tank recirculation, the inspectors verified that the tank recirculation procedure provided sufficient mixing.

The inspectors verified that the licensee's PCP correctly described the current methods and procedures for dewatering waste.

Waste Characterization and Classification

The inspectors identified radioactive waste streams, and verified that PSEG's radiochemical sample analysis results were sufficient to support radioactive waste characterization. The inspectors verified that PSEG's use of scaling factors and calculations to account for difficult-to-measure radionuclides was technically sound and based on current analyses.

The inspectors verified that changes to plant operational parameters were taken into account to (1) maintain the validity of the waste stream composition data between the annual or biennial sample analysis update, and (2) verified that waste shipments continued to meet applicable requirements.

The inspectors verified that the licensee had established and maintained an adequate QA program to ensure compliance with applicable waste classification and characterization requirements.

Shipment Preparation

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 licensee verification of shipment readiness. The inspectors verified that the requirements of any applicable transport cask certificate of compliance had been met. The inspectors verified that the receiving licensee was authorized to receive the shipment packages

The inspectors verified that the shippers were knowledgeable of the shipping regulations and that shipping personnel demonstrated adequate skills to accomplish the package preparation requirements for public transport. The inspectors verified PSEG's training program provided training to personnel responsible for the conduct of radioactive waste processing and radioactive material shipment preparation activities.

Shipping Records

The inspectors identified non-excepted package shipment records and verified that the shipping documents indicate the proper shipper name; emergency response information and a 24-hour contact telephone number; accurate curie content and volume of material; and appropriate waste classification, transport index, and international shipping identification number. The inspectors verified that the shipment placarding was consistent with the information in the shipping documentation.

Identification and Resolution of Problems

The inspectors verified that problems associated with radioactive waste processing, handling, storage, and transportation, were being identified by the licensee at an appropriate threshold, were properly characterized, and verified the appropriateness of the corrective actions for a selected sample of problems. The licensee generated six notifications to document material condition deficiencies identified during this inspection.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification (71151 – 2 samples)

a. Inspection Scope

During the period December 16-19, 2013, the inspector reviewed various corrective action documents to determine if issues met the report threshold for the occupational exposure control effectiveness PI or the threshold for the public exposure control effectiveness PI. The inspector used PI definitions and guidance contained in the Nuclear Energy Institute Document 99 02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 31, 2013, to determine the accuracy of the PI data reported.

Occupational Exposure Control Effectiveness

The inspector reviewed EPD dose alarms, dose reports, and dose assignments for any intakes that occurred during the time period reviewed to determine if there were any potentially unrecognized PI occurrences. The inspector also conducted walk-downs of accessible locked High and Very High Radiation Area entrances to determine the adequacy of the controls in place for these areas.

RETS/ODCM Radiological Effluent Occurrences

The inspector reviewed the corrective action report database and selected individual reports generated since this indicator was last reviewed to identify any potential occurrences such as unmonitored, uncontrolled, or improperly calculated effluent releases that may have impacted offsite dose. The inspector reviewed gaseous and liquid effluent summary data and the results of associated offsite dose calculations to determine if indicator results were accurately reported. The inspector also reviewed methods for quantifying gaseous and liquid effluents and determining effluent dose.

b. Findings

No Findings were identified.

4OA2 Problem Identification and Resolution (71152 – 5 samples)

.1 Routine Review of Problem Identification and Resolution Activities

a. Inspection Scope

As required by Inspection Procedure (IP) 71152, "Problem Identification and Resolution (PI&R)," the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that PSEG entered issues into the CAP at an appropriate threshold, gave adequate attention to timely corrective actions, and identified and addressed adverse trends. In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the CAP and periodically attended notification screening meetings.

b. Findings

No findings were identified.

.2 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a semi-annual review of site issues, as required by IP 71152, "Problem Identification and Resolution," to identify trends that might indicate the existence of more significant safety issues. In this review, the inspectors included repetitive or closely-related issues that may have been documented by PSEG outside of the corrective action program, such as trend reports, performance indicators, major equipment problem lists, system health reports, maintenance rule assessments, and maintenance or corrective action program backlogs. The inspection also reviewed PSEG's corrective action program database for the period of June 2013 to December 2013 to assess the notifications written as well as individual issues identified during the NRC's daily condition report review (Section 4OA2.1). The inspectors reviewed the Hope Creek station performance improvement integrated matrix (PIIM) for the first cycle of 2013, conducted under procedure LS-AA-125-1006, "Performance Improvement Integrated Matrix," to verify that PSEG personnel were appropriately evaluating and trending adverse conditions in accordance with applicable procedures.

b. Findings and Observations

No findings were identified directly during this trend review. One Green self-revealing finding related to PSEG's failure to identify and correct an adverse trend regarding Bailey module failures is documented in Section 4OA3 of this report.

Plant engineering trends equipment performance through a series of industry accepted performance indicators which include: Critical Component Failures, Maintenance Rule Functional Failures, and Mitigating System Performance Index. Plant engineering maintains these performance indicators in a shared location for common access and routinely presents the trend data to station management. Engineering trends overall equipment reliability using procedure ER-AA-2200, "Equipment Reliability Performance Objectives and Criteria Bubble Chart Analysis." Using this process, engineering reviews the documentation associated with several types of equipment reliability events gathered over the previous 24-month period. Engineering codes each event into various "bubbles" depending upon the casual factors that contributed to the equipment failure.

The process throughput is engineering identified concerns and adverse trends associated with equipment reliability. In December 2013, an engineering bubble chart analysis identified a potential degraded performance trend (14 notifications over two years) where part issues have impacted equipment reliability (notification 20633952). This potential adverse trend in part issues is in the process of being evaluated by a common cause evaluation.

The inspectors also noted that PSEG personnel identified the following trends and entered them into the corrective action program: an adverse trend in drywell floor drain leakage (20613799, 20614384, 20615125); an adverse trend in operability determination screening quality and timeliness (20627981, 20627982); an adverse trend in compliance with administrative procedures (20615840); an adverse trend in the increase in the preventative maintenance change evaluation backlog (20616758); and an adverse trend in security facilities. The inspectors also reviewed the 2013 first cycle Hope Creek Station PIIM and noted that PSEG identified the following fundamentals in variance: rigor associated with the configuration management process rigor (70155767); maintenance rule program implementation (70151301); corrective action program evaluation quality (70151290); and operating experience response timeliness (70151244).

The inspectors also documented in Section 4OA3.2 of this report, a Green self-revealing finding related to PSEG's failure to identify and correct an adverse trend regarding Bailey module failures. During the review of the root cause associated with the 'B' CW pump trip (70155514), the inspectors noted that PSEG's extent of condition related to the identified programmatic weakness regarding the performance monitoring and trending program for circuit card failures did not take action to review performance monitoring and trending programs of other similar components (circuit cards, auxiliary relays, fuse modules, etc.). The inspectors determined this extent of condition weakness to be minor in accordance with IMC 0612 because it had no actual consequences. PSEG appropriately captured this weakness in their CAP as notification 20636783.

Based on the comprehensive review of PSEG's trending, the inspectors concluded that PSEG was appropriately identifying and entering issues into the corrective action program, adequately evaluating the identified issues, and appropriately identifying adverse trends before they become more safety significant problems.

.3 Annual Sample: Unacceptable Preconditioning Extent-of-Condition Review

a. Inspection Scope

The inspectors performed an in-depth review of PSEG's evaluations and corrective actions associated with an NRC-identified NCV related to unacceptable preconditioning during a reactor building-to-torus vacuum breaker surveillance test. The inspectors assessed PSEG's problem identification threshold, cause analysis, extent-of-condition reviews, and the prioritization and timeliness of corrective actions to evaluate whether PSEG was appropriately identifying, characterizing, and correcting problems associated with this issue and whether the planned and/or completed corrective actions were appropriate. The inspectors compared the actions taken to the requirements of PSEG's corrective action program, 10 CFR Part 50 Appendix B, and PSEG Technical Specifications. The inspectors performed an independent review of a risk-informed sample of maintenance and operations procedures for potential unacceptable preconditioning and reviewed a sample of PSEG preconditioning evaluations to verify alignment with NRC guidance. In addition, the inspectors performed a walkdown of the reactor building-to-torus vacuum breakers, including control room instrumentation, to independently assess the material condition, operating environment, and configuration control.

Findings and Observations

No findings were identified.

In April 2012, NRC inspectors identified unacceptable preconditioning in surveillance test procedure HC.MD-ST.GS-0002, "Reactor Building to Torus Vacuum Relief Valve 18 Month Testing." In addition, inspectors noted that PSEG missed an opportunity to identify the issue in response to unacceptable preconditioning of a HPCI condensate storage suction valve identified in November 2009. Specifically, PSEG's extent-of-condition review narrowly focused on operations' surveillance test procedures and did not expand the review to include maintenance procedures that were used to satisfy technical specification surveillance requirements.

In response to the April 2012 NCV (see NRC Inspection Report 05000354/2012003, Section 1R12), PSEG promptly revised the vacuum breaker surveillance test procedure, performed an extent-of-condition review of maintenance procedures that control vacuum breaker testing, reviewed a sample of 12 maintenance surveillance test procedures, and initiated a long-term corrective action to review all Instrumentation and Controls, electrical, and mechanical maintenance surveillance test procedures for unacceptable preconditioning. In addition, PSEG initiated an apparent cause evaluation (ACE) to assess why a cross disciplined extent-of-condition review was not performed for previously identified unacceptable preconditioning concerns. PSEG determined that the apparent cause was that inadequate assignments were made during corrective action program screening due to deficiencies in corrective action program procedures regarding the guidance for selection of evaluation type. In response, PSEG implemented changes to the applicable corrective action program procedures regarding risk and uncertainty analysis for assigned evaluations, and provided additional guidance to Management Review Committee and Station Ownership Committee members.

The inspectors concluded that PSEG had taken timely and appropriate actions in accordance with the 10 CFR Part 50 Appendix B, PSEG technical specifications, and PSEG's corrective action program. The inspectors determined that PSEG's associated ACE and preconditioning evaluations were sufficiently thorough and based on the best available information, sound judgment, and relevant industry operating experience. PSEG's assigned corrective actions were aligned with the identified causal factors, adequately tracked, appropriately documented, and completed as scheduled. Based on the documents reviewed, control room and plant walkdowns, and discussions with maintenance and operations personnel, the inspectors noted that PSEG personnel identified problems and entered them into their corrective action program at a low threshold. The inspectors noted that PSEG procedure PP-AA-3001, "Position Paper on Preconditioning," was appropriately aligned with regulatory positions and guidance regarding preconditioning, including NRC Inspection Manual Part 9900 Technical Guidance, "Maintenance - Preconditioning of Structures, Systems, and Components before Determining Operability." The inspectors also noted numerous preconditioning caution statements in PSEG surveillance test procedures that demonstrated an appropriate sensitivity to potential preconditioning.

.4 Annual Sample: EDG Room Cooler Recirculation Fan Trips

a. Inspection Scope

The inspectors performed an in-depth review of PSEG's common cause analysis, troubleshooting plans, extent-of-condition reviews, and short and long term corrective actions associated with multiple unexpected diesel room recirculation fan trips, with recent trips occurring on July 26, 2013 (notification 20616442 on BV412) and April 18, 2013 (notification 20604153 on BV412). Since January 2005, PSEG identified 30 unexpected diesel room recirculation fan trips, this inspection focused on PSEG's problem identification, evaluation, and resolution associated with the EDG recirculation fan trips and potential reliability challenges.

The inspectors assessed PSEG's common cause analyses, troubleshooting plans, extent-of-condition reviews, compensatory actions, and the prioritization and timeliness of PSEG's corrective actions to determine whether PSEG was appropriately, identifying, characterizing, and correcting problems associated with the diesel room recirculation fan trips and whether the planned or completed corrective actions were appropriate. The inspectors compared the actions taken to the requirements of PSEG's CAP.

b. Findings and Observations

No findings were identified.

The inspectors determined that PSEG appropriately identified, characterized, and implemented corrective actions associated with unexpected diesel room recirculation fan trips. The inspectors noted that a significant portion of these Schneider Electric Square D MasterPact breaker failures involved false trips on circuit faults and/or internal binding issues. The troubleshooting efforts have not revealed any actual fault trips. PSEG has identified that frequent cycling of these breakers, based on temperature demands, as a possible factor in the ongoing breaker failures. Inspectors noted that PSEG has planned corrective actions to modify the diesel room recirculation fan logic to utilize motor starters for normal start and stop operation of the fans. The motor starters will allow breakers to remain closed and be available for circuit protection. The inspectors determined that the planned corrective actions associated with false circuit faults were reasonable. The inspectors also reviewed the breaker binding issue and determined that PSEG's planned corrective actions to install interval timer relays in place of conventional aux relays to generate proper duration pulsed signals were appropriate.

The inspectors determined PSEG's overall response to the issue was commensurate with the safety significance, was timely, and the actions taken and planned were reasonable to resolve the unexpected diesel room recirculation fan trips.

.5 Annual Sample: Review of Containment Vent Backup Pneumatic Supply Calculation

a. Inspection Scope

The inspectors performed an in-depth review of Hope Creek's corrective actions associated with notification 20558322 regarding an instrument air check valve (1KBV-300). Specifically, inspectors observed that there was no preventive maintenance on the containment vent instrument air system check valve 1KBV-300, and that design calculation H-1-KB-MDC-1007, "Backup Pneumatic Supply for

1GSHV-4964 and 1GSHV-11541 Valves," did not account for potential leakage through the check valve 1KBV-300.

The inspectors assessed Hope Creek's problem identification threshold and the prioritization and timeliness of corrective actions to determine whether Hope Creek was appropriately identifying, characterizing, evaluating, and correcting problems associated with this issue and whether the completed corrective actions were appropriate. In addition, the inspectors performed field walkdowns and interviewed operations and engineering personnel to assess the effectiveness of the implemented corrective actions.

b. Findings and Observations

Introduction. The inspectors identified a finding of very low safety significance (Green) for PSEG's failure to ensure evaluations addressed identified issues in accordance with PSEG procedure LS-AA-125, "Corrective Action Program." Specifically, PSEG failed to adequately assess the functionality of the containment vent following NRC identification of inadequate maintenance practices for an instrument air check valve (1KBV-300) and that design calculation H-1-KB-MDC-1007, "Backup Pneumatic Supply for 1GSHV-4964 and 1GSHV-11541 Valves," did not account for leakage through the valve.

Description. In a severe accident, air-operated valves 1GSHV-4964 and 1GSHV-11541 are used to relieve containment pressure by providing an elevated release point through a hard-pipe vent. The containment vent valves are credited in the station probabilistic risk analysis for beyond design basis events, and were originally installed in accordance with NRC Generic Letter 89-16, "Installation of a Hardened Wetwell Vent." These valves are normally operated using the instrument air system, and have backup nitrogen gas bottles for operation upon a loss of instrument air. Instrument air check valve 1KBV-300 ensures that the air from the backup supply is not diverted to the instrument air system.

Ensuring functionality of the containment vent is important for reducing containment pressure. However, it is also important for operation of the safety relief valves (SRVs). The SRVs are used to automatically depressurize the reactor so that low pressure emergency core cooling systems can provide cooling as needed. Hope Creek's Licensed Operator Training for, HC.OP-EO.ZZ-0102, "Primary Containment Control Drywell (Temperature / Pressure and Hydrogen)," states that the maximum primary containment pressure at which the SRVs can be opened and remain open is 69 psig. If insufficient nitrogen pressure exists to stroke the containment vent valve as needed to maintain containment pressure below 65 psig, as required by procedure, this could also impact the use of the SRVs and the low pressure emergency core cooling systems.

The inspectors reviewed nitrogen capacity calculation H-1-KB-MDC-1007, "Backup Pneumatic Supply for 1GSHV-4964 and 1GSHV-11541 Valves," and operator rounds log HC.OP-DL.ZZ-0004, "Log 4 Reactor Building Data Log," to ensure the backup nitrogen bottles would supply sufficient nitrogen to vent containment for beyond design basis events. The nitrogen capacity calculation developed the basis for the minimum allowable standby nitrogen bottle pressure to ensure the containment vent valves could be operated as described in emergency operating procedures when instrument air was not available.

In 2006, component design basis inspection team inspectors observed that there was no preventive maintenance on the hardened vent system check valve 1KBV-300. PSEG generated notification 20306042 in response to this question and evaluated the condition with order 70064117. That evaluation incorrectly determined that no preventive maintenance or testing was required for that check valve.

In 2012, inspectors noted that the design calculation H-1-KB-MDC-1007, "Backup Pneumatic Supply for 1GSHV-4964 and 1GSHV-11541 Valves," did not account for potential leakage through instrument air check valve KBV-300. Additionally, the inspectors observed that 1KBV-300 had no preventive maintenance schedule or leakage testing associated with it to validate potential leakby through the check valve. PSEG entered this issue into their corrective action program as notification 20558322 and validated 1KBV-300 had no maintenance history since installation. The functionality assessment performed for this issue maintained that the combination of a loss of instrument air and the 1KBV-300 check valve failure could prevent the operation of the containment vent, however manually opening these valves can be accomplished using hydraulic jacks and guidance contained in the containment venting procedure. However, the inspectors identified there is no requirement to periodically operate the containment vent using the hydraulic jacks. The hydraulic jacks have only ever been used to operate the containment vent in 1992 following their installation.

PSEG's corrective actions included installation of a design change to modify instrument air piping to support leak rate testing of 1KBV-300 and addition of 1KBV-300 to its check valve monitoring and preventive maintenance program. Subsequent leak rate testing of the 1KBV-300 measured an as-found leakage of 267 sccm. Taking this as-found leakage into account equates to a minimum standby bottle pressure of 543 psig. The minimum pressure required by the design calculation, with no leakage through 1KBV-300 and added margin was 550 psig. Accounting for leakage through 1KBV-300 reduced the available margin from 131 psig to 7 psig. The inspectors reviewed past nitrogen bottle pressure data for the past three years and noted that minimum standby bottle pressure did not fall below 600 psig. PSEG also completed a revision to design calculation H-1-KB-MDC-1007 to credit up to 500 sccm of leakage through 1KBV-300. This revision increased the minimum allowable standby nitrogen gas bottle pressure from the previous limit of 550 psig to 850 psig. This calculation revision was not completed until September 2013, after the design change installation and leakrate testing.

Analysis. The inspectors determined PSEG's failure to ensure evaluations addressed identified issues in accordance with PSEG procedure LS-AA-125, "Corrective Action Program," was a performance deficiency which was reasonably within the licensee's ability to foresee and prevent." Specifically, PSEG failed to adequately assess the functionality of the containment vent when inspectors identified that design calculation H-1-KB-MDC-1007, "Backup Pneumatic Supply for 1GSHV-4964 and 1GSHV-11541 Valves," did not account for potential leakage through instrument air check valve 1KBV-300. The functionality assessment performed for this issue relied on manual containment vent operation using the hydraulic jacks and guidance contained in the containment venting procedure. However, the inspectors identified there is no requirement to periodically operate the containment vent using the hydraulic jacks. The hydraulic jacks have only been used to operate the containment vent in 1992 following their installation. The inspectors concluded that PSEG did not adequately assess that the containment vent system would function using manual operation or the backup nitrogen air system.

This issue was more than minor because it was associated with the design control attribute of the mitigating systems cornerstone, and affected the cornerstone's objective to ensure the availability and reliability of systems that respond to initiating events to prevent undesirable consequences. Specifically, inability to operate the containment vent valves, when required for beyond design basis scenarios, could challenge operation of SRVs. The inspectors determined the finding to be of very low safety significance (Green) in accordance with IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power, dated June 19, 2012." Using Exhibit 2, which contains the screening questions for the Mitigating Systems Cornerstone, the inspectors determined that the finding screened as Green because: it was not a deficiency affecting the design or qualification of the containment vent; it did not represent a loss of system or function; it did not represent the loss of function for any TS system, train, or component beyond the allowed TS outage time; and it did not represent an actual loss of function of any non TS trains of equipment designated as high safety-significant in accordance with the PSEG's maintenance rule program.

The inspectors determined that the finding had a cross cutting aspect in the Human Performance area associated with Resources, because PSEG did ensure that personnel, equipment, procedures, and other resources are available and adequate to assure nuclear safety, specifically, those necessary for maintaining long term plant safety by maintenance of design margins, minimization of long-standing equipment issues, minimizing preventative maintenance deferrals, and ensuring maintenance and engineering backlogs which are low enough to support safety. Specifically, PSEG did not ensure maintenance of design margin for the containment vent system when concerns were identified regarding its functionality. This included PSEG relying upon operation of the containment vents with hydraulic jacks that have not been operated since 1992 following their installation [H.2(a)].

Enforcement. This finding was not a violation of NRC requirements because the performance deficiency involved a beyond design basis event. Further, the containment vent function is not covered by technical specifications, and is not a part of Hope Creek's licensing basis. PSEG entered this problem into their corrective action program as notification 20558322. **(FIN 05000354/2013005-03, Inadequate Evaluation of Containment Vent Functionality)**

.6 Annual Sample: Safety Relief Valve SetPoint Drift and Seat Leakage

a. Inspection Scope

The inspectors performed an in-depth review of PSEG's evaluations and the effectiveness of the corrective actions associated with long standing Hope Creek Generating Station (HCGS) main steam (MS) SRV deficiencies. Specifically, at HCGS, SRV setpoints have exceeded the TS allowable tolerance during as-found lift testing since the first operating cycle (1988). HCGS uses vendor supplied two-stage SRVs. Industry operating experience has shown the two-stage SRVs to be problematic with respect to seat leakage and setpoint drift, as documented in numerous NRC correspondences. This inspection was performed to determine if PSEG was continuing to appropriately identify and evaluate SRV issues at the station and taking appropriate corrective actions to ensure the SRVs remained capable of performing the intended safety function.

The inspectors assessed PSEG’s problem identification threshold, associated root cause analyses and evaluations, extent-of-condition reviews, and the prioritization and timeliness of actions to evaluate whether HCGS was appropriately identifying, characterizing, and correcting problems associated with the issue; and whether the planned or completed corrective actions were appropriate and met the requirements of their corrective action program. The inspectors compared the actions taken to the requirements of PSEG’s corrective action program and 10 CFR Part 50, Appendix B. The inspectors reviewed the applicable notifications and associated documents, including work orders, maintenance procedures, and as-found test results. The inspectors reviewed the licensee’s actions to address other possible or contributing causes. The inspectors walked down the controls in the control room and interviewed operators and engineering personnel to assess the acceptability and effectiveness of the implemented corrective actions and to evaluate the adequacy of PSEG's administrative controls for SRV seat leakage.

b. Findings and Observations

No findings were identified.

The inspectors determined that while PSEG continues to experience repetitive issues associated with MS SRV as-found lift setpoints and SRV seat leakage, PSEG personnel are appropriately identifying, monitoring, and documenting the circumstances surrounding the issue, and are continuing to evaluate the matter in accordance with PSEG’s procedures. Further, the inspectors observed that PSEG personnel continue to investigate possible causes and solutions to these problems. The inspectors found PSEG’s numerous activities, past approaches, and on-going plans to be adequate to determine the cause or causes of the problems with their MS system two-stage SRVs. Examples of these efforts include: changing out all SRVs during every outage; conducting studies of system vibration and natural frequency configurations, as well as steam chemistry investigations; performing comparisons with other utility sites; researching, testing, and evaluating a potential alternative valve; and developing an alternate plan to replace the two-stage valves with 3-stage SRVs from the same vendor.

Additionally, the inspectors noted that an extensive effort to improve HCGS’s documentation is underway, such as including photos and descriptions of proper insulation installation to avoid or reduce possible thermal effects or distortions to the pilot or main valve assemblies. Overall, the inspectors determined that the licensee’s corrective actions were adequate and reasonable in addressing the probable and contributing causes of the longstanding problem with SRV setpoint drift at the station.

4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153 – 7 samples)

.1 Plant Events

a. Inspection Scope

For the plant events listed below, the inspectors reviewed and/or observed plant parameters, reviewed personnel performance, and evaluated performance of mitigating systems. The inspectors communicated the plant events to appropriate regional personnel, and compared the event details with criteria contained in IMC 0309, "Reactive Inspection Decision Basis for Reactors," for consideration of potential reactive inspection activities. As applicable, the inspectors verified that PSEG made appropriate emergency classification assessments and properly reported the event in accordance with 10 CFR Parts 50.72 and 50.73. The inspectors reviewed PSEG's follow-up actions related to the events to assure that PSEG implemented appropriate corrective actions commensurate with their safety significance.

- Turbine trip followed by a reactor trip on high moisture separator level due to normal and emergency level control failures on December 1, 2013 (Event Notification (EN) 49592)
- Turbine trip followed by a reactor trip on high moisture separator level due to an emergency level control failure on December 5, 2013 (EN 49608)
- Loss of reactor building ventilation fans resulting in a loss of secondary containment negative pressure on December 19, 2013 (EN 49665)
- Loss of the 'A' control room chiller during scheduled maintenance on the 'B' control room chiller, resulting in both control room air conditioning subsystems being inoperable on December 20, 2013 (EN 49671)

b. Findings

No findings were identified.

.2 (Closed) LER 05000354/2013-002-00(-01): Reactor Scram due to Degrading Condenser Vacuum

a. Inspection Scope

On June 12, 2013, at 1:33 pm, Hope Creek was manually scrammed from 100 percent power due to degrading main condenser vacuum. This condition occurred due to the trip of the 'B' CW pump with the 'B' CW pump discharge valve stuck in the full-open position. During the scram response, the operating reactor feed pump tripped due to degrading vacuum and the operators manually placed the RCIC system in service for reactor inventory control. The operators completed the scram response procedures and placed the plant in a stabilized hot shutdown condition.

This event was reported under 10 CFR 50.73(a)(2)(iv)(A) for a valid manual actuation of the reactor protection system and manual initiation of the RCIC system. The inspectors reviewed PSEG's LER, root cause evaluation report (70155514), supporting documentation, station procedures, and interviewed several members of station staff and management regarding the event. One finding was identified and is discussed below. This LER is closed.

b. Findings

Introduction: A Green self-revealing finding was identified for PSEG's failure to identify and correct an adverse trend regarding 48 Bailey module failures across multiple systems since 2005, including six Bailey module failures in the CW system. As a result of continued problems associated with this previously unidentified adverse trend, on June 12, 2013, the ‘B’ CW pump tripped resulting in a manual scram of the reactor due to degrading condenser vacuum. The CW pump trip was determined to be caused by the activation of a normally de-energized auxiliary relay due to conductive filament growth (metallic whiskers) that created a short circuit on the Bailey auxiliary relay card.

Description: In response to the manual scram on June 12, 2013, involving the trip of the ‘B’ CW pump, and a second trip of the ‘B’ CW pump on August 1, which resulted in a power reduction to 70 percent due to a recirculation pump runback, PSEG performed a root cause report (notification 70155514). PSEG’s root cause report determined that site engineering had not implemented a program for metallic whiskers on circuit cards, no preventative or predictive maintenance programs existed for CW pump Bailey auxiliary relay cards, and no trending program had been in place to identify the existing adverse trend in Bailey module failures (48 Bailey module failures since 2005).

The root cause report discovered as part of the extent of condition review that industry operating experience concerning circuit card related problems had been documented and evaluated in notification 20525505 and order 70128730 in September 2011. These evaluations, in part, recommended trending circuit card failures to identify any adverse trends and periodically perform an aggregate assessment of circuit card failures within and across systems. Prior to the ‘B’ CW pump trips on June 12 and August 1, 2013, no actions regarding the need to trend or perform an aggregate assessment of circuit card failures were taken by PSEG.

PSEG found during the root cause that as part of the Hope Creek operating license condition 2.C.5, Amendment 40 required the implementation of a Bailey Solid State Logic Module (SSLM) reliability program. The intent of this reliability program was, in part, to obtain reliability data, failure characteristic information and the root cause of any failure of either safety related or non-safety related SSLMs. PSEG procedure HC.IC-AP.ZZ-0017(Q), Bailey Module Reliability Program, Revision 0, established the required reliability program. This program only trends logic module failures, not auxiliary relay card or fuse module failures. PSEG’s root cause report determined that had a trending program been in place for the Bailey fuse module and auxiliary relay card failures, additional actions would have been identified to replace these modules prior to failure.

The inspectors reviewed the root cause report and PSEG procedures. The inspectors determined that PSEG’s CAP procedure, LS-AA-125, requires issue trending, identification of trends and actions to resolve adverse trends and deficiencies and that PSEG procedure, ER-AA-20, “Equipment Reliability Program,” requires system and component performance monitoring and trending.

To address the programmatic weakness identified regarding the performance monitoring and trending program for circuit card failures, PSEG is amending the Bailey Module Reliability Program to include fuse module and auxiliary card failures. PSEG has entered this issue into their CAP as notification 20611721.

Analysis: The inspectors determined that PSEG’s failure to identify and correct the adverse trend regarding Bailey module failures across multiple systems was a performance deficiency which was reasonably within the licensee’s ability to foresee and prevent. Specifically, PSEG’s Equipment Reliability Program, described in ER-AA-20, should have identified the trend. The finding was more than minor because it was associated with the Initiating Events cornerstone and affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, PSEG's failure to identify and correct the adverse trend regarding Bailey module failures resulted in a manual scram from 100 percent power due to the trip of the 'B' CW pump concurrent with the 'B' CW discharge valve stuck in the full-open position. The finding was determined to be of very low safety significance (Green) in accordance with Appendix A of IMC 0609, "The Significance Determination Process for Findings At-Power," because the finding did not contribute to 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.

The inspectors determined that this finding had a cross-cutting aspect in the area of Problem Identification and Resolution, Corrective Action Program, because PSEG did not trend and assess information from the CAP and other assessments in the aggregate to identify programmatic and common cause problems. Specifically, PSEG failed to trend or perform an aggregate assessment of Bailey module and auxiliary card failures. [P.1(b)]

Enforcement: Enforcement action does not apply because the performance deficiency did not involve a violation of regulatory requirements. Specifically, the CW pump Bailey fuse module and auxiliary cards are not safety-related components. Because the finding does not involve a violation of regulatory requirements and has very low safety significance, it is identified as **FIN 05000354/2013005-04, Failure to Identify Adverse Trend Regarding Bailey Module and Auxiliary Card Failures.**

.3 (Closed) LER 05000354/2012-004-00: Operations with a Potential to Drain the Reactor Vessel (OPDRV) Without Secondary Containment

On October 15, 20, and 23, 2013, during a planned refueling outage and the reactor cavity flooded up in Mode 5, Hope Creek conducted multiple OPDRVs without an operable secondary containment. The conduct of an OPDRV without establishing secondary containment integrity is a condition prohibited by TS as defined by 10 CFR 50.73(a)(2)(i)(B). Secondary containment is required by TS 3/4.6.5.1 in Operational Condition***,** which is a condition during an OPDRV. The required action for this specification is to suspend OPDRV operations.

In this case, the specific OPDRVs were the 'A' Recirculation Pump seal replacement (5:00 p.m. on October 15, 2013, through 11:43 a.m. on October 30, 2013), the control rod drive mechanism and local power range monitor replacement (12:00 p.m. on October 20, 2013, through 7:06 p.m. on October 24, 2013), and the reactor water cleanup local leak rate test (8:13 p.m. on October 23, 2013, through 7:06 p.m. on October 24, 2013). The OPDRVs were completed in accordance with PSEG procedure OP-HC-108-102, "Management of Operations with the Potential to Drain the Reactor Vessel." These OPDRVs were completed and exited at 11:43 a.m. on October 30, 2013.

The NRC issued EGM 11-003, Revision 1, "Enforcement Guidance Memorandum On Dispositioning Boiling Water Reactor Licensee Noncompliance With Technical Specification Containment Requirements During Operations With A Potential For

Draining The Reactor Vessel," on December 20, 2012, which provides, in part, for the exercise of enforcement discretion if the licensee demonstrates that it has met four specific criteria during an OPDRV activity. The inspectors determined that PSEG's implementation of these four criteria during these OPDRV activities was adequate and met the intent of EGM 11-003, Revision 1.

The inspectors' assessments of PSEG's implementation of these four criteria during each of the multiple OPDRV activities are described below:

- The inspectors observed that, as required by the EGM, the OPDRV activity was logged in the control room narrative logs and that the log entry appropriately recorded the safety-related pump ('D' RHR) that was the standby source of makeup designated for the evolution.
- The inspectors noted that the reactor vessel water level was maintained at least 22 feet and 2 inches over the top of the RPV flange. Although this did not meet the specific requirement of at least 23 feet as listed in EGM 11-003, Revision 1, which was based on the BWR/4 Standard TS LCO 3.9.8 applicability, the inspectors concluded that the water level maintained by PSEG was acceptable because it was in compliance with the minimum water level allowed by HC TS LCO 3.9.8 applicability. The inspectors also noted that at least one safety-related pump was the standby source of makeup designated in the control room narrative logs for the evolution. PSEG reported that the worst case estimated time to drain the reactor cavity to the RPV flange was 26.8 hours, which met the EGM criteria of >24 hours.
- The inspectors verified that the OPDRV was not conducted in Mode 4 and that PSEG did not move irradiated fuel during the OPDRV. The inspectors noted that PSEG had in place a contingency plan for isolating the potential leakage path. The inspectors verified that two independent means of measuring RPV water level (one alarming) were available for identifying the onset of loss of inventory events with sufficient time to close secondary containment before reactor water level reached the top of the RPV flange.
- In preparation for taking the plant to Operational Condition * (OPDRV), PSEG performed a risk assessment and with the exception of not setting secondary containment, the inspectors did not identify any other equipment where PSEG did not follow the TS applicability and action requirements for Operational Condition *.

TS 3.6.5.1 is applicable in Operational Conditions 1, 2, 3 and * requires that secondary containment integrity shall be maintained. Operational Condition* is defined, in part, as being during OPDRV. TS 3.6.5.1, action b, states, in part, in operational condition, * suspend operations with a potential for draining the reactor vessel. Contrary to the above, between 1700 on October 15, 2013, and 1143 on October 30, 2013, Hope Creek Generating Station did not maintain secondary containment integrity while conducting OPDRV activities. Because the violation was identified during the discretion period described in EGM 11-003 Revision 1, the NRC is exercising enforcement discretion in accordance with Section 3.5, "Violations Involving Special Circumstances," of the NRC Enforcement Policy and, therefore, will not issue enforcement action for this violation.

In accordance with EGM 11-003 Revision 1, each licensee that receives discretion must submit a license amendment request within 4 months of the NRC staff's publication in the Federal Register of the notice of availability for a generic change to the Standard Technical Specifications to provide more clarity to the term OPDRV. This time period was extended to 12 months in the December 2013 revision. The inspectors observed that PSEG is tracking the need to submit a license amendment request in its corrective action program as notification 20559547. This LER is closed.

.4 (Closed) LER 05000354/2012-006-00: Operations with a Potential to Drain the Reactor Vessel Without Secondary Containment Operable

On October 31, 2013, at 9:30 a.m., during H1R18, the RWCU system was placed in letdown to radwaste to control RPV inventory. EGM 11-003 Revision 1 states: "The addition and removal of small volumes of water inventory from the RPV, for example control rod drive cooling water, is considered steady-state water level control and not an OPDRV provided the instrumentation and valves for automatic isolation of the drain-down path remain available." Contrary to this, the operators identified at 4:31 p.m. on October 31, 2013, that the Level 2 isolation instrumentation used to support the automatic isolation of the drain-down path was removed from service for refueling outage maintenance. TS 3.6.5.1, Secondary Containment Integrity, requires that secondary containment to be operable during OPDRV activities. The removal of this automatic isolation function placed Hope Creek in an OPDRV condition in which secondary containment is required to be operable and was not operable. The operating crew identified that this automatic function had been removed due to other maintenance and restored compliance with the EGM isolation requirements at 4:31 p.m. on October 31, 2013.

This condition was reportable under 10 CFR 50.73(a)(2)(i)(B) as an operation or condition which was prohibited by TSs. The inspectors performed an in-depth review of this LER and the prompt investigation and operability determination associated with notifications 20631218 and 20628041. The inspectors documented a licensee-identified violation associated with a condition prohibited by Technical Specifications in Section 4OA7 of this report. This LER is closed.

4OA6 Meetings, Including Exit

On January 16, 2014, the inspectors presented the inspection results to Mr. P. Davison, Site Vice President of Hope Creek, and other members of the Hope Creek staff. The inspectors verified that no proprietary information was retained by the inspectors or documented in this report.

4OA7 Licensee-Identified Violation

The following violation of very low safety significance (Green) was identified by PSEG and is a violation of NRC requirements which meet the criteria of the NRC Enforcement Policy for being dispositioned as an NCV.

- Technical Specification 3.6.5.1, "Secondary Containment Integrity" requires, in part, that secondary containment be operable during OPDRV activities in OPCON*. The action statement with secondary containment inoperable in OPCON* is to suspend operations with a potential to drain the reactor vessel. Contrary to the above, from 9:30 a.m. until 5:31 p.m. on October 31, 2013, Hope Creek did not comply with this TS action statement or the EGM 11-003 Revision 1 guidance, and secondary containment should have been operable. The failure to comply with this guidance resulted in a condition prohibited by TSs until the condition was corrected by restoring the automatic isolation function of the drain-down path, complying with both the EGM guidance and TSs at 5:31 p.m. on October 31, 2013. PSEG entered this issue into the CAP as notification 20631218. The inspectors determined that the finding was of very low safety significance (Green) in accordance with NRC IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process," because the finding did not represent a finding that required quantitative assessment.

ATTACHMENT: SUPPLEMENTARY INFORMATION

SUPPLEMENTARY INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

P. Davison, Site Vice President
E. Carr, Plant Manager
M. Bacca, Dosimetry Supervisor
M. Biggs, Maintenance Rule Coordinator
K. Bittner, Winter Readiness Coordinator
P. Bonnett, Senior Compliance Engineer
K. Breslin, Operations Shift Manager
E. Casuli, Manager, Plant Engineering
V. Chandra, Design Engineer
S. Connelly, System Engineer
R. Cummins, Supervisor, System Engineering
C. Dahms, Regulatory Assurance
S. Dennis, Operations Licensed Operator Requalification Exam Instructor
P. Duca, Senior Compliance Engineer
R. Ficarra, Senior Reactor Operator
T. Gingerich, Senior System Engineer
M. Kelly, Senior Program Engineer
R. Kocher, System Engineer
A. Kraus, Manager, Nuclear Environmental Affairs (NEA)
W. Kopchick, Director, Operations
S. Kugler, Supervisor, System Engineering
C. Johnson, Senior Program Engineer
T. Morin, Senior Compliance Engineer
T. Neufang, Rad Engineering Superintendent
A. Ochoa, Senior Engineer
M. Oliveri, NDE Superintendent
M. Pyle, Principal Engineer
G. Rich, Chemistry Manager
M. Rooney, System Engineer
J. Russell, Nuclear Environmental Specialist
S. Simpson, Manager, Regulatory Assurance
L. Sinclair, Maintenance Superintendent
R. Smith, System Engineer
C. Thompson, radiological Engineer, Hope Creek
C. Wend, Radiological Engineering
B. White, NDE Manager

LIST OF ITEMS OPENED, CLOSED, DISCUSSED, AND UPDATED

Opened/Closed

05000354/2013005-01	NCV	Failure to Follow Procedure for Configuration Control Activity Adversely Affected Unidentified Leakage in the Drywell (Section 1R15)
05000354/2013005-02	NCV	Failure to Follow the Primary Containment Closeout Procedure when Declaring the Drywell Ready for Power Operation (Section 1R20)
05000354/2013005-03	FIN	Inadequate Evaluation of Containment Vent Functionality (Section 4OA2)
05000354/2013005-04	FIN	Failure to Identify Adverse Trend Regarding Bailey Module and Auxiliary Card Failures (Section 4OA3)

Closed

05000354/2013-002-00	LER	Reactor Scram due to Degrading Condenser Vacuum (Section 4OA3)
05000354/2013-002-01	LER	Reactor Scram due to Degrading Condenser Vacuum (Section 4OA3)
05000354/2013-004-00	LER	Operations With A Potential To Drain The Reactor Vessel (OPDRV) Without Secondary Containment (Section 4OA3)
05000354/2013-006-00	LER	Operations With A Potential To Drain The Reactor Vessel (OPDRV) Without Secondary Containment Operable (Section 4OA3)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Procedures
HC.OP-AR.GQ-0001, Intake Structure HVAC Local Panel 1EC581, Revision 9
HC.OP-SO.AP-0001, Condensate Storage and Transfer System Operation, Revision 37
HC.OP-SO.GM-0001, Diesel Area Ventilation System Operation, Revision 20
HC.OP-SO.GD-0001, Fire Pump House Ventilation System Operation, Revision 0
HC.OP-GP.ZZ-0003, Station Preparations for Winter Conditions, Revisions 28
OP-AA-108-111-1001, Severe Weather and Natural Disaster Guidelines, Revision 9
OP-SO.GD-001, Fire Pump House Ventilation System Operation, Revision 0
SH.FP-TI.FP-0001, Freeze Prevention and Winter Readiness of Fire Protection Systems, Revision 4
WC-AA-107, Seasonal Readiness, Revision 12
WC-AA-107, Seasonal Readiness, Revision 13

Notifications

20463793	20576019	20622914	20479131	20585973	20625563
20523017	20620763	20630707			

Orders

30244612	60110798	60112425	60105459	60111193	60112426
60109596	60112398	60112963			

Miscellaneous
2013 Hope Creek Winter Seasonal Readiness Affirmation dated October 1, 2013

Section 1R04: Equipment Alignment

Procedures
HC.OP-ST.BC-0001(Q), RHR System Piping and Flow Path Verification – Monthly, Revision 22
HC.OP-SO.BC-0002(Q), Decay Heat Removal Operation, Revision 31
HC.OP-ST.EA-0001, Service Water Flow Path Verification – Monthly, Revision 11
HC.MD-ST.PB-0003(Q), Class 1E 4.16 kV Feeder Degraded Voltage Monthly Instrumentation Channel Functional Test, Revision 26
HC.OP-AB.HVAC-0001(Q), HVAC, Revision 9
HC.OP-SO.PB-0001(Q), 4.16 kV System Operation, Revision 29
HC.OP-ST.KJ-0008(Q), Integrated Emergency Diesel Generator 1DG400 Test – 18 months, Revision 48
HC.OP-ST.ZZ-0001(Q), Power Distribution Line-up Weekly, Revision 36

Notifications

20325469	20622694	20627730	20634063	20593568	2062734
20634061					

Orders

30107437	30222091	70072347

Drawings
E-0069, Sheet 001, Revision 8
E-3060, Sheet 001, Revision 16
M-10-1, Sheet 1, Service Water, Revision 54
M-10-1, Sheet 2, Service Water, Revision 42
M-24-0, Sheet 3, Service Water Hypochlorination, Revision 38
M-51-1, Sheet 1, Residual Heat Removal, Revision 43
M-51-1, Sheet 2, Residual Heat Removal, Revision 40
M-81-1, Sheet 1, Misc Structures & Yard Bldgs. Air Flow Diagram, Revision 6

Miscellaneous
Hope Creek Operator Narrative Logs, 12/18-20/13
Operator’s Risk Evaluation for Hope Creek, 12/19/13

Section 1R05: Fire Protection

Procedures
FRH-II-132, Hope Creek Pre-Fire Plan, Turbine Building, Elevation: 102'-0", Revision 3
FRH-II-435, Hope Creek Pre-Fire Plan, Steam Tunnel, RCIC, HPCI, Pipe Chases, CRD Removal & Repair Area, Elevation: 102'-0", Revision 4
FRH-II-471, Hope Creek Pre-Fire Plan, Refuel Floor, Elevation: 201'-0", Revision 4
FRH-II-563, Hope Creek Pre-Fire Plan, Control Area HVAC Equipment Rooms, Elevations: 155-3" & 175'-0", Revision 6

Section 1R07: Heat Sink Performance

Procedures
ER-AA-340-1002, Service Water Heat Exchanger and Component Inspection Guide, Revision 5
HC.OP-FT.EA-0001, Validating SSWS Flow Through SACS HXs, Revision 15

Notifications

20625096	20626216	20631620	20625130	20626367

Orders

30158631	30214169	30259412	30256199
30158800	30214170	30251679	30249946

Miscellaneous
EA-0033, Biofouling Monitoring and Trending Calculation, Revision 0
EG-0047, HCGS Ultimate Heat Sink Temperature Limits – EPU, Revision 5
PSEG Letter LR-N97411 (E. C. Simpson) to USNRC regarding Update on the Implementation of Commitments Made in Response to Generic Letter 89-13, Hope Creek Generating Station, Facility Operating License NPF-57, Docket No. 50-354, dated August 1, 1997
USNRC Letter (D. H. Jaffe) to PSEG (L. R. Eliason) regarding Change to Commitments Associated with Generic Letter (GL) 89-13, "Service Water System Problems Affecting Safety-Related Equipment," July 18, 1989, for Hope Creek Generating Station (TAC No. M99369), dated September 19, 1997

Section 1R08: In-Service Inspection Activities

Procedures
54-ISI-363-007, Remote Underwater In-Vessel Visual Inspection of Reactor Pressure Vessel Internals, Components, and Associated Repairs in Boiling Water Reactors
54-ISI-880-000, Encoded Phased Array Ultrasonic Examination of Dissimilar Metal Piping Welds, Revision C
OU-AA-335-002, Liquid Penetration Examination, Revision 3
OU-AA-335-003, Magnetic Particle Examination, Revision 3
OU-AA-335-018, Detailed and General VT-1 and VT-3 Visual Examination of ASME Class MC and CC Containment Surfaces and Components, Revision 5

Notifications

20625732	20626631

Drawings
HC-1036, Internal Core Spray Piping ISO, Revision 2
HC-1037, Core Spray Sparger Detail for A&B Spargers, Revision 3
HC-1038, Core Spray Sparger Detail for C&D Spargers, Revision 3
JP-VIP-41, Jet Pump VIP 41 Weld ID Layout
M-42-1, Sheet 1, Weld/Hanger Identification Figure Inservice Inspection Drawing, Revision 2
WUH/700.121721/1A/1,Low Pressure FW Heater No. LP-3A and LP-3C Horizontal, Revision 1

Miscellaneous
Containment Visual Examination Record, Summary 820013
Containment Visual Examination Record, Summary 820020
MT-13-002, Magnetic Particle Examination Report for RPV1-W20 Head to Flange
PT-13-002, Liquid Penetrant Examination Report for RPV1-WSB (1-8) Stabilizer Brackets
UT Examination Report for N5A Core Spray A Nozzle, Summary 100745
Weld Process Traveler 904823-FHW3A-001, FW Inlet Nozzle, Revision 0
Weld Process Traveler 904823-FHW3A-005, Steam Inlet Nozzle, Revision 0
Weld Process Traveler 904823-FHW3A-007, Drain Outlet Nozzle, Revision 0
Welding Procedure Specification (WPS) 1 MN-GTAW/SMAW-HT, Revision 3
WPS 5A MN-GTAW/SMAW, Revision 1
WPS 14 MN-GTAW/SMAW, Revision 1
Welding Performance Qualification for Welder A – M2235
Welding Performance Qualification for Welder B – M2069

Section 1R11: Licensed Operator Requalification Program and Licensed Operator Performance

Procedures
HC.OP-IO.ZZ-0003, Startup From Cold Shutdown to Rated Power, Revision 104
HC.OP-IO.ZZ-0004, Shutdown From Rated Power To Cold Shutdown, Revision 98
HU-AA-1211, Pre-Job Briefings, Revision 11
OP-AA-101-111-1004, Operations Standards, Revision 4

Miscellaneous
Scenario Guide (SG)-678, Loss of 10A403 Bus with HVAC-001 entry/LOP with EDG Malfunction/LOCA with HPCI Failure, dated December 11, 2013

Section 1R12: Maintenance Effectiveness

Procedures
ER-AA-310, Implementation of the Maintenance Rule, Revision 11
ER-AA-310-1003, Maintenance Rule – Performance Criteria Selection, Revision 5
ER-AA-310-1004, Maintenance Rule – Performance Monitoring, Revision 9
ER-AA-310-1004, Maintenance Rule – Performance Monitoring, Revision 10
ER-AA-310-1005, Maintenance Rule – Dispositioning Between (a)(1) and (a)(2), Revision 8
ER-AA-310-1005, Maintenance Rule – Dispositioning Between (a)(1) AND (a)(2), Revision 9
ER-HC-310-1009, Maintenance Rule System Function and Risk Significant Guide, Revision 8
ER-HC-310-1009, Maintenance Rule System Function and Risk Significant Guide, Revision 9
ER-HC-310-1009, Maintenance Rule System Function and Risk Significant Guide, Revision 10
ER-AA-310, Implementation of the Maintenance Rule, Revision 11
ER-AA-310-1005, Maintenance Rule – Dispositioning Between (a)(1) AND (a)(2), Revision 9

ER-HC-310-1009, Maintenance Rule System Function and Risk Significant Guide, Revision 10
HC.FP-PM.QB-0039, Appendix "R" Standby Self-Contained 8 Hour Battery Powered Emergency Light Unit Inspection and Preventive Maintenance, Revision 6
HC.FP-ST.QB-0039, Standby Self Contained 8 Hour Battery Powered Emergency Light Unit Test and Inspection, Revision 8
HC.FP-ST.QB-0070, Standby Self Contained 8 Hour Battery Powered Emergency Light – 8 Hour Functional Test, Revision 6
HC.OP-IS.BC-0101(Q), Residual Heat Removal Subsystem A Valves, Revision 34
HC.OP-IS.BC-0105(Q), Residual Heat Removal System Valves – Cold Shutdown – Inservice Test, Revision 32
LS-AA-120, Issue Identification and Screening Process, Revision 11
LS-AA-125, Corrective Action Program, Revision 16
MA-AA-716-004, Conduct of Troubleshooting, Revision 12

<u>Notifications</u> (*NRC identified)

20458951	20536872	20580607	20584953
20490729	20536952	20580678	20584955
20491608	20536995	20580679	20585034
20492989	20536996	20580680	20585035
20492990	20537135	20580681	20585036
20494244	20537136	20580682	20585037
20497524	20537138	20580683	20585038
20498141	20537139	20580684	20585062
20502072	20537140	20580747	20585063
20502430	20537224	20580748	20585064
20502431	20537225	20580749	20585613
20502432	20539713	20580750	20585614
20503347	20539974	20580751	20585787
20506316	20548106	20580752	20585798
20506317	20548112	20580754	20585799
20506779	20549369	20580755	20585800
20511383	20563390	20580756	20585801
20511532	20563391	20580758	20585802
20511533	20568133	20580759	20585804
20511534	20569803	20580760	20585805
20511700	20573442	20580761	20585806
20511702	20575374	20580762	20585807
20511703	20575376	20580763	20585808
20511704	20575377	20580764	20585809
20511913	20575378	20580765	20585831
20511918	20575500	20584942	20585833
20511969	20575501	20584943	20585930
20512967	20576144	20584944	20586061
20512968	20576149	20584945	20586307
20513840	20580174	20584946	20596317
20513944	20580175	20584947	20597767
20516838	20580180	20584948	20602350
20522564	20580181	20584950	20602498
20536870	20580605	20584951	20604944
20536871	20580606	20584952	20607448

20608179	20619998	20622686	20627182
20614188	20620577	20622971	20627783
20614619	20620578	20623228	20632581
20619818	20620579	20623498	20632636
20619913*	20620580	20623503*	
20619997	20620683	20623744	
20632897			
20632925			

Drawings
E-6108, Sheet 2, Electrical Schematic Diagram, Residual Heat Removal System, Shutdown Cooling Inbd. Isol. Vlv. 1HV-F009, Revision 10
M-051, Sheet 1, Residual Heat Removal, Revision 43
M-55-1, High Pressure Coolant Injection, Revision 40
M-56-1, HPCI Pump Turbine, Revision 32

Orders

60099151	60113568	70019623	70143140
60113481	60114400	70145914	70152218

Miscellaneous
Hope Creek Expert Panel Meeting Minutes, MTG-2013-00024, dated September 23, 2013
PM 40023089

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures
ER-AB-331-1006, BWR Reactor Coolant System Leakage Monitoring and Action Plan, Revision 0
HC.MD-CM.ZZ-0013(Q), Electrically Backseating Motor-Operated Valve Remotely from a Motor Control Center, Revision 0
HC.OP-AB.CONT-0006(Q), Drywell Leakage, Revision 8 (4/16/13)
HC.OP-GP.ZZ-0005(Q), Drywell Leakage Source Detection, Revision 9
OP-AA-108-116, Protected Equipment Program, Revision 8
OU-AA-101-1006, Outage Management Risk and Impact Assessment, Revision 2
OU-HC-105, Shutdown Safety Management Program - Hope Creek Annex, Revision 3
WC-AA-105, Work Activity Management, Revision 2

Notifications

20627220	20627342	20627440	20628167	20629522	20630428
20630906	20631179	20630154	20630429	20631084	

Orders
60104388

Miscellaneous
BWR0G-06032, BWROG Reactor Coolant System Leakage Monitoring Best Practice Guideline, October 27, 2006
HC 13-013 (OTDM), Drywell Floor Drain Sump Unidentified Leakage, 11/19/13
HC 13-014 (ACM), Post-RF18 Drywell Leakage, 11/19/13

Hope Creek Shutdown Risk Status Sheet, November 3, 2013
MA-AA-716-004, Att. 4, Troubleshooting Control Form Number 1 (20629522), Revision 12
ORAM-SENTINEL – Electrical Power, All Other, September 25, 2013
ORAM-SENTINEL – Shutdown Cooling, Gate Out, Decay Heat, September 25, 2013
Protected Equipment Log –MCA/BS2 Switchyard Window – Plant 7, October 15, 2013
RF18 Shutdown Safety Management Plan P1.23a, Revision 2

Section 1R15: Operability Determinations and Functionality Assessments

Procedures
HC.IC-GP.ZZ-0114, Transmitter Isolation/Restoration Sensitive Rack Instrumentation Instrument Rack 10C-027 – RPV Channel B, Revision 16
HC.OP-AB.RPV-0003, Recirculation System / Power Oscillations, Revision 27
HC.OP-IO.ZZ-0009, Refueling Operations, Revision 38
HC.OP-IS.BJ-0001, HPCI Main and Booster Pump Set – 0P204 and 0P217 – Inservice Test, Revision 62
HC.OP-SO.BC-0002, Decay Heat Removal Operation, Revision 31
HC.OP-SO.BG-0001, Reactor Water Cleanup System, Revision 57
HC.OP-SO.BJ-0001, High Pressure Coolant Injection System Operation, Revision 48
HC.OP-SO.BC-0003, RHR Alternate Cooling Modes, Revision 6
MA-AA-716-230-1001, Oil Analysis Interpretation Guideline, Revision 9
OP-HC-108-102, Management of Operations with the Potential to Drain the Reactor Vessel, Revision 2
OU-HC-105, Shutdown Safety Management Program – Hope Creek Annex, Revision 3

Notifications

20518456	20623803	20625603	20629387
20608121	20623804	20625716	20631218
20622306	20623805	20625727	20631819
20622443	20623806	20625924	20631904
20623802	20624340	20628041	20632003

Orders

30227966	60111605	70127704	70161398
30229696	60113104	70135995	80080969
50149960	70042003	70159288	80110848
50158908	70090491	70160136	
50161033	70103591	70160514	

Miscellaneous
EGM 11-003, Dispositioning BWR License Nonconformance with Technical Specification Containment Requirements During OPDRVs, Revision 1
Shutdown Safety Evaluation and Approval for RF18, 10/12/13
R18 OPDRV Contingency Plans
10855-D3.38, Design, Installation and Test Specification for High Pressure Coolant Injection System for the Hope Creek Generating Station, Revision 9

Section 1R19: Post-Maintenance Testing

Procedures
HC.IC-FT.SN-009(Q), ADS and Safety Relief Valve Operability Test, Revision 5a
HC.IC-LC.AF-0007, Moisture Separator Drain Tank Level Tuning, Revision 2
HC.MD-CM.KJ-0020(Q), Diesel Generator Speed/Load Control System Alignment 2301A Governor System, Revision 1
HC.OP-AR.ZZ-0027(Q), CRIDS Computer Points Book 8 D4800 thru D5326, Revision 12
HC.OP-AR.ZZ-0014(Q), Overhead Annunciator Window Box D3, Revision 35
HC.OP-AR.ZZ-0015(Q), Overhead Annunciator Window Box E1, Revision 26
HC.OP-SO.BB-0002(Q), Reactor Recirculation System Operation, Revision 97
HC.OP-ST.KJ-0001(Q), Emergency Diesel Generator 1AG400 Operability Test – Monthly, Revision 78
MA-AA-716-100, Maintenance Alterations Process, Revision 11
MA-AA-723-303, Inspection/Instructions for Crimping and Lugging of Wiring, Revision 4
OP-AA-300, Reactivity Management, Revision 3
SH.IC-TI.ZZ-0001(Q), Electronic Soldering/Desoldering, Revision 5

Notifications

20453135	20626555	20630111	20632298
20621894	20626559	20630140	20632456
20622933	20626579	20630902	20632564
20623532	20626670	20631820	20632846
20626068	20626677	20631909	20632933
20626372	20629910	20631913	20632935
20626493	20629924	20631940	20633078
20626494	20629976	20632021	

Orders

30228881	50136953	60114166	80110856
30229045	50149551	60114285	
50123107	60089103	60114286	
50123145	60104674	80076662	

Drawings
10855-I-P-AC-02, Moisture Separator A Isometric, Revision 17
10855-I-P-AC-05, Moisture Separator B Isometric, Revision 15
C-2090, Maximum Allowable Deflections for ‘A’ & ‘B’ Moisture Separator Drain Lines, Revision 2
M-01-1, Sheet 1, Main Steam, Revision 52
M-02-1, Sheet 1, Extraction Steam, Revision 29

Miscellaneous
60089330, DEH-100031, Excessive Vibration of Piping Associated with Valve H1AC-1ACLV-1039A Full Open
80106575, DEH-120180, Excessive Vibration of Piping Associated with Valve H1AC-1ACLV-1039B Full Open
80110871, (DEH-13-0289) Technical Evaluation to Determine the Maximum Power Level to Restore Moisture Separator Normal Drain Control
HC 13-015, OTDM for Returning the Moisture Separator to Service without a Failure Mode Being Identified for Dump Valve Level Control

HC 2013-53, Standing Order for Moisture Separator Dump Valve Controls
LER 354/04-010, Manual Reactor Scram Due to Moisture Separator Re-heater Drain Line Failure

Section 1R20: Refueling and Other Outage Activities

Procedures
ER-AB-331-1006, BWR Reactor Coolant System Leakage Monitoring and Action Plan, Revision 0
HC.IC-LC.AF-0007, Moisture Separator Drain Tank Level Tuning, Revision 2
HC.OP-AB.RPV-0001, Reactor Power, Revision 13
HC.OP-AR.ZZ-0014, Overhead Annunciator Window Box D3, Revision 35
HC.OP-AR.ZZ-0027, CRIDS Computer Points Book 8 D4800 Thru D5326, Revision 13
MA-AA-716-004, Conduct of Troubleshooting, Revision 12
OP-AA-108-108, Unit Restart Review, Revision 11
OP-HC-108-114-1001, Hope Creek Post-Trip Data Collection Guidelines, Revision 6
OU-HC-105, Shutdown Safety Management Program – Hope Creek Annex, Revision 3
HC.OP-GP.ZZ-0002, Primary Containment Closeout, Revision 15
HC.OP-IO.ZZ-0001, Refueling To Cold Shutdown, Revision 31
HC.OP-IO.ZZ-0003, Startup From Cold Shutdown To Rated Power, Revision 104
HC.OP-IO.ZZ-0004, Shutdown From Rated Power To Cold Shutdown, Revision 98
HC.OP-IO.ZZ-0005, Cold Shutdown To Refueling, Revision 37
HC.OP-IO.ZZ-0009, Refueling Operations, Revision 38
HC.OP-SO.BC-0003, RHR Alternate Cooling Modes, Revision 6
LS-AA-106, Plant Operations Review Committee, Revision 5
OP-AA-108-102, Management of Operations with the Potential to Drain the Reactor Vessel, Revision 2
OP-AA-108-108, Unit Restart Review, Revision 11
OP-AA-108-108-1001, Drywell/Containment Closeout, Revision 3
OP-AA-108-114, Post Transient Review, Revision 4
OP-HC-108-114-1001, Hope Creek Post-Trip Data Collection Guidelines, Revision 6
OP-HC-108-102, Management of Operations with the Potential to Drain the Reactor Vessel, Revision 2
OU-AA-101-1006, Outage Management Risk and Impact Assessment, Revision 2
OU-HC-105, Shutdown Safety Management Program – Hope Creek Annex, Revision 3

Notifications (*NRC-identified)

20618492	20628404	20631913	20632621
20621211	20629147*	20631940	20632633
20623162	20629240	20632021	20632694
20623199	20629634	20632407	20632780
20625727	20631218	20632456	20632935
20626118	20631351	20632457	20633244
20626153	20631819	20632501	
20627091	20631820	20632542	
20628041	20631904	20632581	

Drawings
M-01-1, Sheet 1, Main Steam, Revision 52
M-02-1, Sheet 1, Extraction Steam, Revision 29

Orders

60114285	70127704	70161047	80110871	70103591
70135995	80110470			

Miscellaneous

REMA 2013-0102, Reactivity Maneuver Plan for Shutdown to RF18, Revision 0
HRE 2013-0119, Startup Package for December 2013 Forced Outage, dated December 2, 2013
OTDM HC 13-015, Decision to return the moisture separator to service without identifying why the dump failed to control level, dated December 9, 2013
Shutdown Risk Assessment for Forced Outage F133, dated December 6, 2013

Section 1R22: Surveillance Testing

Procedures

HC.OP-IS.AB-0102, Main Steam System Valves – Cold Shutdown – Inservice Test, Revision 23
HC.OP-IS.BC-0002, CP202, C Residual Heat Removal Pump In-Service Test, Revision 43
HC.OP-IS.EA-0003, C Service Water Pump – CP502 – Inservice Test, Revision 54
HC.OP-IS.KJ-0007, Integrated Emergency Diesel Generator 1CG400 Test – 18 Months, Revision 46
HC.OP-IS.ZZ-0001, Inservice System Leakage Test of the Reactor Coolant Pressure Boundary, Revision 43
HC.OP-LR.FC-0002, Containment Isolation Valve Type C Leak Rate Test 1FCHV-F084 – RCIC Steamline Vacuum Relief Penetration #204, Revision 3
HC.OP-SO.KJ-0001, Emergency Diesel Generator Operation, Revision 65
HC.OP-ST.BH-0002, Standby Liquid Control (SLC) Flow Test – 18 Month, Revision 28
HC.OP-ST.KJ-0004, Emergency Diesel Generator 1DG400 Operability Test – Monthly, Revision 76
PP-AA-3001, Position Paper on Preconditioning, Revision 0

Drawings

M-49-1, Reactor Core Isolation Cooling, Revision 18

Notifications (*NRC identified)

20614812	20626959	20627840	20634400
20624453	20626960	20627841	20634621
20626785	20627157	20633458	
20626786	20627159	20633585 *	

Orders

30227429	50160964	60096119	50149612	50161478	80111008

Miscellaneous

ACM 13-016, Monitoring of CP202 RHR pump noise, dated December 27, 2013

Section 1EP4: Emergency Action Level and Emergency Plan Changes

Procedures

EP-HC-111-100, “Hope Creek Event Classification Guide,” Revision 00
EP-HC-111-200, “EAL Technical Basis Document,” Revision 00

Section RS01: Access Control to Radiologically Significant Areas

Procedures
RP-AA-460, Rev. 15 and 16, Control for High and Very High Radiation Areas
RP-AA-463, Rev. 3, High Radiation Area Key Control
RP-AA-15, Rev. 2, Radioactive Contamination Control Description
HC.RP-TI.ZZ-0105(Q), Rev. 22, radiation Protection Shift Duties and Responsibilities

Documents
High Radiation Area Key Inventory
Radiological Survey data (various)
Corrective Action Documents (various)
Dosimeter - NVLAP certification data
Contamination Control – Personnel Contamination Data
Audit NOSA HPC-1308, October 10, 2013

Section RS02: Occupational ALARA Planning and Controls

Procedures
RP-AA-400, Rev. 6, ALARA program
RP-AA-401, Rev. 12, Operational ALARA Planning and Control
RP-AA-403, Rev. 3, Administration of the Radiation Work Permit Program

Documents
Emergent Dose data (2013)
Five year ALARA Plan
Corrective Action Documents (various)

Section RS03: In-plant Airborne Radioactivity Control and Mitigation

Procedures
RP-AA-825, Rev. 4, Maintenance, Care and Inspection of Respiratory Protection Equipment
NC.RP-TI.ZZ-403(Q), Rev. 3, Operation of Breathing Air System
RP-AA-440, Rev. 10, Respiratory Protection Program

Documents
Occupational Dose Summary
Radiological Source Term Data – 10 CFR 61 waste stream report
Airborne Radioactivity Intake Assessments
Corrective Action Documents (various)

Section RS04: Occupational Dose Assessment

Procedures
RP-AA-203-1001, Rev. 7, Personnel Exposure Investigations
RP-AA-220, Rev. 8, Bioassay Program
RP-AA-221, Rev. 4, Radiation Protection Whole Body Count (WBC) and WBC data Review
RP-AA-222, Rev. 6, Methods for Estimating Internal Exposure from In-Vivo and In Vitro
RP-AA-223, Rev. 0, Effective Dose Equivalent
RP-AA-270, Rev. 6, Prenatal Radiation Exposure

Documents
Post Outage Report
EPD/TLD –Error Resolution Report
Positive Whole Body Count Data (various)
Positive Skin Contamination Data (various)
Radiation Protection Technical Bases Document- Plant Radionuclide Mix Evaluation for Dosimetry Performance
Exposure Control and Dose Records (various)
General Source Term Data
Personnel Contamination Event Logs
Personnel Intake Investigations
Corrective Action Documents (various)

Section RS05 Radiation Monitoring Instrumentation

Procedures
NC.CH-RC.ZZ-2575(Q), Rev. 3, Gamma Spectroscopy System Calibration
RP-AA-220, Rev. 7, Bioassay Program
RP-AA-221, Rev. 3, Whole Body Count Data Review
RP-AA-222, Rev. 5, Methods for Estimating Internal Exposure from In Vivo and In Vitro Bioassay data
RP-AA-224, Rev. 1, Evaluation of Bioassay Data
CY-AA-130-205, Rev. 0, Radiochemistry Quality Control

Documents
Whole Body Counter Calibration Data (March 2013)
Intra-laboratory Quality Assurance data
Gamma Spectroscopy System Calibration Data
Vendor Laboratory Quality Assurance Data

Section RS06 Radioactive Gaseous and Liquid Effluent Treatment

Procedures
SC.CH-TI.ZZ-0180(Q), Rev. 66, Sample Schedule and Chemistry Specification
CY-AA-130-200, Rev. 9, Chemistry Quality Control

Documents
Chemistry Sampling Data
Meteorological Data

Section 2RS7 Radiological Environmental Monitoring Program

Procedures
EN-AA-170-000, Rev. 0, Radioactive Effluent and Environmental Monitoring Program
EN-AA-170-1001, Rev. 1, REMP Vendor Dosimetry and Laboratory QA program

Documents
2011, 2012 Annual Radioactive Environmental and Effluent Release Reports
Land Use Census

Section 2RS8: Radioactive Solid Waste Processing and Radioactive Material Handling, Storage and Transportation

Procedures
Lesson Plan NRP9902RMATC-02, Rev 1, Radioactive Materials Shipping
RP-AA-605, Rev 1, 10 CFR 61 Program
RP-AA-605-1001, Rev 1, Evaluation of 10 CFR 61 Sample Results
RW-AA-100, Rev 8, Process Control Program for Radioactive Wastes
Teledyne Brown Engineering report of Analysis for 10 CFR 61, waste streams: dry active waste; bead resin; powdex resin; condensate prefilter septa; clean-up phase separators
Radioactive Material Shipments: 13-023; 13-025; 13-073; 13-104; 13-113

Notifications
20545849
20559247
20577746
20592404
20615708
20615873

Miscellaneous
NOSA-HPC-12-04, Chemistry, Radwaste, Effluents and Environmental Monitoring Functional Area Audit Report
NUPIC Mega Audit/Survey # 23201, EnergySolutions
Check-In Self –Assessment 70156023, Pubic Radiation Safety – Radioactive Solid Waste Processing and Radioactive Material Handling, Storage and Transportation – Pre NRC Inspection 71124

Section 4OA1, Performance Indicator Verification

Documents
2011, 2012 Annual Radioactive Effluent Release and Environmental Monitoring Reports
Corrective Action Documents (various)
Radiation Dose Data (Occupational, Public) (past four quarters)

Section 4OA2: Problem Identification and Resolution

Procedures
CC-AA-13, Margin Management, Revision 0
CC-AA-309, Control of Design Analyses, Revision 10
CC-AA-309-1001, Guidelines for Preparation and Processing Design Analyses, Revision 6
ER-AA-2003, System Performance Monitoring and Analysis, Revision 9
ER-AA-3002, Component Cross-System Monitoring & Component Health Reporting, Revision 3
HC.IC-CC.AB-0042, Main Steam - Division 1 Channels B21-F022A, F022B, F028A, and F028B MSIV Closure Logic A1 Trip, Revision 2
HC.IC-CC.KL-0003, Containment Instrument Gas Low Low Pressure 1KLPSLL-5132A, Revision 32
HC.IC-CC.SB-0005, RPS - Division 1 Channel C71-N005A Turbine Control Valve Fast Closure, Revision 9

HC.IC-CC.SE-0013, Nuclear Instrumentation System Channel A Average Power Range Monitor, Revision 32
HC.IC-FT.GK-0001, Control Room Emergency Filtration System Flow Measurements, Revision 10
HC.IC-FT.SK-0003, HPCI - Division 1 Steam Leak Detection Temperature Monitor 1SKXR-11501, Revision 16
HC.IC-FT.SN-0009(Q), ADS and Safety Relief Valve Operability Test, Revision 5
HC.MD-CM.AB-0006(Q), Main Steam Safety/Relief Valve Removal and Installation, Revision 24
HC.MD-ST.AB-0001, MSIV Closure Trip Channel 18 Month Calibration, Revision 26
HC.MD-ST.AB-0003, Safety Relief Valve Discharge Piping Vacuum Breaker In-Place Setpoint Test, Revision 2
HC.MD-ST.GK-0001, Control Room Emergency Filtration System 18 Month Heater Test, Revision 14
HC.MD-ST.GS-0001, Torus to Drywell Vacuum Relief Valve 18 Month Testing, Revision 11
HC.MD-ST.GS-0002, Reactor Building to Torus Vacuum Relief Valve 18 Month Testing, Revision 11
HC.MD-ST.GU-0001, FRVS Recirculation Unit 18 Month Heater Test, Revision 15
HC.MD-ST.GU-0002, FRVS Ventilation Unit 18 Month Heater Test, Revision 10
HC.MD-ST.PJ-0002, 250 Volt Quarterly Battery Surveillance, Revision 31
HC.MD-ST.PJ-0008, 250 Volt Station Batteries 18 Month Service Test Using BCT-2000 with Windows Software and Associated Surveillance Testing, Revision 5
HC.MD-ST.PJ-0012, 10-A-404 Class 1E 4.16 KV 18 Month Vital Bus Loss of Voltage Instrumentation Channel Calibration & Functional Test, Revision 12
HC.MD-ST.SE-0002, Tip System Shear Valve Explosive Squib 18 Month Initiation Test, Revision 13
HC.MD-ST.ZZ-0006, Recirc Motor Generator Field Breaker (AKR-NB30F) P.M., Revision 22
HC.MD-ST.ZZ-0009, Motor Operated Valve Thermal Overload Protection Surveillance, Revision 20
HC.MD-ST.ZZ-0012, Masterpact Low Voltage Air Circuit Breaker Inspection and Preventive Maintenance, Revision 9
HC.OP-DL.ZZ-0004, Log 4 Reactor Building Data Log, Revision 13
HC.OP-DL.ZZ-0004, Log 4 Reactor Building Data Log, Revision 14
HC.OP-DL.ZZ-0004-F1, HC Reactor Bldg Log 4, Revision 14
HC.OP-DL.ZZ-0006-F1, HC Auxiliary Building Log 6, Revision 19
HC.OP-DL.ZZ-0007, Log 7 Yard Operators Log, Revision 46
HC.OP-EO.ZZ-0102, Containment Control, Revision 12
HC.OP-EO.ZZ-0318, Containment Venting, Revision 8
HC.OP-FT.BJ-0001, HPCI Main and Booster Pump Set - OP204 and OP217 - Functional Test, Revision 2
HC.OP-IS.AB-0102, Main Steam System Valves - Cold Shutdown - Inservice Test, Revision 23
HC.OP-IS.BC-0101, Residual Heat Removal Subsystem A Valves - Inservice Test, Revision 34
HC.OP-IS.BH-0001, Standby Liquid Control Pump - AP208 - Inservice Test, Revision 43
HC.OP-IS.EA-0102, Service Water Subsystem B Valves - Inservice Test, Revision 55
HC.OP-IS.GS-0101, Containment Atmosphere Control System Valves – Inservice Test, Revision 48
HC.OP-LR.KB-0001, Leakage Test of KB System Air Supply In-Line Check Valves, Revision 1
HC.OP-ST.SH-0001(Q), Accident Monitoring Instrumentation Channel Check – Monthly, Revision 34
HC.OP-SO.GM-0001(Q), “Diesel Area Ventilation System Operation,” Revision 19
HC.OP-ST.BD-0004, RCIC System Response Time and Flow Test - 18 Months, Revision 8

HC.OP-ST.GS-0003, Reactor Building/Suppression Chamber Vacuum Breaker Operability Test - Monthly, Revision 8
HC.OP-ST.GS-0004, Suppression Chamber/Drywell Vacuum Breaker Operability Test - Monthly, Revision 13
HC.OP-ST.KJ-0011, Diesel Fuel Oil Transfer Operability - 18 Months, Revision 6
HC.OP-ST.PB-0008, 4.16 KV Bus 10A404 Undervoltage Test & Return to Service - D Channel, Revision 16
HC.RA-ST.GK-0001, Control Room Emergency Filtration System Surveillance Test, Revision 7
HC.RE-ST.BF-0001, Control Rod Scram Time Surveillance, Revision 33
LS-AA-1006, NRC Cross-Cutting Analysis and Trending, Revision 2
LS-AA-120, Issue identification and Screening Process, Revision 12
LS-AA-125, Corrective Action Program, Revision 17
LS-AA-125-1001, Root Cause Evaluation Manual, Revision 9
LS-AA-125-1003, Apparent Cause Evaluation Manual, Revision 13
LS-AA-125-1006, Performance Improvement Integrated Matrix (PIIM), Revision 5
NWS-R-38, NWS Technologies Repair of Target Rock 2 Stage Main Steam Safety Relief Valves, Revision 1
NWS-T-25, NWS Test Procedure for Hope Creek Nuclear Station Target Rock 7567F 2 Stage Main Steam Safety Relief Valves, Revision 7

OP-AA-108-115, Operability Determinations, Revision 3
OP.HC-108-115-1002, Technical Specification Matric, Revision 10
OP.AA-111-101-1001, Use and Development of Operating Logs, Revision 5
PP-AA-3001, Position Paper on Preconditioning, Revision 0

Notifications (*NRC identified)

20636783*	20600825	20615840	20629882
20264907	20601216	20616442	20629943
20370021	20601611	20616582	20629995
20430611	20602340	20616758	20630278
20447156	20602676	20618492	20631351*
20447157	20602677	20618936	20631783
20483383	20604153	20620169	20632746
20525076	20604270	20622033	20632747
20554080	20613799	20624328	20632748
20559112	20614384	20627941	20632749
20562914	20614812	20627981	20632925
20570978	20614816	20627982	20633414
20575763	20614817	20628443	20633952
20592528	20615022	20629423	
20598096	20615125	20629880	

Drawings
H-88-0, Sheet 5, Diesel Area Diesel Generator Room Recirculation System, Revision 14
7567F-010, Target Rock Model 7567F 6x10 Relief Valve, Revision 9

Orders

30116792	60112100	70128080	70149571
30117966	60113785	70128407	70153312
30150950	70052075	70137157	70156307
30234962	70052848	70138789	70158815
30247542	70086624	70140986	80110005
40022920	70102111	70142155	80110470
40023040	70103136	70143235	80110809
40023062	70112981	70143647	80110866
50123119	70127765	70145115	

Miscellaneous

CD-783F, Nuclear Department Action Tracking System Response, dated 8/20/90
HC.MD-ST.AB-0003, Safety Relief Valve Discharge Piping Vacuum Breaker In-Place Setpoint Test (1ABPSV-F037A), performed 4/19/12
HC.MD-ST.AB-0003, Safety Relief Valve Discharge Piping Vacuum Breaker In-Place Setpoint Test (1ABPSV-F037C), performed 10/20/10
HC.MD-ST.AB-0003, Safety Relief Valve Discharge Piping Vacuum Breaker In-Place Setpoint Test (1ABPSV-F037E), performed 4/19/12
HC.MD-ST.GS-0002, Reactor Building to Torus Vacuum Relief Valve 18 Month Testing (PSV-5030), performed 9/18/12
HC.MD-ST.GS-0002, Reactor Building to Torus Vacuum Relief Valve 18 Month Testing (PSV-5032), performed 11/6/13
HC.OP-ST.GS-0003, Reactor Building/Suppression Chamber Vacuum Breaker Operability Test - Monthly, performed 9/22/13 and 11/6/13
HC.RA-ST.GK-0001, Control Room Emergency Filtration System Surveillance Test (1A-VH400), performed 6/26/13
HC.RA-ST.GK-0001, Control Room Emergency Filtration System Surveillance Test (1B-VH400), performed 3/27/13
Hope Creek Engineering PIIM Report 1st Cycle 2013 Presentation, dated 8/31/13
Hope Creek Work Week Schedule, week of 11/10/13 & 11/17/13
NRC Information Notice 96-24: Preconditioning of Molded-Case Circuit Breakers before Surveillance Testing, dated 4/25/96
NRC Information Notice 97-16: Preconditioning of Structures, Systems and Components before ASME Code Inservice Testing or Technical Specification Surveillance Testing, dated 4/4/97
NRC Information Notice 2012-16: Preconditioning of Pressure Switches before Surveillance Testing, dated 8/29/12
NRC Inspection Manual Part 9900: Technical Guidance, Maintenance - Preconditioning of Structures, Systems, and Components before Determining Operability, dated 9/28/98
NRC NUREG 1482, Guidelines for Inservice Testing at Nuclear Power Plants, Revision 2
Vendor Technical Document (VTD) PM150AQ-0013, 1-GS-PSV-4946A thru H Instruction Manual, Revision 7
VTD PM150Q-0050-02, 1-GS-PSV-5030 & 5032 Instruction Manual, Revision 5
DEH120045, SRV Setpoint Drift Root Cause Evaluation with MRC Comments Incorporated, Revision 0, dated 2/17/12
DEH120171, Hope Creek RF17 As-Found Testing Data of Removed SRVs, dated 5/10/12
DEH120171, Hope Creek RF18 As-Found Testing Data of Removed SRVs, dated 11/14/13

Form NVR-1, NWS Technologies Report of Repair/Replacement Certification Record of Nuclear Pressure Relief Devices, dated 8/7/12
Licensee Event Report 2009-002, dated 6/3/2009
Licensee Event Report 2010-002, dated 12/2/2010
Licensee Event Report 2010-002-01, dated 4/7/2011
Licensee Event Report 2012-004-01, dated 12/10/2012
VTD 322869-01, Safety Review for Hope Creek Generating Station Safety/Relief Valve Tolerance Analysis, dated April 1996

Section 4OA3: Follow-up of Events and Notices of Enforcement Discretion

Procedures
ER-AA-20, Equipment Reliability Program Description, Revision 2
ER-AA-2003, System Performance Monitoring and Analysis, Revision 9
ER-AA-2030, Conduct of Plant Engineering Manual, Revision
ER-AA-3002, Component Cross-System Monitoring & Component Health Reporting, Revision 3
HC.IC-AP.ZZ-0017, Bailey Module Reliability Program, Revision 0
HC.IC-GP.ZZ-0055, General Procedure Bailey Type 862 Module Functional Test, Revision 3
HC.MD-GP.ZZ-0014, Single Cell Battery Charging, Replacement and Jumpering, Revision 25
HC.OP-AB.ZZ-0173, Loss of 4.16kV Bus 10A404, D Channel, Revision 6
HC.OP-SO.BG-0001, Reactor Water Cleanup System Operation, Revision 57
HC.OP-ST.PB-0002, AC Power Supply Transfer Functional Test – 18 Months, Revision 11
LS-AA-125, Corrective Action Program, Revision 17
MA-AA-716-004, Conduct of Troubleshooting, Revision 12
OP-HC-108-102, Management of Operations with the Potential to Drain the Reactor Vessel, Revision 2
OU-AA-101, Refuel Outage Management, Revision
OU-AA-103, Att. 1, Shutdown Safety Evaluation and Approval [70103591 / 70127704 / 70135995], Revision 21

Notifications

20526739	20628041	20634064	20634439
20549908	20631218	20634073	2063444
20564014	20632003	20634146	
20616716	20634061	20634170	
20620263	20634063	20634247	

Drawings
M-43-1, Sheet 1, Reactor Recirculation System, Revision 34

Orders

30159979	60111335	70072347	70155514
30229697	60114688	70128730	70160136
60098764	70047997	70155234	70160514

Miscellaneous
OE20688, Tin Whiskers on Electric Circuit Cards
NRC Information Notice 05-25, Inadvertent Reactor Trip and Partial Safety Injection Actuation Due to Tin Whisker, dated 8/25/05

LER 2013-002-00(-01), Reactor Scram due to Degrading Condenser Vacuum
DEH120014, Technical Evaluation 80105570-0010 – Reactor Vessel Drain Down Time for CRD Maintenance Window
DEH120017, Technical Evaluation 80105570-0020 – Reactor Vessel Drain Down Time for LPRM Maintenance Window
EGM 11-003, Enforcement Guidance Memorandum – Dispositioning Boiling Water Reactor Licensee Noncompliance with Technical Specification Containment Requirements During Operations with a Potential for Draining the Reactor Vessel, Revision 1
Hope Creek H1R18 OPDRV Contingency Plans

LIST OF ACRONYMS

10 CFR	Title 10 of the *Code of Federal Regulations*
ACE	apparent cause evaluation
ADAMS	Agencywide Documents Access and Management System
ALARA	As Low As is Reasonably Achievable
AR	Action Request
ASME	American Society of Mechanical Engineers
CAP	corrective action program
CFR	*Code of Federal Regulations*
CW	circulating water
CY	calendar year
DAC	Derived Air Concentration
ECCS	emergency core cooling system
ED	Electronic Dosimeter
EDEX	Effective Dose Equivalent for External Exposure
EDG	emergency diesel generator
EGM	Enforcement Guidance Memorandum
EN	event notification
EPD	Electronic Personal Dosimeter
EPIP	Emergency Plan Implementing Procedures
FSAR	Final Safety Analysis Report
HEPA	High Efficiency Particulate Air
HCGS	Hope Creek Generating Station
HPCI	high pressure coolant injection
HRA	High Radiation Area
HX	heat exchanger
IMC	Inspection Manual Chapter
IP	inspection procedure
ISI	inservice inspection
IST	in-service test
IVVI	in-vessel visual inspection
kV	kilovolt
LER	licensee event report
LHRA	Locked High Radiation Area
LLD	Lower Limits of Detection
LOCA	loss of coolant accident
MDA	Minimum Detectable Activity
MS	main steam
MSHA	Mine Safety and Health Administration
MT	magnetic particle testing
NCV	non-cited violation
NDE	non-destructive examination
NIOSH	National Institute for Occupational Safety and Health
NOS	Nuclear Oversight
NRC	Nuclear Regulatory Commission
NSIR	Office of Nuclear Security and Incident Response
NVLAP	National Laboratory Accreditation Program

ODCM	Offsite Dose Calculation Manual
OPDRV	operation with the potential to drain the reactor vessel
P&ID	Piping and Instrument Diagram
PCM	Personnel Contamination Monitor
PCP	process control program
PD	Performance Deficiency
PI	Performance Indicators
PI&R	problem identification and resolution
PIIM	performance improvement integrated matrix
PM	Portal Monitor
PSEG	Public Service Enterprise Group
PT	liquid penetrant test
QA	quality assurance
RCA	Radiological Controlled Area
RCIC	reactor core isolation cooling
RCS	reactor coolant system
REMP	Radiological Environmental Monitoring Program
RETS	Radiological Effluents Technical Specification
RG	Regulatory Guide
RHR	residual heat removal
RPM	Radiation Protection Manager
RPV	reactor pressure vessel
RRP	reactor recirculation pump
RTP	rated thermal power
RWCU	reactor water cleanup
RWP	Radiation Work Permit
SACS	safety auxiliaries cooling system
SAM	Small Article Monitor
SCBA	self-contained breathing apparatus
SCCM	standard cubic centimeter per minute
SDC	shutdown cooling
SDP	Significance Determination Process
SLC	standby liquid control
SRV	safety relief valve
SSC	structure, system, or component
SSLM	solid state logic module
SSW	station service water
TLD	Thermoluminescent Dosimeter
TS	Technical Specification
TYRA	3-year Rolling Average
UFSAR	Updated Final Safety Analysis Report
UIL	unidentified leakage
UT	ultrasonic testing
VHRA	Very High Radiation Area
VT	visual testing
VTD	Vendor Technical Document
WBC	Whole Body Counter
WCD	work control document