UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
2100 RENAISSANCE BLVD., SUITE 100
KING OF PRUSSIA, PA 19406-2713

July 28, 2015

EA-15-146
EA-15-147

Mr. Robert Braun
President and Chief Nuclear Officer
PSEG Nuclear LLC – N09
P.O. Box 236
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION UNIT 1 – INTEGRATED INSPECTION REPORT AND EXERCISES OF ENFORCEMENT DISCRETION 05000354/2015002

Dear Mr. Braun:

On June 30, 2015, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Hope Creek Generating Station (HCGS). The enclosed inspection report documents the inspection results, which were discussed on July 16, 2015 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 one Severity Level IV non-cited violation (NCV) and one self-revealing finding of very low safety significance (Green). These findings were determined to involve violations of NRC requirements. 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.a of the NRC Enforcement Policy. If you contest any NCVs 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 to any finding, 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 HCGS.

Additionally, the inspectors reviewed Licensee Event Report (LER) and LER supplement, 50-354/2015-001-00 and -01, which described the details associated with a failed breaker control device for the ‘A’ Core Spray (CS) pump. The failed breaker control device resulted in the inoperability of multiple CS subsystems and could have prevented the fulfillment of a safety function. This issue constitutes a violation of NRC requirements, in that PSEG operated HCGS with multiple CS subsystems inoperable without taking actions to restore the subsystems to an operable status in accordance with Technical Specifications (TSs). However, the NRC concluded that the cause of the inoperability, a failed spring inside the sealed breaker control device that was still within the manufacturer’s recommended life span, was due to a manufacturing defect that could not have been identified during inspection and testing or avoided through management controls. Additionally, there is no previous operating experience of this type of failure at HCGS. Therefore, no performance deficiency associated with the violation was identified. The NRC performed a risk evaluation of the issue and determined it to be of very low safety significance. Based on these facts, I have been authorized, after consultation with the Director, Office of Enforcement, and the Regional Administrator, to exercise enforcement discretion, in accordance with NRC Enforcement Policy Section 2.2.4, “Exceptions to Using Only the Operating Reactor Assessment Program,” and Section 3.5 “Violations Involving Special Circumstances,” and refrain from issuing enforcement for this violation.

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 April 14, 15, 17, 20, 27 and 29, 2015, while all other TSs were met, HCGS conducted several OPDRVs without establishing secondary containment operability, which is a violation of TS 3.6.5.1, “Secondary Containment.” NRC issued EGM 11-003, “Enforcement Guidance Memorandum on Dispositioning Boiling Water Reactor (BWR) Licensee Noncompliance with Technical Specification 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, in accordance with NRC Enforcement Policy Section 2.2.4, “Exceptions to Using Only the Operating Reactor Assessment Program,” and Section 3.5 “Violations Involving Special Circumstances,” and refrain from issuing enforcement for the violation, subject to a timely LAR being submitted.

In accordance with Title 10 of the *Code of Federal Regulations* (CFR) 2.390 of the NRCs "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's Public Document Room or from the Publicly Available Records component of the NRC's Agencywide Documents Access Management System (ADAMS). ADAMS is accessible from the NRC website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Ho K. Nieh, Director
Division of Reactor Projects

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

Enclosure:
Inspection Report 05000354/2015002
w/Attachment: Supplementary Information

cc w/encl: Distribution via ListServ

R. Braun

In accordance with Title 10 of the *Code of Federal Regulations* (CFR) 2.390 of the NRCs "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's Public Document Room or from the Publicly Available Records component of the NRC's Agencywide Documents Access Management System (ADAMS). ADAMS is accessible from the NRC website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Ho K. Nieh, Director
Division of Reactor Projects

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

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	50-354
License No.	NPF-57
Report No.	05000354/2015002
Licensee:	Public Service Enterprise Group (PSEG) Nuclear LLC
Facility:	Hope Creek Generating Station (HCGS)
Location:	P.O. Box 236 Hancocks Bridge, NJ 08038
Dates:	April 1, 2015 through June 30, 2015
Inspectors:	J. Hawkins, Senior Resident Inspector S. Haney, Resident Inspector H. Gray, Senior Reactor Inspector M. Draxton, Project Engineer R. Nimitz, Senior Health Physicist R. Vadella, Reactor Engineer B. Fuller, Operations Engineer C. Bixler, Operator Licensing Assistant T. OHara, Reactor Engineer T. Burns, Reactor Inspector
Approved By:	Glenn T. Dentel, Chief Reactor Projects Branch 3 Division of Reactor Projects

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SUMMARY

IR 05000354/2015002; 4/01/2015 – 6/30/2015; Hope Creek Generating Station; In-Service Inspection Activities, Other Activities, and Follow-up of Events and Notices of Enforcement Discretion.

This report covered a three-month period of inspection by the resident inspectors and announced inspections performed by regional inspectors. Inspectors identified one Severity Level IV non-cited violation (NCV) and one finding of very low safety significance (Green), which was an NCV. The findings were determined to be violations of NRC requirements. 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 April 29, 2015. Cross-cutting aspects are determined using IMC 0310, "Components Within Cross-Cutting Areas," dated December 4, 2014. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy, dated February 4, 2015. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 5.

Cornerstone: Mitigating Systems

- Severity Level IV. During a review of recently issued operator licenses, the NRC identified an NCV of 10 CFR 50.9 associated with the licensee's failure to request a Generic Fundamentals Examination (GFE) waiver for a Senior Operator License applicant. Compliance was restored on May 4, 2015, when the licensee submitted a letter to the NRC which provided additional information concerning the issue. The Senior Reactor Operator (SRO) applicant had completed classroom instruction and successfully passed a licensee administered GFE on August 16, 2013, and had passed an NRC prepared GFE when previously licensed as a reactor operator at another utility. The applicant met the requirements to request a waiver to sit for the exam and would have been granted a waiver if it had been requested.

  The inspectors determined that traditional enforcement applied to this performance deficiency (PD), as the issue impacted the NRC's ability to perform its regulatory function. Specifically, the NRC relies upon the licensee to ensure all license applicants have completed the preparation requirements of NUREG-1021. The PD was determined to be Severity Level IV because it fits the SL-IV example of Enforcement Policy Section 6.4.d.1.a, "Violation Examples: Licensed Reactor Operators." This section states, "Severity Level IV violations involve for example …cases of inaccurate or incomplete information inadvertently provided to the NRC that does not contribute to the NRC making an incorrect regulatory decision as a result of the originally submitted information." Because the applicant met the requirements for a waiver and the waiver would have been granted if it had been requested, the performance deficiency did not cause the NRC to make an incorrect regulatory decision. The performance deficiency was screened against the Reactor Oversight Process (ROP) per the guidance of IMC 0612, Appendix B, "Issue Screening." No associated ROP finding was identified and no cross-cutting aspect was assigned. (Section 4OA5)

Cornerstone: Barrier Integrity

- Green. A self-revealing Green NCV of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions," was identified involving PSEG's failure to promptly identify and correct a condition adverse to quality. Specifically, PSEG did not identify and initiate a Corrective Action Process Notification Report for numerous tooling marks on the Reactor Coolant System (RCS) inlet piping connecting the Safety Relief Valves (SRVs) to the primary system following periodic removal and replacement. PSEG determined that the tooling marks could have resulted in stress risers on the RCS piping, making the pipe prone to cracking, and reduced the margin to the piping minimum wall thickness. PSEG's corrective actions included blending the tooling marks on all 14 SRV inlet pipes, verifying thickness above the minimum wall value, completing ultrasonic thickness measurements and magnetic particle surface examinations of the piping, and completing an RCS operational pressure test to verify the operability and functionality of the SRV inlet piping.

  This finding was more than minor because it was associated with the human performance attribute of the Barrier Integrity cornerstone and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system and containment) protect the public from radionuclide releases caused by accidents or events. The inspectors used IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, which states in the Barrier Integrity section that for all non-pressurized thermal shock issues, the inspectors should evaluate the issue under the initiating events cornerstone (Exhibit 1). Using Exhibit 1 for Transient Initiators, the inspectors determined that the finding was of very low safety significance (Green), because after a reasonable assessment of the degradation; the condition did not adversely impact RCS leakage or functionality of available Loss of Coolant Accident (LOCA) mitigation capabilities. Specifically, the SRV inlet piping safety-related function, relied upon for accident mitigation and pressure relief, remained operable. The inspectors determined this finding has a cross-cutting aspect in Human Performance, Work Management, because the organization did not implement a process of planning, controlling, and executing work activities such that nuclear safety is the overriding priority. The work process did not include the identification of risk (risk of the torque tool damaging the SRV pipe, and the failure to identify damage during inspections when performing maintenance on the SRV's) commensurate to the work and the need for coordination with different groups or job activities [H.5]. (Section 1R08)

REPORT DETAILS

Summary of Plant Status

The Hope Creek Generating Station began the inspection period at 100 percent of rated thermal power (RTP). On April 5, 2015, operators reduced power to approximately 94 percent RTP to remove the '6B' feedwater heater from service. Operators returned the unit to full power on the same day. The unit was manually scrammed on April 10, 2015 from 20 percent RTP to start Hope Creek's planned 19th refueling outage (H1R19). On May 9, 2015, the reactor mode switch was placed in start-up, criticality was reached later that day, and the unit was synchronized to the grid on May 12, 2015. The unit was returned to 100 percent RTP on May 15, 2015. The unit remained at or near full RTP for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01 – 2 samples)

.1 Readiness for Seasonal Extreme Weather Conditions

a. Inspection Scope

The inspectors performed a review of PSEG's readiness for the onset of seasonal high temperatures. The review focused on the safety auxiliaries cooling system (SACS) and station service water (SSW) system. The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR) and TS 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 verify that no unidentified issues existed that could challenge the operability of the systems during hot weather conditions. Documents reviewed for each section of this inspection report are listed in the Attachment.

b. Findings

No findings were identified.

.2 Summer Readiness of Offsite and Alternate Alternating Current (AC) Power Systems

a. Inspection Scope

The inspectors performed a review of plant features and procedures for the operation and continued availability of the offsite and alternate AC power system to evaluate readiness of the systems prior to seasonal high grid loading. The inspectors reviewed PSEG's procedures affecting these areas and the communications protocols between the transmission system operator and PSEG. This review focused on changes to the established program and material condition of offsite alternate AC power equipment. When required, the inspectors assessed whether PSEG established and implemented appropriate procedures and protocols to monitor and maintain availability and reliability

of both the offsite AC power system and the onsite alternate AC power system. The inspectors evaluated the material condition of the associated equipment by interviewing responsible PSEG personnel, reviewing the switchyard summer readiness letter, and walking down portions of the offsite and alternate AC power systems including the main transformers and the 500 kilovolt (kV) and 13.8 kV switchyards.

b. Findings

No findings were identified.

1R04 Equipment Alignment

.1 Partial System Walkdowns (71111.04 – 3 samples)

a. Inspection Scope

The inspectors performed partial walkdowns of the following systems:

- SRV piping and system configuration following the discovery of indications on the 'A' SRV inlet piping on April 15, 2015
- 'D' Emergency Diesel Generator (EDG) following governor replacement on May 5, 2015
- 'A', 'B', and 'C' SSW pumps during 'D' SSW pump planned maintenance on May 21, 2015

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, TSs, work orders (WOs), 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.

.2 Full System Walkdown (71111.04S – 1 sample)

a. Inspection Scope

On April 11, 2015, the inspectors performed a complete system walkdown of the 'A' Residual Heat Removal (RHR) system in its shutdown cooling alignment to verify the equipment lineup was correct. The inspectors reviewed operating procedures, surveillance tests, drawings, equipment lineup procedures, and the UFSAR to verify the

system was aligned to perform its required safety functions. The inspectors also reviewed electrical power availability, component lubrication and equipment cooling, hangar and support functionality, and operability of support systems. The inspectors performed field walkdowns of accessible portions of the systems to verify system components and support equipment were aligned correctly and 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. Additionally, the inspectors reviewed a sample of related condition reports and work orders to ensure PSEG appropriately evaluated and resolved any deficiencies.

b. Findings

No findings were identified.

1R05 Fire Protection

Resident Inspector Quarterly Walkdowns (71111.05Q - 6 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 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, in accordance with procedures.

- FRH-II-452, Reactor Water Cleanup (RWCU) pipe chase, Room 4505, on April 17, 2015
- FRH-II-415, Torus Compartment, Room 4102, on April 24, 2015
- FRH-III-114, Condensate Piping Area in the Turbine Building, Room 1112, on April 28, 2015
- FRH-III-151, ‘A’ and ‘B’ Reactor Recirculation Pump (RRP) Motor Generator rooms in the Turbine Building, Rooms 1516 and 1517, on April 28, 2015
- FRH-II-411, Torus Water Cleanup (TWCU) Pump, Room 4101, on May 5, 2015
- FRH-II-461, Standby Liquid Control Area, Room 4606, on June 22, 2015

b. Findings

No findings were identified.

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

a. Inspection Scope

The inspectors reviewed the 'A' RHR heat exchanger and the 'B1' and 'B2' SACS heat exchangers to determine their readiness and availability to perform their safety functions. The inspectors reviewed the design basis for the component and verified PSEG's commitments to NRC Generic Letter 89-13, "Service Water System Requirements Affecting Safety-Related Equipment." The inspectors observed actual performance tests for the heat exchangers and reviewed the results of previous inspections on the heat exchangers. 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 April 20 to April 24, 2015, the inspectors conducted a review of PSEG's implementation of in-service inspection (ISI) program activities for monitoring degradation of the RCS pressure boundary, risk significant piping and components, and containment systems for the Hope Creek Generating Station. The sample selection for this inspection was based on the inspection procedure objectives and risk priority of those pressure retaining components in these systems where degradation would result in a significant increase in risk. The inspectors observed in-process nondestructive examinations (NDE), reviewed documentation, and interviewed inspection personnel to verify that the nondestructive examination activities performed as part of Interval 3, of Period 3 of the Hope Creek ISI program during the Hope Creek 19th refueling outage, were conducted in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI, 2001 Edition, 2003 Addenda.

Nondestructive Examination (NDE) and Welding Activities

The inspectors performed direct observations of NDE activities in process or reviewed records of NDE listed below:

ASME Code Required Examinations

The inspectors performed a field observation of the ultrasonic testing (UT) examination of weld PRV1-W24F, a meridional weld on the Hope Creek Reactor Vessel Head. The inspectors observed the calibration of the testing equipment, reviewed the results of previous inspection of the weld and reviewed the testing procedure being used. After observing the technicians performing the inspection the inspectors reviewed the completed, approved ASME Code data sheets.

The inspectors performed a field observation of the UT of reactor vessel nozzle N8A to reactor vessel shell weld. The inspectors observed the calibration of the testing equipment, reviewed the results of a previous inspection of the weld and reviewed the testing procedure. The inspectors observed the technicians performing the inspection and reviewed the completed, approved ASME Code data sheets.

The inspectors observed the calibration of the phased array UT equipment used to examine dissimilar metal weld 14-A-46DM, nozzle N5A safe end to nozzle weld. The inspectors also reviewed the completed Examination Summary Sheet, Report No.169694 for the completed dissimilar metal weld examination.

The inspectors reviewed a sample of IWE General Visual inspection results from the Hope Creek Drywell. These inspections were clearly documented and complete. Inspected areas and components were described, and inaccessible areas were noted. There were no recordable indications noted in accessible areas.

The inspectors completed a field observation of the UT examination of the inner radius of reactor vessel head nozzle PRVI-N7IR. The inspectors observed the equipment calibration and reviewed the previous inspection results. The inspectors observed the scanning of the nozzle radius and reviewed the completed ASME data sheets.

The inspectors reviewed a sample of personnel certifications for NDE technicians performing ASME Code examinations and verified the inspections were performed in accordance with approved procedures and that the results were reviewed and evaluated by certified Level III NDE personnel.

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

The inspectors reviewed a sample of inspection records of visual examinations conducted on reactor vessel internals components during the 19th refueling outage. These inspections were carried out in accordance with the industry initiative under the Boiling Water Reactor Vessel and Internals Project (BWRVIP), In Vessel Visual Inspection (IVVI) Program. These inspections monitor and record the condition of the reactor vessel internal components. Specifically, the inspectors reviewed Visual Testing examination data records and reviewed the disposition of indications noted by the inspectors. The inspectors verified that the activities were performed in accordance with applicable examination procedures and industry guidance. All recorded indications were recorded and dispositioned by the NDE examiner and the licensee as acceptable for further service.

The inspectors reviewed the ultrasonic examination results of two dissimilar metal welds in the reactor water cleanup system. These welds are part of the industry initiative, BWRVIP-75, and guidance is provided by Nuclear Energy Institute (NEI 03-08) initiative to inspect dissimilar metal welds (in this case, carbon steel to stainless steel) susceptible to primary water stress corrosion cracking. The inspectors reviewed the data reports of the dissimilar metal welds and determined that the welds, Summary No. 109783, RWCU System Flow Element to Pipe Stub, and Summary No. 109782, RWCU System Pipe Stub to Flow Element, dissimilar metal weld, had been inspected. However, PSEG was unable to perform these inspections consistent with the guidance of Electric Power Research Institute (EPRI) Technical Report, Nondestructive Evaluation: Guideline for

Conducting Ultrasonic Examinations of Dissimilar Metal Welds, Revision 1, 300200091, Final Report, May 2013. PSEG was not able to perform encoded weld examinations. PSEG will report this as a deviation to NEI 03-08 on their outage report. Re-inspection with encoding is planned in the future. There were no reportable indications with the non-encoded results.

Review of Originally Rejectable Indications Accepted by Evaluation

During R18 and R19 there were no examples of originally rejectable indications which were accepted by evaluation.

Repair/Replacement Activities Including Welding Activities

The inspectors reviewed WO 60113722, to remove a blind flange from a containment penetration, which had failed to pass its leak rate test, and installed a new flange per procedure CC-AA-112, section 4.2. The inspectors verified the work was completed in accordance with ASME Code requirements.

The inspectors reviewed the replacement of a 4" spool piece welded to valve 1-BC-050-S05 and welded to valve H1BC-034, in the reactor building per WO 60118109. The inspectors reviewed the record of the welds and the dye penetrant testing and the final radiographic examination of the weld. The inspectors verified that this repair met the ASME Code requirements.

Identification and Resolution of Problems (71152)

The inspectors reviewed a sample of PSEG corrective action reports (Notifications), which identified NDE indications, deficiencies and other nonconforming conditions since the previous,18th refueling outage. The inspectors verified that nonconforming conditions were, in general, properly identified, entered into the corrective action program, characterized, evaluated, and corrective actions identified and completed.

b. Findings

Introduction. A self-revealing Green NCV of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions," was identified involving PSEG's failure to promptly identify and correct a condition adverse to quality. Specifically, PSEG did not identify and initiate a Corrective Action Process Notification Report for numerous tooling marks on the RCS inlet piping connecting the SRVs to the primary system following periodic removal and replacement. PSEG determined that the tooling marks could have resulted in stress risers on the RCS piping, making the pipe prone to cracking, and reduced the margin to the piping minimum wall thickness.

Description. At Hope Creek there are 14 target rock model 7567F two-stage, pilot-operated SRV's consisting of two assemblies: a pilot stage assembly and a main stage assembly. These two assemblies are directly coupled to the RCS piping via a bolted flanged connection. Half of these SRVs are replaced by PSEG every refueling outage. The SRVs provide for the pressure relief function or system depressurization at full rated flow with a set pressure range of 1025 to 1190 psig. The SRV inlet piping has a nominal wall thickness of 0.875 inches or 28/32 inches.

On April 15, 2015, PSEG NDE personnel were attempting to perform an ASME Code required ultrasonic examination of a weld on the ‘A’ SRV inlet piping, just below the bolted flange, when NDE personnel discovered tooling marks in the area of the weld preventing them from performing the weld examination. PSEG initiated Notification 20685195 documenting the presence of the tooling marks on the ‘A’ SRV and conducted an extent of condition inspection which further determined that each of the 14 SRV inlet pipes had similar tooling marks. PSEG determined that the tooling marks were the result of using a hard surfaced tool to de-torque the flange bolts. The deepest of these tooling marks was measured to be approximately 0.1 inches or 3/32 inches. PSEG also performed a historical search of the CAP to see if these marks had been previously identified and/or evaluated. On April 20, 2015, PSEG initiated Notification 20686147 stating that no previous identification or evaluation of the tooling marks could be found in the CAP.

PSEG Nuclear Maintenance Procedure, HC.MD-CM.AB-0006, Main Steam Safety/Relief Valve Removal and Installation, specifies the use of a torqueing tool to assist maintenance personnel in loosening the flange bolts. No evidence could be found to support prior evaluation of this torque tool’s potential effects on the SRV inlet piping.

Prior to PSEG blending the marks out, the inspectors questioned the PSEG engineering staff about the specified design minimum wall thickness of the SRV inlet piping that would be used to control the depth of the blending process. The PSEG staff initially considered a pressure of approximately 1000 psi, but after questions from the inspectors and consulting with the reactor vendor, PSEG staff used a minimum wall thickness value based upon a pressure of 1250 psi. This pressure corresponded to an SRV inlet piping minimum wall thickness of 0.52 inches.

While the completed blending of the deepest tooling marks did not reduce any of the SRV inlet piping below the acceptable minimum wall thickness of 0.52 inches, the blending did result in the RCS piping wall thickness margin above minimum wall thickness being reduced by approximately 34 percent in some locations. The inspectors observed that each use of the torque tool on the RCS piping likely caused unquantified degradation to the affected RCS piping. The inspectors’ review of PSEG’s technical evaluation, SRV work history, and procedures determined that these tooling marks should have been identified and evaluated as a condition adverse to quality by PSEG prior to April 2015, and as early as the first usage of the torque tool for SRV maintenance applications which started per HC.MD-CM.AB-0006 Revision 17 in October 2004. In addition, the inspectors identified that PSEG procedure, HC.MD-CM.0006, Sections 5.4 Valve Body Removal, 5.5 Valve and Piping Inspections and 5.6 Valve Body Installation contained supervisory hold points for maintenance supervision to verify work task completion. Specifically, the inspectors identified that Sections 5.5 and 5.6 required visual inspection of the SRV inlet and outlet piping as well as notes that any nicks, pits and grooves that are greater than 0.062 inches in depth are to be evaluated by the engineering staff.

Following the completion of blending all 14 SRV inlet piping, PSEG completed ultrasonic thickness measurements and performed a magnetic particle surface examination to ensure there were no surface cracks where the tooling marks had been identified. On May 12, 2015, PSEG successfully completed an RCS operational pressure test to verify the operability and functionality of the SRV inlet piping.

Analysis. The inspectors determined that PSEG's failure to identify and evaluate conditions adverse to quality associated with the tooling marks on the SRV inlet piping was a PD. The PD was more than minor because it was associated with the human performance attribute of the Barrier Integrity cornerstone and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system and containment) protect the public from radionuclide releases caused by accidents or events. The PD is also similar to examples 3.j and 3.k of NRC IMC 0612, Appendix E. The inspectors used IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated April 29, 2015, which states in the Barrier Integrity section that for all non-pressurized thermal shock issues, the inspectors should evaluate the issue under the initiating events cornerstone (Exhibit 1). Using Exhibit 1 for Transient Initiators, the inspectors determined that this finding was of very low safety significance (Green), because after a reasonable assessment of the degradation; the condition did not adversely impact RCS leakage or functionality of available LOCA mitigation capabilities. Specifically, the SRV inlet piping safety-related function, relied upon for accident mitigation and pressure relief, remained operable.

The inspectors determined this finding has a cross-cutting aspect in Human Performance, Work Management because the organization did not implement a process of planning, controlling, and executing work activities such that nuclear safety is the overriding priority. The work process did not include the identification of risk (risk of the torque tool damaging the SRV pipe, and the failure to identify damage during inspections when performing maintenance on the SRV's) commensurate to the work and the need for coordination with different groups or job activities [H.5].

Enforcement. 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," requires in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected. Contrary to the above, PSEG failed to promptly identify and correct a condition adverse to quality. Specifically, since October 2004, PSEG did not identify and initiate a Corrective Action Notification Report to identify the condition adverse to quality despite repeated use of the torque tool to loosen the SRV bolts, until April 2015.

PSEG's corrective actions included blending the tooling marks on all 14 SRV inlet pipes, verifying thickness above the minimum wall value, completing ultrasonic thickness measurements and magnetic particle surface examinations of the piping, and completing an RCS operational pressure test to verify the operability and functionality of the SRV inlet piping. Because this finding was of very low safety significance and because it was entered into PSEG's CAP as Notification 20687515, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000354/2015002-01, Failure to Identify and Correct a Condition Adverse to Quality Associated with Safety Relief Valve Inlet Piping)**

1R11 Licensed Operator Requalification Program

.1 Quarterly Review of Licensed Operator Requalification Testing and Training (71111.11Q – 1 sample)

a. Inspection Scope

The inspectors observed and reviewed licensed operator simulator training on June 15, 2015, which included plant restart activities during summer peak generation. The inspectors evaluated operator performance during the simulated plant restart and verified completion of 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 of the technical specification action statements entered by the shift. Additionally, the inspectors assessed the ability of the crew and 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 (71111.11Q – 1 sample)

a. Inspection Scope

The inspectors observed plant restart activities for planned refueling outage, H1R19, on May 9, 2015. The inspectors observed reactivity control briefings to verify that the briefings met the criteria specified in OP-AA-101-111-1004 “Operations Standards,” Revision 5 and HU-AA-1211, “Pre-Job Briefings,” Revision 12. 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.

1R12 Maintenance Effectiveness (71111.12Q – 2 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.

- ‘B’ RRP Motor Generator controller found broken on April 15, 2015
- Reactor Core Isolation Cooling (RCIC) pump failure to rotate during low pressure testing on May 9, 2015

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 technical specification requirements and inspected portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met.

- Troubleshooting of the ‘B’ control room ventilation train and chiller trip during normal operations on April 1, 2015
- ‘B’ EDG and ‘D’ EDG unavailable with one offsite source available on April 14, 2015
- ‘C’ Source Range Monitor (SRM) Counts Spiking on April 22, 2015
- Planned yellow risk condition during refueling outage for common shutdown cooling suction line testing on April 23, 2015
- ‘A’ control room emergency filtration system planned maintenance during operations with the potential to drain the reactor vessel on April 24, 2015

b. Findings

No findings were identified.

1R15 Operability Determinations and Functionality Assessments (71111.15 – 6 samples)

a. Inspection Scope

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

- High pressure coolant injection (HPCI) pump and HPCI booster pump tie down bolt torque values less than code required values found the week of April 1, 2015 (WO 70174682)
- Operability evaluation and apparent cause evaluation for sporadic Masterpact breaker trips and abnormal indications during the week of April 9, 2015 (WO 70174219)
- ‘H’ SRV accumulator found with no pressure during the week of April 15, 2015 (notification [NOTF] 20684902)
- ‘B’ EDG fuel oil leak during Loss of Power (LOP) / Loss of Coolant Accident (LOCA) testing the week of April 23, 2015 (NOTF 20686838)
- RCIC pump failure to start on May 9, 2015 (WO 70176529)
- Hope Creek evaluation of GE Energy Safety Communication SC05-03 (WO 70045899)

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 TSs 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. Findings

No findings were identified.

1R18 Plant Modifications (71111.18 – 3 samples)

.1 Temporary Modifications

a. Inspection Scope

The inspectors reviewed the temporary modifications listed below to determine whether the modifications affected the safety functions of systems that are important to safety. The inspectors reviewed 10 CFR 50.59 documentation and post-modification testing results, and conducted field walkdowns of the modifications to verify that the temporary modifications did not degrade the design bases, licensing bases, and performance capability of the affected systems.

- Temporary Configuration Change Package Number 4HT-15-002, Revision 0 – Jumper B EDG Jacket Water Keepwarm Heater H1KJ-1B-E-407

b. Findings

No findings were identified.

.2 Permanent Modifications

a. Inspection Scope

The inspectors evaluated modifications to the moisture separator dump valves and the degraded coating in the 10 inch circulating water dewatering line implemented by design change packages (DCPs) 80112615, 80112166 and 80110856. These DCPs: 1) removed the degraded coating in the dewatering line and reapplied the coating using the correct coating procedures; and, 2) modified the moisture separator dump valve controls to ensure the valves can cycle through a thermal transient without binding. The inspectors verified that the design bases, licensing bases, and performance capability of the affected systems were not degraded by the modification. In addition, the inspectors reviewed modification documents associated with the upgrade and design change, including the valve operation and the coating repair. The inspectors also reviewed revisions to the PSEG coating procedures, control room alarm response procedure and interviewed engineering and operations personnel to ensure the procedures could be reasonably performed.

- Degraded Coating in 10 inch Dewatering Line on April 17, 2015
- Moisture Separator Dump Valve Modification on April 23, 2015

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19 – 10 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’ Core spray pump breaker replacement after failing to close during in-service testing (IST) on April 1, 2015 (Order 60122623)
- ‘A’ Residual heat removal (RHR) test return valve failed to stroke open from the control room when placing torus cooling in service on April 3, 2015 (Order 60122629)
- HPCI pump and HPCI booster pump tie down bolt torque checks on April 4, 2015 (Order 60122715)

- Outboard Main Steam Isolation Valve 1ABHV-F028B out of specification too fast on April 11, 2015 (Order 50162634)
- ‘E’ Intermediate Range Monitor (IRM) replacement on April 20, 2015 (Order 60116859)
- ‘B’ EDG governor tuning following failure of large load reject test on April 22, 2015 (Order 60123016)
- ‘D’ channel 125 volt battery bank replacement on April 24, 2015 (Orders 30269307 and 30269308)
- Retest and post-installation inspection of RWCU snubber after it failed visual inspection and functional testing on April 28, 2015 (Order 60122798)
- ‘B’ Filtration, Recirculation and Ventilation System (FRVS) 18-month preventive maintenance on June 10, 2015 (Order 30261243)
- ‘D’ EDG after disassembly for preventive maintenance and extent of condition (EOC) inspections of the CAM Lobes on June 18, 2015 (Order 60119065)

b. Findings

No findings were identified.

1R20 Refueling and Other Outage Activities (71111.20 – 1 sample)

a. Inspection Scope

The inspectors reviewed the station’s work schedule and outage risk plan for the Hope Creek’s 19th refueling outage (H1R19), which was conducted April 10 through May 9, 2015. 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 TSs 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 TSs 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 TSs
- 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 licensee event report (LER) 05000354/2015-002-00. This LER and PSEG's use of EGM 11-003 will be reviewed and dispositioned in Section 4OA3.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22 – 7 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 TSs, 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-IS.BJ-0101, HPCI Valve In-Service Test on April 1, 2015 (in-service test)
- HC.OP-LR.GS-0009, Type C Leak Rate Test – CIV Test of 1GSHV-5031 & 5032 on April 13, 2015 (containment isolation valve)
- HC.OP-IS.KJ-0006, Integrated Emergency Diesel Generator 1CG400 Test – 18 months on April 21, 2015
- HC.OP-IS.KJ-0008, Integrated Emergency Diesel Generator 1DG400 Test – 18 months on April 23, 2015
- HC.OP-IS.ZZ-0001, In-service System Leakage Test of the Reactor Coolant Pressure Boundary on May 4, 2015 (reactor coolant system leakage)
- HC.RE-ST.BF-0001, Control Rod Scram Time Surveillance on May 4, 2015
- HPCI Main and Booster Pump 2 year Comprehensive Pump Test on June 2, 2015 (in-service test)

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06 – 1 sample)

Training Observations

a. Inspection Scope

The inspectors observed a simulator training evolution for licensed operators on June 23, 2015 which required emergency plan implementation by an operations crew. PSEG planned for this evolution to be evaluated and included in performance indicator (PI) data regarding drill and exercise performance. The inspectors observed event classification activities performed by the crew. The inspectors also attended the post-evolution critique for the scenario. The focus of the inspectors' activities was to note any weaknesses and deficiencies in the crew's performance and ensure that PSEG evaluators noted the same issues and entered them into the CAP.

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstone: Occupational and Public Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

a. Inspection Scope

The inspectors reviewed PSEG performance in assessing and controlling radiological hazards in the workplace. The inspectors used the requirements contained in 10 CFR 20, TSs, applicable Regulatory Guides (RGs), and the procedures required by TSs as criteria for determining compliance.

Inspection Planning

The inspectors reviewed the PIs for the occupational exposure cornerstone, radiation protection (RP) program audits, and reports of operational occurrences in occupational radiation safety since the last inspection.

Radiological Hazard Assessment

The inspectors reviewed recent plant radiation surveys and any changes to plant operations since the last inspection to identify any new radiological hazards for onsite workers or members of the public. The inspectors conducted walkdowns of the facility and reviewed risk-significant work activities.

Instructions to Workers

The inspectors observed containers of radioactive materials and assessed whether the containers were labeled and controlled in accordance with requirements.

The inspectors reviewed occurrences where a worker's electronic personal dosimeter (EPD) alarmed. The inspectors reviewed PSEG's evaluation of the incidents, documentation in the corrective action program, and whether compensatory dose evaluations were conducted when appropriate.

Contamination and Radioactive Material Control

The inspectors observed the monitoring of potentially contaminated material leaving the radiological control area and inspected the methods and radiation monitoring instrumentation used for control, survey, and release of that material.

Radiological Hazards Control and Work Coverage

The inspectors evaluated in-plant radiological conditions and performed independent radiation measurements during facility walk-downs and observation of radiological work activities. The inspectors assessed whether posted surveys, radiation work permits (RWPs), worker radiological briefings, the use of continuous air monitoring (CAM) and dosimetry monitoring was consistent with the present conditions. The inspectors examined the control of highly activated or contaminated materials stored within the spent fuel pools and the posting and physical controls for selected high radiation areas (HRAs), locked high radiation areas (LHRAs) and very high radiation areas (VHRA) to verify conformance with the occupational PI.

Risk-Significant HRA and VHRA Controls

The inspectors reviewed the controls and procedures for HRAs, VHRAs, and radiological transient areas in the plant.

Problem Identification and Resolution

The inspectors evaluated whether problems associated with radiation monitoring and exposure control were identified at an appropriate threshold and properly addressed in the corrective action program.

b. Findings

No findings were identified.

2RS2 Occupational ALARA Planning and Controls (71124.02)

a. Inspection Scope

The inspectors assessed PSEG's performance with respect to maintaining occupational individual and collective radiation exposures as low as is reasonably achievable (ALARA). The inspectors used the requirements contained in 10 CFR 20, applicable RGs, TSs, and procedures required by TSs as criteria for determining compliance.

Inspection Planning

The inspectors conducted a review of Hope Creek collective dose history and trends; ongoing and planned radiological work activities; radiological source term history and trends; and ALARA dose estimating and tracking procedures.

Radiological Work Planning

The inspectors selected and reviewed radiological work activities (drywell valve work, control rod drive removal, turbine building work, refueling floor work). For each of these activities, the inspectors reviewed: ALARA work activity evaluations; exposure estimates; exposure reduction requirements; exposure results achieved (dose rate reductions, actual dose); person-hour estimates and results achieved; and post-job reviews that identify lessons learned.

Verification of Dose Estimates and Exposure Tracking Systems

The inspectors reviewed the current annual collective dose estimate; collective dose basis methodology; and measures to track, trend, reduce, and adjust occupational doses for ongoing work activities.

Source Term Reduction and Control

The inspectors reviewed the current plant radiological source term and historical trend, plans for plant source term reduction, and contingency plans for changes in the source term as the result of changes in plant fuel performance or changes in plant primary chemistry.

Radiation Worker Performance

The inspectors observed radiation worker and radiation protection technician performance during work activities to determine if workers demonstrate the ALARA philosophy in practice and to determine whether the training and skill level was sufficient with respect to the radiological hazards involved.

Problem Identification and Resolution

The inspectors evaluated whether problems associated with ALARA planning and controls were identified at an appropriate threshold and properly addressed in the corrective action program.

b. Findings

No findings were identified.

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

a. Inspection Scope

The inspectors reviewed the control of in-plant airborne radioactivity and the use of respiratory protection devices in these areas. The inspectors used the requirements in

10 CFR 20, RG 8.15, RG 8.25, NUREG-0041, TS, and procedures required by TS as criteria for determining compliance.

Inspection Planning

The inspectors reviewed the UFSAR to identify ventilation and radiation monitoring systems associated with airborne radioactivity controls and respiratory protection equipment staged for emergency use. The inspectors also reviewed respiratory protection program procedures and current PI for unintended internal exposure incidents.

Engineering Controls

The inspectors reviewed operability and use of both permanent and temporary ventilation systems, and the adequacy of airborne radioactivity radiation monitoring in the plant based on location, sensitivity, and alarm set-points.

Use of Respiratory Protection Devices

The inspectors reviewed the use of respiratory protection devices in the plant to include applicable ALARA evaluations, respiratory protection device certification, respiratory equipment storage, air quality testing records, and individual qualification records.

Problem Identification and Resolution

The inspectors evaluated whether problems associated with the control and mitigation of in-plant airborne radioactivity were identified at an appropriate threshold and addressed by PSEG's corrective action program.

b. Findings

No findings were identified.

2RS4 Occupational Dose Assessment (71124.04)

a. Inspection Scope

The inspectors reviewed the monitoring, assessment, and reporting of occupational dose. The inspectors used the requirements in 10 CFR 20, Regulatory Guides, TSs, and procedures required by TSs as criteria for determining compliance.

Inspection Planning

The inspectors reviewed: radiation protection program audits and procedures associated with dosimetry operations.

External Dosimetry

The inspectors reviewed: dosimetry National Institute for Occupational Safety and Health National Voluntary Laboratory Accreditation Program (NVLAP) accreditation; onsite storage of dosimeters; the use of "correction factors" to align EPD results with

NVLAP dosimetry results; dosimetry occurrence reports; and corrective action program documents for adverse trends related to external dosimetry.

Internal Dosimetry

The inspectors reviewed: internal dosimetry procedures; whole body counter measurement sensitivity and use; adequacy of the program for whole body count monitoring of plant radionuclides; adequacy of the program for dose assessments based on air sample monitoring and the use of respiratory protection; and internal dose assessments for any actual internal exposures.

Special Dosimetric Situations

The inspectors reviewed: PSEG's worker notification of the risks of radiation exposure to the embryo/fetus; the dosimetry monitoring program for declared pregnant workers; external dose monitoring of workers in large dose rate gradient environments; and dose assessments performed since the last inspection that used multi-badging, skin dose or neutron dose assessments.

Problem Identification and Resolution

The inspectors evaluated whether problems associated with occupational dose assessment were identified at an appropriate threshold and properly addressed in the corrective action program.

b. Findings

No findings were identified.

2RS5 Radiation Monitoring Instrumentation (71124.05)

a. Inspection Scope

The inspectors reviewed performance in assuring the accuracy and operability of radiation monitoring instruments used to protect occupational workers and for effluent monitoring and analysis. The inspectors used the requirements in 10 CFR 20, 10 CFR 50, Appendix I; TSs; Offsite Dose Calculation Manual (ODCM); RGs; applicable industry standards; and procedures required by TSs as criteria for determining compliance.

Inspection Planning

The inspectors reviewed: PSEG 2013 and 2014 annual effluent and environmental reports; UFSAR; ODCM; RP audits; records of in-service survey instrumentation; and procedures for instrument source checks and calibrations.

Walkdowns and Observations

The inspectors conducted walk-downs of plant area radiation monitors, continuous air monitors and radioactive gaseous. The inspectors assessed material condition of these systems and that the monitor configurations aligned with the ODCM and the UFSAR. The inspectors checked the calibration and source check status of various portable

radiation survey instruments and contamination detection monitors for personnel and equipment.

Calibration and Testing Program

The inspectors reviewed calibration, functional testing results and alarm set-points (as applicable) for: portal monitor (GEM 5), contamination monitors (RM-14, SAC-4, ARGOS); survey meters (Telepole, RO2A); and personal air samplers (Gillian).

Instrument Calibrator

The inspectors reviewed the calibration standards used for portable instrument calibrations and response checks to verify that instruments were calibrated by a facility that used National Institute of Science and Technology traceable sources.

Calibration and Check Sources

The inspectors reviewed the plant waste stream characterization to assess whether the calibration sources used were representative of the radiation encountered in the plant.

Problem Identification and Resolution

The inspectors verified that problems associated with radiation monitoring instrumentation were identified at an appropriate threshold and properly addressed in the corrective action program.

b. Findings

No findings were identified.

Cornerstone: Public Radiation Safety (PS)

2RS6 Radioactive Gaseous and Liquid Effluent Treatment (71124.06)

a. Inspection Scope

The inspectors reviewed the treatment, monitoring, and control of radioactive gaseous and liquid effluents. The inspectors used the requirements in 10 CFR 20, 10 CFR 50, Appendix I; TS; ODCM; applicable industry standards; and procedures required by TSs as criteria for determining compliance.

Inspection Planning

The inspectors conducted in-office review of the PSEG's 2013 and 2014 annual radioactive effluent and environmental reports, radioactive effluent program documents, UFSAR, ODCM, and applicable event reports.

Walk-downs and Observations

The inspectors walked down the gaseous effluent monitoring systems to assess the material condition and verify proper alignment according to plant design. The inspectors

also observed potential unmonitored release points and reviewed radiation monitoring system (RMS) surveillance records and the routine processing and discharge of gaseous and liquid radioactive wastes. The inspectors observed collection of gaseous and liquid effluent samples.

Sampling and Analyses

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

Dose Calculations

The inspectors reviewed: changes in reported dose values from the previous annual radioactive effluent release reports; several liquid and gaseous radioactive waste discharge permits; the scaling method for hard-to-detect radionuclides; ODCM changes; land use census changes; public dose calculations (monthly, quarterly, annual); and records of abnormal gaseous or liquid radioactive releases.

Groundwater Protection Initiative (GPI) Implementation

The inspectors reviewed: groundwater monitoring results; changes to the GPI program since the last inspection; anomalous results or missed groundwater samples; leakage or spill events including entries made into the decommissioning files (10 CFR50.75(g)); and PSEG's evaluation of any positive groundwater sample results including appropriate stakeholder notifications and effluent reporting requirements.

Problem Identification and Resolution

The inspectors evaluated whether problems associated with the radioactive effluent monitoring and control program were identified at an appropriate threshold and properly addressed in PSEG's corrective action program.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification (71151)

Reactor Coolant System (RCS) Specific Activity and RCS Leak Rate (2 samples)

a. Inspection Scope

The inspectors reviewed PSEG's submittal for the RCS specific activity and RCS leak rate PI for the period of April 1, 2014, through March 31, 2015. To determine the accuracy of the PI data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance

Indicator Guideline," Revision 7. The inspectors also reviewed RCS sample analysis and control room logs of daily measurements of RCS leakage, and compared that information to the data reported by the PI.

b. Inspection Findings

No findings were identified.

4OA2 Problem Identification and Resolution (71152 – 2 samples)

.1 Routine Review of Problem Identification and Resolution Activities

a. Inspection Scope

As required by Inspection Procedure 71152, "Problem Identification and Resolution," the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that PSEG entered issues into the corrective action program 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 corrective action program and periodically attended condition report screening meetings. The inspectors also confirmed, on a sampling basis, that, as applicable, for identified defects and non-conformances, PSEG performed an evaluation in accordance with 10 CFR Part 21.

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 Inspection Procedure 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 CAP, such as trend reports, performance indicators, major equipment problem lists, system health reports, maintenance rule assessments, and maintenance or CAP backlogs. The inspectors also reviewed PSEG's CAP database for the first and second quarters of 2015 to assess notifications written in various subject areas (equipment problems, human performance issues, etc.), as well as individual issues identified during the NRC's daily condition report review (Section 4OA2.1). The inspectors reviewed PSEG's quarterly trend reports for the first and second quarters of 2015, conducted under LS-AA-125, Corrective Action Program, 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.

<u>Locked High Radiation Area Doors</u>

The inspectors found a potential adverse trend in the controls for high radiation areas due to a number of NOTFs written concerning inadequate posting and dock locks. The inspectors reviewed an event during H1R19, on April 16, 2015, that involved the Locked High Radiation Area (LHRA) watertight (WT) door 4403 to the ‘A’ RWCU pump room found to be unlocked when challenged by a radiation protection technician. PSEG’s initial investigation into the door’s locking mechanism found that the metal locking tab was degraded making the door lock non-functional. The investigation also revealed that the door had last been verified locked on April 10, 2015, as part of an extent of condition walkdown of LHRA doors following the identification of Turbine Building HRA door 1112A being found unlocked.

This type of WT door is secured by a series of ten dogs which are connected in a dogging mechanism which tightens or releases the dogs by a single rotating hand wheel that is engaged to the dogging mechanism by three steel pins located in the hand wheel. The hand wheel includes a locking mechanism which when rotated, pushes a metal locking tab downward, mechanically separating the hand wheel from the dogging mechanism and disengaging the three steel pins.

The inspectors reviewed PSEG’s Apparent Cause Evaluation (ACE 70169885) which determined that the LHRA WT doors had ineffective locking mechanisms due to the wearing down of the metal locking tab during repeated usage. PSEG’s EOC found 16 of 22 similar doors were deficient and required repairs to the metal locking tab to ensure the locking mechanism operated correctly.

The inspectors reviewed radiation protection procedures and previous issues at Hope Creek involving LHRA doors failing to provide adequate control of HRAs. During a review of a recent Hope Creek issue in October 2013, involving LHRA WT doors, the inspectors reviewed PSEG’s ACE (70159998) involving a failure of the locking mechanism for door 4503 to the ‘B’ RWCU filter holding pump room. PSEG’s ACE concluded that the failure of the locking mechanism was due to the loosening of screws that secure the metal locking tab in place. The ACE also stated that “a visual inspection of the locking mechanism could be added to the watertight door inspection PM with limited additional resource requirements. This inspection would provide defense in depth to future similar watertight door lock failures.” The inspectors reviewed PSEG’s EOC and completed corrective actions for the 2013 ACE, and found that: 1) the EOC only included 7 other LHRA WT doors [not the 22 LHRA WT doors that were part of the 2015 ACE EOC]; and, 2) the corrective action to revise the annual PM per Hope Creek procedure HC.MD-PM.ZZ-0007, did not revise the procedure to include a visual inspection of the locking mechanism, but only an inspection of the screws that secure the metal locking tab in place.

The inspectors determined that the LHRAs with deficient locking mechanisms never exceeded the TS 6.12.2 limit of 1000 millirem per hour (a restricted high radiation area) which shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry into the area. Based on this, the inspectors determined that these corrective action performance deficiencies were of minor significance in accordance with IMC 0612, Appendix B.

.3 Annual Follow-up Sample: EDG Jacket Water Coolant Leakage

a. Inspection Scope

The inspectors performed an in-depth review of PSEG's evaluations and the effectiveness of corrective actions associated with consistent but minor HCGS EDG jacket water coolant leakage. The inspectors selected for review a sample of EDG notification documents from January 2009 through January 2012. Three main issues were identified in the notifications reviewed: design deficiencies, loose bolted connections and maintenance practices. The largest contributor to EDG leakage was identified as loose bolted connections. This inspection was performed to determine if PSEG was continuing to appropriately identify and evaluate EDG leakage issues at the station and taking appropriate corrective actions to ensure the EDGs remained capable of performing the intended safety function.

The inspectors assessed whether PSEG's common cause analysis and prescribed corrective actions were focused on the identified cause(s). Additionally, the inspectors walked down the EDGs and observed activities related to EDG maintenance in the field.

b. Findings and Observations

No findings were identified.

The inspectors determined that while PSEG personnel identified repetitive issues associated with the EDG jacket water coolant leakage, PSEG personnel were appropriately identifying, monitoring, and documenting the circumstances surrounding the issue, and were evaluating the leaks in accordance with PSEG procedures. Further, the inspectors observed that PSEG personnel investigated possible causes and corrective actions to these leaks. The inspectors determined PSEG's activities, approaches, and on-going plans to be adequate to determine the causes of the EDG jacket water leakage and their corrective actions were focused on ensuring correct torque was applied to the bolted connections and appropriate maintenance practices were in place to improve performance of the EDG.

Additionally, the inspectors noted that the licensee reviewed all EDG leaks documented from January 2009 through January 2012 to improve their monitoring program. The various leaks included jacket water, fuel oil and lube oil. Overall, the inspectors determined that PSEG's corrective actions were adequate and reasonable in addressing the causes of EDG coolant leakage.

4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153 – 3 samples)

.1 (Closed) LER 05000354/2013-011-00: Filtration, Recirculation, and Ventilation System (FRVS) Exceeded Technical Specification Allowed Outage Time

a. Inspection Scope

On March 26, 2015, the NRC Problem Identification and Resolution (PI&R) inspection team issued an NCV 05000354/2015008-02, Failure to Submit a Licensee Event Report for a Condition Prohibited by Technical Specifications (TSs) because PSEG did not provide a written LER to the NRC within 60 days of identifying a condition prohibited by

the plant's TS. Specifically, PSEG personnel did not submit a 50.73 report for the inoperability of a 'B' FRVS recirculation fan that exceeded its TS allowed outage time. The inspectors determined that the 'B' FRVS recirculation fan was not operable for 21 days, from June 3 to June 24, 2013, which is longer than the FRVS recirculation TS allowed outage time of 7 days. NUREG-1022, "Event Report Guidelines 10 CFR 50.72 and 50.73," Revision 3, section 3.2.2, "Operation or Condition Prohibited by Technical Specifications" states, in part, that "an LER is required if a condition existed for a time longer than permitted by the TS even if the condition was not discovered until after the allowable time had elapsed and the condition was rectified immediately upon discovery." PSEG entered this issue into the CAP as NOTF 20678572, and planned corrective actions include submitting a LER per 10 CFR 50.73 and performing a causal evaluation.

PSEG submitted LER 2013-011-00 for FRVS exceeding the TS allowed outage time on March 25, 2015. PSEG's review of plant conditions determined that there were two periods of TS non-compliance due to the 'B' FRVS recirculation unit inoperability. The first was from June 10, 2013, at 3:38 p.m., and lasted until June 13, 2013, at 4:00 p.m., a period of 3 days, 22 minutes. The second non-compliance occurred on June 17, 2013, at 10:02 a.m., when a mode change was made into Operational Condition 2, Startup, with the 'B' FRVS recirculation unit inoperable, contrary to the requirements of TS 3.0.4. The plant operated for 7 days, 5 hours, 28 minutes, with the 'B' FRVS recirculation unit inoperable, until June 24, 2013, at 4:30 p.m., when the 'B' FRVS recirculation fan was restored to operable. In addition, PSEG's review determined that the 'A' EDG was inoperable for barring on June 18, 2013, from 7:41 p.m. until 7:52 p.m., a period of 11 minutes. With the 'A' EDG inoperable, the 'A' and 'E' FRVS recirculation units did not have a backup emergency power supply. As a result, three FRVS recirculation units were inoperable for a period of 11 minutes. Per the Hope Creek UFSAR, a minimum of four FRVS recirculation units are required for the first 10 minutes following a design basis accident to provide cooling to the Reactor Building air, and to reduce the offsite dose.

In response to the LER and associated NCV, the inspectors reviewed PSEG's causal evaluation (70174057) for the failure to submit an LER for 'B' FRVS recirculation fan. The evaluation determined that PSEG failed to effectively implement the appropriate risk management challenges and reviews to properly assess the operability of the 'B' FRVS recirculation fan. Specifically, faced with an intermittent connection and inadequate preventative maintenance, Hope Creek's management review committee should have concluded that operability of the 'B' FRVS recirculation fan during this time period could not be assured. PSEG's corrective actions for this issue included submitting the required LER per 10 CFR 50.73 and implementing the use of a 'devil's advocate' as a human performance tool to ensure better decision making.

The inspectors reviewed PSEG's LER, evaluation, NRC PI&R Inspection Report 2015008, supporting documentation, and station procedures and associated TS regarding the FRVS inoperability event. One finding was previously identified and discussed in NCV 05000354/2015008-02. No additional findings were identified during the review of this LER. This LER is closed.

.2 (Closed) LER 05000354/2015-001-00 and -01: Conditions Prohibited by Technical Specifications Due to Core Spray Inoperabilities

a. Inspection Scope

On March 31, 2015, at 1:42 p.m., the breaker for 'A' Core Spray (CS) pump failed to close during normal surveillance testing. Technical Specification (TS) 3.5.1.a was entered for one inoperable CS subsystem. The breaker was replaced and the surveillance was satisfactorily performed, and the 'A' CS subsystem was declared operable on March 31, 2015, at 8:00 p.m. PSEG performed troubleshooting which indicated that the failure in the breaker control device most likely existed since the last breaker operation on January 8, 2015, at 10:00 a.m., and vendor failure analysis concluded that the spring in the breaker control device failed due to cyclic fatigue, preventing the breaker from closing. Accordingly, PSEG determined that the ‘A’ CS subsystem was inoperable for longer than the TS allowed outage time (7 days). Therefore, the condition was determined to be reportable per 10 CFR 50.73(a)(2)(i)(B) as any operation or condition prohibited by TS. During the review of this event, PSEG also determined that 'B' CS subsystem was inoperable from February 9, 2015, at 3:00 a.m., until February 10, 2015, at 3:32 p.m. (36 hours and 32 minutes) when planned maintenance was performed on the 'B' EDG. This condition was determined to be reportable per 10 CFR 50.73(a)(2)(v) as an event or condition that could have prevented the fulfillment of a safety function.

The inspectors reviewed the LER and LER supplement, the associated causal analysis (ACE 70175101) and corrective actions, the completed vendor failure analysis on the breaker control device, interviewed PSEG staff, related corrective action program (CAP) notifications and walked down associated components. The inspectors found that the vendor failure analysis indicated:

1. Fatigue where the spring bends, or kinks, to form the hook that attaches the spring to the contact carrier inside the control device; and,
2. Permanent deformation, or a visible gap, in the spring coil turns.

In discussing the failure analysis with PSEG, the inspectors determined that the bend, or kink, in the spring for the hook is a known high stress location and the kink introduces an additional stress riser that promotes fatigue crack initiation, which occurred over several stress cycles, suggesting that the spring failed due to an accumulation of operations of the breaker control device. PSEG engineering also indicated that the permanent deformation, or visible gap, in the spring coil turns were most likely caused during manufacturing, prior to the breaker control device assembly.

Based on a review of PSEG’s preventative maintenance strategy, CAP documents, ABB safety and non-safety related breaker failure history, no previous operating experience, and the fact that the cause of the inoperability, a failed spring inside the sealed breaker control device that was still within the manufacturer’s recommended life span, was due to a manufacturing defect that could not have been identified during inspection and testing or avoided through management controls, the inspectors determined that this type of failure was not within PSEG’s ability to foresee and correct. Therefore, the inspectors determined there was no licensee performance deficiency associated with the violation of the TS 3.5.a.1 limiting conditions for operation. NRC Inspection Manual Chapter 0612, Appendix B, “Issue Screening,” directs disposition of

such issues using traditional enforcement in accordance with the Enforcement Policy. The inspectors used Enforcement Policy, Section 6.1.d.1, “Reactor Operations,” to evaluate the significance of this violation, and concluded that the violation was more than minor and best characterized as a Severity Level IV violation in that the issue was associated with a failure to comply with a technical specification action requirement. In reaching this conclusion, the inspectors considered that the underlying technical finding would have been evaluated as having very low safety significance (i.e. Green) under the Reactor Oversight Process using NRC IMC 0609, Appendix A, Exhibit 2, “Mitigating Systems Screening Questions,” dated June 19, 2012 because, although the issue involved the potential loss of system and/or function and therefore required a detailed risk evaluation, the calculated delta core damage frequency (CDF) was mid E-8. Because this change in CDF was less than 1E-7, no further evaluation of external events or large early release frequency was required.

Because it was not reasonable for PSEG to have been able to foresee and prevent the violation, the NRC determined no performance deficiency existed. Thus, the NRC has decided to exercise enforcement discretion in accordance with NRC Enforcement Policy Section 2.2.4, “Exceptions to Using Only the Operating Reactor Assessment Program,” and Section 3.5, “Violations Involving Special Circumstances,” and refrain from issuing enforcement action for the violation (EA-15-147). Further, because PSEG’s action and/or inaction did not contribute to this violation, it will not be considered in the assessment process or the NRC’s action matrix. This LER is closed.

.3 (Closed) LER 05000354/2015-002-00: Operations with a Potential to Drain the Reactor Vessel (OPDRV) Without Secondary Containment

a. Inspection Scope

On April 14, 15, 17, 20, 27 and 29, 2015, 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 removal of the scram air header from service (2:00 to 5:15 p.m. on April 14, 2015), ‘B’ RRP seal replacement (4:36 a.m. on April 15, 2015, through 2:55 a.m. on April 24, 2015), control rod drive replacements (2:17 p.m. on April 17, 2015, through 1:02 p.m. on April 20, 2015), Local power range monitor replacements (3:13 a.m. on April 20, 2015, through 6:40 a.m. on April 23, 2015), scram discharge volume tagging (1:14 to 1:26 p.m. on April 27, 2015), and the fill and vent for the ‘B’ RRP seal (8:41 p.m. on April 29, 2015, through 6:45 a.m. on April 30, 2015). 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 6:45 a.m. on April 30, 2015.

The NRC issued EGM 11-003, Revision 2, “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 13, 2013, which provides, in part, for

the exercise of enforcement discretion only if the licensee demonstrates that it has implemented specific interim actions during any OPDRV activity. The inspectors determined that PSEG's implementation of these specific interim actions during these OPDRV activities were adequate and met the intent of EGM 11-003, Revision 2.

The inspectors' assessments of PSEG's implementation of these 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 ('A' 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 in compliance with the minimum water level allowed by Hope Creek 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 with the capability to inject water equal to, or greater than, the maximum potential leakage rate from the RPV for a minimum time period of 4 hours.
- PSEG reported that the worst case estimated time to drain the reactor cavity to the RPV flange was 26 hours, which met the EGM criteria of greater than 24 hours.
- The inspectors verified that the OPDRV was not conducted in Mode 4 and that PSEG did not move recently 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.

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 2:00 p.m. on April 14, 2015, and 6:45 a.m. on April 30, 2015, 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 2, the NRC is exercising enforcement discretion in accordance with NRC Enforcement Policy Section 2.2.4, "Exceptions to Using Only the Operating Reactor Assessment Program," and Section 3.5, "Violations Involving Special Circumstances," and, therefore, will not issue enforcement action for this violation.

In accordance with EGM 11-003 Revision 2, 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 STS to provide more clarity to the term OPDRV. 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.

4OA5 Other Activities

a. Inspection Scope

The inspectors performed a review of recently issued Hope Creek Operator licenses. Inspection activities were performed using NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9, Supplement 1.

b. Findings

Introduction. The NRC identified an NCV of 10 CFR 50.9, "Completeness and Accuracy of Information," for submitting NRC Form 398, "Personal Qualification Statement – Licensee," for one SRO applicant, without requesting a GFE waiver as required for an individual who had not taken an NRC developed exam within two years of the date of the application for their NRC license. This violates the requirement of 10 CFR 50.9 that "Information provided to the Commission by an applicant for a license or by a licensee… shall be complete and accurate in all material respects."

Description. On February 6, 2015, the facility licensee submitted the final license applications for the February 23, 2015 exam, including NRC Form 398 ("Personal Qualification Statement – Licensee") for eight individuals. Per NUREG-1021, ES-202.C.1.f, the facility licensee certifies that an applicant has completed all of its requirements and commitments for a license by placing a check in Item 19.B of NRC Form 398, having the form signed by the senior management representative on site, and submitting it to the NRC. These requirements include experience, control manipulations, training and medical information. The NRC Form 398 for the SRO applicant in question had a check in Item 19.B and was signed by the senior management representative; with Item 4.G checked and the date 09/13 entered in the blank indicating that an NRC prepared GFE had been passed in September 2013, which met the requirement to pass a GFE within two years of the application for an NRC Senior Operator license. NRC Licensing Assistants identified, during a review of recently issued Hope Creek operator licenses that the SRO in question had been enrolled for the September 2013 GFE, but had withdrawn before the exam was administered. No examination results were on file with the NRC and no waiver had been requested on the NRC Form 398. Subsequent to this discovery, Region I staff contacted the facility licensee on April 28, 2015, to inform them of the error. In a letter (LR-N15-0108) dated May 4, 2015, the facility licensee acknowledged the error and provided the following information: The SRO in question had completed classroom instruction and passed a GFE prepared by PSEG in accordance with NUREG-1021, ES-205, Section D and administered under controlled conditions. The Facility licensee administered GFE was given on August 16, 2013. The applicant had previously passed an NRC prepared GFE, as part of Initial License Training when he was employed at Oyster Creek in 2010.

Analysis. The failure of the facility licensee to request a GFE waiver on NRC Form 398 for an SRO applicant who had not passed an NRC prepared GFE within two years was a performance deficiency. The inspectors evaluated this issue using the traditional enforcement process because the performance deficiency had the potential for impacting the NRC's ability to perform its regulatory function. Specifically, had the NRC not identified the failure to request a GFE waiver, it could have caused the NRC to issue a Senior Reactor Operator license to an individual who failed to meet the preparation

requirements of NUREG-1021. The performance deficiency was determined to be Severity Level IV because it fits the SL-IV example of Enforcement Policy Section 6.4.d.1.a, "Violation Examples: Licensed Reactor Operators." This section states, "Severity Level IV violations involve, for example …cases of inaccurate or complete information inadvertently provided to the NRC that does not contribute to the NRC making an incorrect regulatory decision as a result of the originally submitted information or an unqualified individual performing the functions of an operator or senior operator…." Because the SRO applicant in question met the requirements for a waiver, and would have been granted a waiver if it had been requested, the performance deficiency did not cause the NRC to make an incorrect regulatory decision.

This finding is being treated as an NCV because the licensee placed the violation into their corrective action program; the violation was non-repetitive and the violation did not involve willfulness. This violation was processed using Traditional Enforcement. No associated ROP finding was identified and no crosscutting aspect was assigned.

Enforcement. 10 CFR 50.9 requires that; "Information provided to the Commission by an applicant for a license or by a licensee…shall be complete and accurate in all material respects." Contrary to the above, the facility licensee provided information to the NRC Region I staff that was not accurate in all material respects. Specifically, on February 6, 2015, the facility submitted NRC Form 398 for an SRO applicant who had not taken an NRC prepared GFE within two years of the license application date without requesting a GFE waiver. The facility submitted documentation on May 4, 2015, clarifying that the applicant had passed a facility prepared GFE under controlled conditions in August, 2013, which would satisfy the requirement for the NRC Region I office to grant a waiver for the GFE. Because this finding is of very low safety significance and was entered into the corrective action program as Notification 20687475, this violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: **NCV 05000354/2015002-02, Failure to Request a Generic Fundamentals Examination Waiver for a Senior Operator License Applicant.**

4OA6 Meetings, including Exit

On July 16, 2015, the inspectors presented the inspection results to Mr. P. Davison, Site Vice President, 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.

ATTACHMENT: SUPPLEMENTARY INFORMATION

SUPPLEMENTARY INFORMATION

KEY POINTS OF CONTACT

<u>Licensee Personnel</u>

P. Davison, Site Vice President
E. Carr, Plant Manager
A. Bauer, Component Maintenance Organization Engineer
L. Cary, Technical Analyst
R. Cary, Environmental Coordinator
R. Chan, Nuclear Oversight Manager
S. Connelly, Lead Nuclear Engineer, Plant Engineering
M. Conroy, Air Operated Valve Program Engineer
M. Crisafulli, Acting Maintenance Manager
B. Daly, Manager, Sustainability, Environmental Affairs
P. Davison, Site Vice-President
D. Denelsbeck, Radiation Protection Support (Salem)
K. Denny, System Engineer
P. Duca, Senior Regulatory Engineer
P. Duke. PSEG Licensing Manager
D. Fisher, Nuclear Oversight
R. Heathwaite, Chemistry Supervisor
A. Kazarfan, System Engineer
M. Kelly, Senior Component Maintenance Organization Engineer
J. Kepley, Operations Training Instructor
K. Kinade, Engineering Manager
J. Krall, Reactor Engineering Manager
A. Kraus, Manager, Nuclear Environmental Affairs
S. Kugler, Chemistry Manager
D. Labott, CFAM, Chemistry
T. MacEwen, Hope Creek Compliance Engineer
S. Madden, Design Manager
E. Maloney, Programs Engineer, Inservice Inspection
M. Mazucco, In Vessel Visual Inspection Engineer
M. Meltzer, Chemistry
F. Mooney, Maintenance Director
D. Nestle, CFAM, Radiation Protection
S. Nevelos, Regulatory Assurance Manager
C. Payne, System Engineer
M. Schaffer, Shutdown Safety Lead
W. Schmick, Project Manager
M. Shaffer, Operations Training Manager
A. Simkins, Systems Engineer
S. Simpson, Regulatory Assurance Manager
J. Stevenson. ISI Program Engineer
K. Timko, System Engineer
A. Tramontana, Hope Creek Programs Engineering Manager
H. Trimble, Radiation Protection Manger
C. Wend, Superintendent, Radiation Protection

PSEG Others

R. Tolbert, Chemistry Staff
D. Wahl, Chemistry Staff

Others

J. Vouglitois, Nuclear Engineer, Nuclear Environmental Engineering Section, State of New Jersey

LIST OF ITEMS OPENED, CLOSED, DISCUSSED, AND UPDATED

Opened/Closed

05000354/2015002-01	NCV	Failure to Identify and Correct a Condition Adverse to Quality Associated with Safety Relief Valve Inlet Piping (Section 1R08)
05000354/2015002-02	NCV	Failure to Request a Generic Fundamentals Examination Waiver for a Senior Operator License Applicant (Section 4OA5)

Closed

05000354/2013-011-00	LER	Filtration, Recirculation, and Ventilation System (FRVS) Exceeded Technical Specification Allowed Outage Time (Section 4OA3)
05000354/2015-001-00 and -01	LER	Conditions Prohibited by Technical Specifications Due to Core Spray Inoperabilities (Section 4OA3)
05000354/2015-002-00	LER	Operations With A Potential To Drain The Reactor Vessel (OPDRV) Without Secondary Containment (Section 4OA3)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Procedures
WC-AA-107, Seasonal Readiness, Revision 13
OP-AA-108-111-1001, Severe Weather and Natural Disaster Guidelines, Revision 12
OP-AA-108-107-1001, Electrical Systems Emergency Operations and Electrical System Operator Interface, Revision 3

Notifications
20691203

Maintenance Orders/Work Orders
70169481

Section 1R04: Equipment Alignment

Procedures
HC.OP-SO.BC-0002, Decay Heat Removal Operation, Revision 33
HC.OP-SO.KJ-0001, Emergency Diesel Generators Operation, Revision 72
HC.OP-ST.EA-0001, Service Water Flow Path Verification – Monthly, Revision 11

Notifications (*NRC-identified)
20552005 20684797* 20684861 20684891 20684972 20688582*

Drawings
M-51-1, Sheet 1, Residual Heat Removal, Revision 45
M-51-1, Sheet 2, Residual Heat Removal, Revision 42
M-30-1, Sheet 1, Diesel Engine Auxiliary Systems Fuel Oil, Revision 27
M-30-1, Sheet 2, Diesel Engine Auxiliary Systems Intercooler and Injector Cooling, Jacket Water, Crankcase Vacuum Air Intake, Exhaust, and Vibration Monitoring Systems, Revision 23
M-30-1, Sheet 3, Diesel Engine Auxiliary Systems Starting Air & Lube Oil, Revision 23
M-10-1, Sheet 1, Service Water, Revision 55

Section 1R05: Fire Protection

Procedures
ER-AA-330-010, Snubber Functional Testing, Revision 6
FRH-II-411, Torus Water Clean-up Pump Room Elev. 54’ – Rm 4101, Revision 3
FRH-II-415, Hope Creek Pre-Fire Plan, Dry Well Pad & Torus Area, Elevations: 54’-0” & 77’-0”, Revision 4
FRH-II-452, Electrical Equipment Area, RWCU Rooms Elev. 145’ – Pipe Chase Rm 4505, Revision
FRH-III-114, Hope Creek Pre-Fire Plan Turbine Building Elev. 54’ – Rm 1112 Condensate Piping Area, Revision 4
FRH-III-151, Hope Creek Pre-Fire Plan Turbine Building Elev. 137’ – Rm 1516 and 1517 Motor Generator Rooms, Revision 4
FRH-II-461, Hope Creek Pre-Fire Plan, FRVS Rooms, MCC Area, Recombiner Areas, Spent Fuel Pool & Gamma Scan Detector Area, Elevations: 162’-0”, Revision 3

MA-AA-796-024,
SH.MD-GP.ZZ-0001, Snubber Removal and Installation, Revision 8

Notifications

20687042	20687047	20687413	20684554	20684564	20685862
20688594	20691508				

Maintenance Orders/Work Orders

60122798	70175591	70165179

Drawings
C-0399-0, Sht. 504, Revision 0

Miscellaneous
10855-C-136, Technical Specification for Installation of Expansion-Type Anchors for the Hope Creek Generating Station, Revision 12
Hope Creek Engineering List of Functional and Visual Testing Results for Snubbers during RF17, RF18, and RF19, dated June 6, 2015

Section 1R07: Heat Sink Performance

Procedures
ER-AA-340-1002, Service Water Heat Exchanger and Component Inspection Guide, Revision 5
HC.MD-PM.EG-0001, Safety Auxiliaries Cooling System (SACS) Heat Exchanger Inspection, Revision 12
HC.OP-ST.BC-0009, Residual Heat Removal System RHR Heat Exchanger Flow Measurement Test - 18 Month, Revision 15

Notifications

20684303	20684642	20684969	20685320	20686051	20688867

Maintenance Orders/Work Orders

30214171	30214172	30228940	30228941	50161799	60122938
70175374	80114188				

Calculations
EA-0033, Biofouling Monitoring and Trending Calculation, Revision 0
EG-0047, HCGS Ultimate Heat Sink Temperature Limits – EPU, Revision 5

Miscellaneous
Hope Creek Troubleshooting Log for NOTF 20684642 – H1BC-BC-HV-F048A67, May 6, 2015

Section 1R08: In-Service Inspection

Notifications (*NRC-identified)

20556380	20559393	20560420	20583219	20602181	20616922
20616922	20623016	20624391	20667168	20673253	20673253
20679015	20679583	20685195*	20685508	20685515	20685516
20685517	20685577	20685580	20685658	20685659	20685660
20685661	20685664	20685820	20685885	20685885	20685965
20686023	20686023	20686079	20686081	20686147	20686355
20686355	20686355	20686355	20687515*	20866344	80113353

NDE Inspection Reports & Data Sheets

Summary No. 100180, Report No.: UT-15-047, Component ID: PRV1-W24F, Meridional Weld, 100%, NRI

Order No.: 50162873, Surface Examination Record: Liquid Penetrant Examination: A, B, C, D, E, F, G, H, J, K, L, M, P, and R, SRV's, 4/18/15; NRI Observed

NWS Technologies, LLC, FORM NWS-1 Report of Repair of Nuclear Pressure Relief Devices; Target Rock, Main Steam Safety Relief Valve, Serial #7567F

MISTRAS Ultrasonic Thickness Examination Record, Drywell Air Gap, 4/12/2015, 5 pages, Order #50162748

MISTRAS Ultrasonic Thickness Examination Record, Drywell Air Gap, 4/12/2015, 3 pages, Order #50162748

MISTRAS Ultrasonic Thickness Examination Record, Drywell Air Gap, 4/12/2015, 4 pages, Order #50162748

Summary No. 101700, Report UT-15-025, (4 pages), Component 1-AB-26DLA-030-1, safe end to pipe, line to N3A, Accept

Summary No. 105470, Liquid Penetrant Exam, Report PT-15-003; Component 1-BB-28VCA-011-5LG (1-4); Ni-B line; Accept

Summary No. 100320, Report HPC-N8A-NV (9 pages), RPV Nozzle to Shell, Accept

Summary No. 820007, Report VE-13-032 (1 page) Accept

Summary No. 100520, Report VEN-15-008 (1 page), Component RPVI-RPVI-6AIR (inside radius) section, 4/24/25, Accept

Summary No. 100180, Report UT-15-047, RPV1-W24F, Meridional Weld, 4/25/15, (2 pages), Accept

Summary No. 100530, Report Ven-15-009, RPVI-N7IR, 4/28/15, Accept

Hope Creek Exam Data Sheet No.:820030; ASME IWE (Class MC) Containment Visual Exam Record, 4/22/15; (11 pages); no recordable indications

Hope Creek Exam Data Sheet No.:820012; ASME IWE (Class MC) Containment Visual Exam Record, 4/22/15; (4 pages); no recordable indications

Hope Creek Exam Data Sheet No.:820015; ASME IWE (Class MC) Containment Visual Exam Record, (4/30/12 to 5/1/12); (3 pages); no recordable indications

Report Number HPC-N7, RPV Axial and Circumferential Weld Coverage Work Sheets; page 7 of9 and 8 of 9; N7 Nozzle to Head Weld Axial Scan Coverage (by calculation) 97.7%; N7 Nozzle to Head Weld Circumferential Scan Coverage (by calculation) 66.7%

Summary No. 109783, Report (UT-15-050), (5 pages); Component ID: 1-BG-6DBA-001-23B; RWCU System Flow Element to Pipe Stub, dissimilar metal weld, 4/28/15; Accept

Summary No. 109782, Report (UT-15-051), (5 pages); Component ID: 1-BG-6DBA-001-23A;
RWCU System Pipe Stub to Flow Element, dissimilar metal weld, 4/28/15; Accept

Summary No.: 165040, Report (VT-15-181), (2 pages), Valve Internal Surfaces, Component
1-AB-HV-F028A-V-032-VIS, 4/22/15

MISTRAS Ultrasonic Thickness Examination Record, Component H1AB-1ABV-32; Order 50162873, 5/20/15

AREVA INR's (IVVI Conditions Reported and evaluated for future monitoring)

INR-HCI-15-004
INR-HCl-15-001
INR-HCI-15-002
INR-HCI-15-003

Engineering Calculations & Evaluations

Order 000070176268 Operation 0010 Long Text, Title: Perform TE supporting resolution of the issue documented in Notification 20688521, Tool Marks on SRV. (05-04-15)

GEH Report 002N6465 R0, ECO-0014983, Class II, April 2015, Hope Creek Generating Station Effects of SRV Branch Pipe Gouged Wall Thickness on Main Steam Piping System

Order 80108395 Operation 0010, Minimum Allowable Wall Thickness Evaluation of 4" RHR Piping

Order 70172257, Potential missed ASME Examination

Order 60122981, Blending of SRV inlet piping gouges

NDE Inspector Certifications

A2289	B2283	B4738	B0861	C5341	P6274
M6527	W8789	Y4315	Z5837		

Self-Assessments

NRC In-service (ISI) Activities Check-In Self-Assessment H1R19

ASME Repair-Replacement Plans/Work Orders

Order 60113722, NUCM Work Order to Install Flange IAW Procedure

Order 60118109 Replacement of spool piece 1-BC-050-S05

Plant Procedures

PSEG Nuclear Maintenance Procedure, HC.MD-CM.AB-0006-Rev. 25; Main Steam Safety/Relief Valve Removal and Installation, 1/15/15

Order 50157199, NUST ST 54M RPLC 1ABPSV-F013H with Spare

PSEG LS-AA-120, Revision 13, Issue Identification and Screening Process

NDE Examination Procedures

PSEG Procedure OU-AA-335-018, Revision 5; Detailed and General, VT-1 and VT-3 Visual Examination of ASME Class MC and CC Containment Surfaces and Components

PSEG Nuclear Procedure OU-AP-335-043, Revision 2; Bare Metal Visual Examination (VE) Of Class 1 PWR Components Containing Alloy 600/82/182 and Class 1 PWR Reactor Pressure Vessel Heads

Areva NP Inc. Nondestructive Examination Procedure 54-ISI-805-008, Issue Date, 10/26/99, Revision Date, 8/7/12

Areva NP Inc. Nondestructive Examination Procedure, Ultrasonic Examination of Ferretic piping Welds, Procedure Number 54-ISI-835-014, Issue Date, 2/16/99, Revision Date, 08/08/11

Areva NP Inc. Nondestructive Examination Procedure, Manual Ultrasonic Examination of BWR Reactor Vessel Nozzle Inner Radius Regions and Nozzle to shell welds (inner 15%), Procedure Number 54-ISI-850-008, Issue Date, 2/15/03, Revision Date, 4/02/13

Areva NP Inc. Nondestructive Examination Procedure; Manual Ultrasonic Examination of Dissimilar Metal Piping Welds; Procedure Number 54-ISI-829-011, 08/03/10

Areva NP Inc. Nondestructive Examination Procedure; Manual Ultrasonic Examination of BWR Reactor Vessel Nozzle Inner Radius Regions and Nozzle to Shell Welds (inner 15%); Procedure Number 54-ISI-850-008, 04/02/13

PSEG Nuclear L.L.C., ER-AA-330-002, Revision 9; In-service Inspection of Section XI Welds and Components

PSEG Nuclear L.L.C. ER-AA-43, Revision 3; Materials Degradation Management Program (MDMP)

PSEG Nuclear L.L.C. OU-AA-335-017, Revision 2; VT-3 Visual Examination of Pump and Valve Internals

EPRI Technical Report, Nondestructive Evaluation: Guideline for Conducting Ultrasonic Examinations of Dissimilar Metal Welds, Revision 1, 300200091, Final Report, May 2013

Maintenance Orders/Work Orders
50157199

Miscellaneous
PSEG Letter LR-N14-0037, Feb. 07-2014, To: USNRC; Subject: Hope Creek Generating Station, Renewed Facility Operating License No.-57, Docket No. 50-354; INSERVICE INSPECTION ACTIVITIES-90 DAY REPORT EIGHTEENTH REFUELING OUTAGE
PSEG Nuclear Program Document, ER-AP-330-1001, Revision 3; Alloy 600 Management Plan

Section 1R11: Licensed Operator Requalification Program

Procedures
HC.OP-IO.ZZ-0003, Startup from Cold Shutdown To Rated Power, Revision 106
HC.OP-IO.ZZ-0004, Shutdown from Rated Power To Cold Shutdown, Revision 99
HU-AA-1211, Pre-Job Briefings, Revision 12
OP-AA-101-111-1004, Operations Standards, Revision 5

Other Documents
REMA 2015-0039, Reactivity Maneuver Plan for Cycle 20 Initial Startup through ~23%, Revision 0

Section 1R12: Maintenance Effectiveness

Procedures
HC.OP-ST.BD-0004, RCIC System Response Time and Flow Test – 18 Months, Rev 8
HC.OP-SO.BD-0001, RCIC System Operation, Rev 43
HC.MD-PM.FC-0001, RCIC Steam Turbine Inspection and P.M. Rev 30
HC.OP-AB.RPV-0001, Reactor Power, Rev 14
HC.OP-SO.BB-0002, Reactor Recirculation System Operation, Revision 101
MA-AA-716-006, Control of Lubricants Program, Revision 11
MA-AA-716-230-1001, Oil Analysis Interpretation Guideline, Revision 11
MA-AA-716-230-1004, Lubricant Sampling Guideline, Revision 3
SH-PBD-AMP.XI.M39, Lubricating Oil Analysis (Program Basis Document), Revision 3

Notifications
20651102 20680046 20685210 20689441

Maintenance Orders/Work Orders
60123401 70174237 70176529 70176824

Miscellaneous
PN1-E51-C001-0055, RCIC Pump Vendor Manual, Revision 3
OTDM HC-14-008, ‘B’ Reactor Recirculation MG Set Speed Control, Revision 3

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures
HC.OP-DL.ZZ-0006, Log 6 Auxiliary Building Log, Revision 54
HC.OP-DL.ZZ-0026, Surveillance Log, Revision 147
HC.OP-ST.SE-0005, SRM Channel Count Rate Surveillance, Revision 9
OP-AA-108-116, Protected Equipment Program, Revision 10
HC.OP-GP.SM-0001, Defeating Isolation Signals During Refuel Operations, Revision 12
OP-HC-108-102, Management of Operations with the Potential to Drain the Reactor Vessel (OPDRV), Revision 4
HC.OP-AB.HVAC-0001,
ER-AA-310-1004-F1, Revision 3

Notifications

20497196	20519328	20681109	20683149	20686471	20688257
20688262	20688370				

Maintenance Orders/Work Orders
60098249 70175460

Drawings
M-78-1, Sheet 1, Aux. Building Control Area Air Flow Diagram, Revision 24
P-9266-1, HVAC Area Drawing Aux. Bldg. Area 26 Plan at El. 155’-3” & 163’-6, Revision 25

Miscellaneous
OP-HC-108-115-1001, Form 1, Technical Specification Action Statement Log, 15-132, ‘A’ CREF Train, dated April 27, 2015
Protected Equipment Log – P3 Window – Plan 4, April 23, 2015
Protected Equipment Log – ‘A’ Fuel Pool Cooling – Plan 10, April 16, 2015
Protected Equipment Log – ‘B’ Fuel Pool Cooling – Plan 11, April 16, 2015
ORAM-SENTINEL – Shutdown Cooling, Gate Out, Decay Heat Medium, February 26, 2015

Section 1R15: Operability Determinations and Functionality Assessments

Procedures
HC.OP-ST.BF-0002, Control Rod Drive Accumulator Operability Check – Weekly, Revision 9
HC.RE-RA.BF-0002, Channel Distortion Testing, Revision 14
HC.RE-ST.BF-0001, Control Rod Scram Time Surveillance, Revision 34
HC.OP-AR.ZZ-0011, Overhead Annunciator Window Box C6, Revision 60
OP-AA-108-115, Operability Determinations & Functionality Assessments, Revision 4
OP-HC-108-115-1001, Operability Assessment and Equipment Control Program, Revision 28
HC.OP-IS.BJ-0001, HPCI Main and Booster Pump Set – 0P207 and 0P217 – Inservice Test, Revision 63
MA-AA-734-461, Bolt Torquing and Bolting Sequence Guidelines, Revision 2
HC.MD-PM.FC-0001
HC.OP-ST.BD-0004
CC-AA-11, Nonconforming Material, Parts, or Components, Revision 5
CC-AA-309-1001, Guidelines for Preparation and Processing Design Analyses, Revision 6
HC.OP-EO.ZZ-0102, Containment Control, Revision 12
HC.OP-EO.ZZ-0313, Suppression Chamber make-up using RCIC, Revision 2
HC.OP-EO.ZZ-0317, Suppression Chamber level reduction using RCIC, Revision 4
LS-AA-120, Issue Identification and Screening Process, Revision 13

OP-AA-108-115, Operability Determinations & Functionality Assessments, Revision 4
HC.OP-LR.ZZ-0003, Leakage Test of Safety/Relief Valve Accumulators, Revision 5
HC.OP-IS.AB-0102, Main Steam System Valves – Inservice Test, Revision 24

Notifications (*NRC-identified)

20367157	20386088	20542008	20552369	20553846	20595338
20661350	20661658	20672793	20672823	20674118	20678611
20678792	20679385	20682603	20682793	20682823	20683767*
20684009	20684132*	20684134	20684147	20684239	20684902
20685229	20689441	20689461	20689561	20689847	20690067
20690750	20691294	20691769			

Maintenance Orders/Work Orders

30274989	50149377	50149976	50162256	50162921	50167721
50173206	60080713	60122715	60122940	60123401	70045899
70076318	70089912	70127666	70134244	70136886	70169453
70172377	70174682	70175001	70176529	80097504	80113547
80116133					

Drawings
M-49-1, Sheet 1, Reactor Core Isolation Cooling, Revision 30
M-50-1, Sheet 1, RCIC Pump Turbine, Revision 30
VTD PN0-E51-4010-0009, RCIC System, Revision 15

Miscellaneous
10855-D3.34, Design, Installation and Test Specification for Reactor Core Isolation Cooling System for the Hope Creek Generating Station, Revision 7
Calculation C-0044, RCIC Pump Suction and Discharge, Revision 4
Calculation C-0045, RCIC Pump Suction and Discharge, Revision 5
Calculation C-0046, RCIC Pump Suction and Discharge, Revision 4
HC.MSPI-001, Hope Creek Generating Station Mitigating Systems Performance Index Basis Document, Revision 7
HC OPEVAL 15-003, Masterpact Breaker Model NW Subjected to Elevated EMI/RFI within 1E Switchgear Rooms, Revision 0

Section 1R18: Plant Modifications

Procedures
HC.MD-GP.ZZ-0109, Disassembly, Inspection, Reassembly of Fisher Moisture Separator Dump Valves, Revision 0
CC-AA-112, Temporary Configuration Changes, Revision 14
CC-AA-112-1001, Temporary Configuration Change Implementation T&RM, Revision 4
HC.OP-AR.KJ-0003, Diesel Generator Remote Engine Control Panel 1BC423, Revision 24
LS-AA-104, 50.59 Review Process, Revision 6
LS-AA-104-1000, 50.59 Resource Manual, Revision 8
LS-AA-104-1002, 50.59 Applicability Review Form, Revision 3

Notifications

20643677	20643678	20662605	20684610	20685616	20685828
20693590	20694103	20694598			

Maintenance Orders/Work Orders

30272979	30272980	60118071	60118072	60119799	60120700
60124176	70161698	70168162	80113622		

Drawings
M-09-1, Circulating Water, Sht. 1, Revision 16
M-09-1, Circulating Water, Sht. 2, Revision 12

Modifications

80110856	80112166	80112615

Miscellaneous
Vendor Technical Document Drawing, PM018-0366, Sheet 25, Revision 10

Section 1R19: Post-Maintenance Testing

Procedures
HC.OP-IS.AB-0102, Control Room Data Sheet, Main Steam System Valves – Inservice Test, Revision 2
IST-HC-INT3-PROGRAM PLAN, Hope Creek Inservice Testing Plan, Revision 6
HC.MD-GP.ZZ-0015, Battery Equalizing Charge, Revision 23
HC.MD-ST.PK-0002, 125 Volt Quarterly Battery Surveillance, Revision 40
HC.MD-ST.PK-0007, 125 Volt Station Batteries 18 Month Service Test Using BCT-2000 with Windows Software and Associated Surveillance Testing, Revision 7
MA-AA-716-012, Post-Maintenance Testing, Revision 20
HC.OP-IS.BJ-0001, HPCI Main and Booster Pump Set – 0P207 and 0P217 – Inservice Test, Revision 63
MA-AA-734-461, Bolt Torquing and Bolting Sequence Guidelines, Revision 2
HC.OP-IS.KJ-0006, Integrated Emergency Diesel Generator 1BG400 Test – 18 months, Revision 47
SH.MD-GP.ZZ-0001, Snubber Removal and Installation, Revision 8
ER-AA-330-010, Snubber Functional Testing, Revision 6
HC.OP-SO.BC-0001, Residual Heat Removal System Operation, Revision 53
HC.OP-ST.SV-0006, A and C Channel Remote Shutdown Control Operability, RSP Transfer – 18 Month, Revision 4
HC.OP-IS.BE-0001, A and C Core Spray Pumps – AP206 and CP206 – In-service Test, Revision 49
HC.OP-ST.GU-0005, FRVS Operability Test (Single Recirculation Fan Method) – Monthly, Revision 4
HC.OP-ST.KJ-0004, Emergency Diesel Generator 1DG400 Test – Monthly, Revision 76

Notifications (*NRC-identified)

20386088	20557762	20624453	20682793	20682823	20683700
20684009	20684049	20684132*	20684133	20684134	20684147
20684239	20684554	20684564	20684779	20685523	20685525
20685619*	20685862	20686398	20686534	20687437*	20688594
20691508	20693301	20693383	20693843	20694009	20694077
20694227					

Maintenance Orders/Work Orders

30261243	30269307	30269308	50161815	50162634	50173442
60080713	60116859	60119065	60122629	60122715	60122798
60123016	70089912	70165179	70168678	70173642	70174682
70175101	70175408	70175591	80097504	80106344	80113547
80114150	80114265				

Miscellaneous

Hope Creek Engineering List of Functional and Visual Testing Results for Snubbers during RF17, RF18, and RF19, dated June 6, 2015

Section 1R20: Refueling and Other Outage Activities

Procedures

OP-HC-108-106-1001, Equipment Operational Control, Revision 4
OU-HC-105, Shutdown Safety Management Program
HC.IC-FT.SN-0009, ADS and Safety Relief Valve Operability Test, Revision 6
HC.MD-CM.AB-0006, Main Steam Safety / Relief Valve Removal and Installation, Revision 25
HC.MD-PM.FC-0001, Reactor Core Isolation Cooling (RCIC) Steam Turbine Inspection and P.M., Revision 29
HC.OP-ST.BD-0004, RCIC System Response Time and Flow Test – 18 Months, Revision 8
HC.OP-ST.KJ-0007, ‘C’ EDG LOP/LOCA Test
HC.OP-IS.ZZ-0001, Inservice System Leakage Test
HC.RP-TI.XX-0001, Primary Containment (Drywell) Entries, Revision 31
HC.RP-TI.XX-0002, Suppression Pool (Torus) Entries, Revision 10
LS-HC-1000-1001, Hope Creek Generating Station Surveillance Frequency Control Program List of Surveillance Frequencies, Revision 6
MA-AA-796-024, Scaffold Installation, Inspection, and Removal, Revision 14
HC.OP-AR.ZZ-0003, Overhead Annunciator Window Box A4, Revision 21
HC.OP-DL.ZZ-0026, Surveillance Log, Revision 148
HC.OP-GP.ZZ-0002, Primary Containment Closeout, Revision 16
HC.OP-IO.ZZ-0001, Refueling To Cold Shutdown, Revision 32
HC.OP-IO.ZZ-0003, Startup From Cold Shutdown To Rated Power, Revision 106
HC.OP-IO.ZZ-0004, Shutdown From Rated Power To Cold Shutdown, Revision 99
HC.OP-IO.ZZ-0005, Cold Shutdown To Refueling, Revision 38
HC.OP-IO.ZZ-0006, Power Changes During Operation, Revision 57
HC.OP-IO.ZZ-0009, Refueling Operations, Revision 39
HC.OP-SO.BC-0002, Decay Heat Removal Operation, Revision 33
HC.OP-SO.BC-0003, RHR Alternate Cooling Modes, Revision 6
LS-AA-106, Plant Operations Review Committee, Revision 5
LS-AA-119, Fatigue Management and Work Hour Limits, Revision 11
LS-AA-119-1001, Fatigue Management, Revision 4
LS-AA-119-1002, Scoping of Work Hour Limits, Revision 6
LS-AA-119-1003, Calculating Work Hours, Revision 7
OP-AA-108-108, Unit Restart Review, Revision 12
OP-AA-108-108-1001, Drywell/Containment Closeout, Revision 3
OP-AA-108-114, Post Transient Review, Revision 4
OP-HC-108-102, Management of Operations with the Potential to Drain the Reactor Vessel, Revision 4
OP-HC-108-114-1001, Hope Creek Post-Trip Data Collection Guidelines, Revision 9
OU-AA-101-1006, Outage Management Risk and Impact Assessment, Revision 2
OU-HC-105, Shutdown Safety Management Program – Hope Creek Annex, Revision 4

Notifications (*NRC-identified)

20592410	20611606	20628328	20681458	20683572	20683572
20684598	20684642	20684708	20684710	20684714	20684715
20684717	20684797*	20684826	20684827	20684828	20684829
20684861	20684862	20684863	20684895	20684897	20684902
20684906	20684937	20684938	20684939	20684947	20684948
20684949	20685001	20685032	20685036	20685049	20685082
20685195	20685210	20685297	20685300	20685317	20685336
20685338	20685357	20685407	20685497	20685609	20685616
20685633	20685815	20685842	20685949	20686027	20686070
20686079	20686081	20686106	20686147	20686355	20686355
20686405	20686471	20686602	20686673	20686699	20686821
20686827	20686841	20687007	20687034	20687131	20687134
20687177	20687234	20687244	20687245	20687246	20687287
20687413	20687413*	20687433*	20687446	20687447	20687448
20687480	20687535	20687597	20687627	20687632	20687638
20687648	20687699*	20688031	20688033	20688114	20688138
20688164	20688175	20688224	20688257	20688348	20688370
20688521	20688662	20688706	20688758	20688763	20688773*
20688828	20688834	20689024	20689064*	20689096	20689441
20689847	20690001	20690067	20690094		

Maintenance Orders/Work Orders

30087391	30205518	60118109	60122829	60122902	60123401
70134244	70149472	70159885	70174987	70175466	70175548
70176529	70176268	80108395	80114126	80114150	80114188

Drawings

M-43-1, Sheet 1, Reactor Recirculation System, Revision 34
M-51-1, Sheet 1, Residual Heat Removal, Revision 45
M-51-1, Sheet 2, Residual Heat Removal, Revision 42
P-1705-1, Sheet 1, Piping Area Drawing Reactor Bldg – Area-17 Partial Plan El. 121’-7½”, Revision 2

Miscellaneous

Hope Creek Cycle 20 Core Loading Plan, Revision 2
Hope Creek RFO 19 Core Verification Video, dated April 27, 2015
REMA 2015-0035, Reactivity Maneuver Plan for Shutdown to RF19, Revision 0
REMA 2015-0039, Reactivity Maneuver Plan for Cycle 20 Initial Startup Through ~23%, Revision 0
RF19 Shutdown Safety Management Plan P1.23a, Revision 2
2015-0035, Reactivity Maneuver Plan – Shutdown to RF19, April 10, 2015
Complex Troubleshooting for RCIC Pump / Turbine not Spinning as Expected on May 9, 2015
Hope Creek List of Critical Preventative Maintenance Deferrals for RF19, April 16, 2015
Hope Creek Generating Station RF19 ORAM Report, February 26, 2015
Hope Creek Long Term Trends, April 7, 2015
Hope Creek OPEVAL 15-006, RCIC Pump / Turbine did not Rotate, Revision 1
RF19 Shutdown Safety Management Plan, Revision 2
VTD 901232, TR-A100-4, Quick Stem Sensor (QSS) Installation Application Guide, Revision 1

Section 1R22: Surveillance Testing

Procedures

HC.OP-IS.KJ-0006, Integrated EDG 1BG400 Test – 18 months, Revision 47
HC.OP-IS.ZZ-0001, Inservice System Leakage Test of the Reactor Coolant Pressure Boundary, Revision 44
HC.OP-IS.ZZ-0001, Inservice System Leakage Test of the Reactor Coolant Pressure Boundary, Revision 44
HC.RE-ST.BF-0001, Control Rod Scram Time Surveillance, Revision 34
OU-AA-335-015, PSEG Nuclear L.L.C VT-2 Visual Examination, Revision 5
HC.OP-ST.KJ-0008, Integrated Emergency Diesel Generator 1DG400 Test – 18 Months, Revision 48
HC.OP-IS.BJ-0001, HPCI Main and Booster Pump Set – 0P204 and 0P217 – Inservice Test, Revision 63

Notifications (*NRC-identified)

20497856	20686017	20686090	20686299	20686303	20686397
20686398	20686432	20686450*	20686487	20686734	20686827
20686837*	20686838*	20686841	20686842	20687059*	20687246
20688533	20692433	20692544			

Maintenance Orders/Work Orders

30264349	30264349	50162660	50162739	50170035	60123016
60124357	70177331	80106344	80114188		

Section 1EP6: Drill Evaluation

Miscellaneous

Hope Creek 15-01, Drill Plan dated June 23, 2015

Section 2RS1: Access Control to Radiologically Significant Areas

Procedures

RP-AA-210, Dosimetry Issuance, Usage, and Control, Revision 13
RP-AA-401-1002, Instructions for Establishing Electronic Dosimeter Alarm Setpoints, Revision 2
RP-AA-400, ALARA Program, Revision 6
RP-AA-401, Operational ALARA Planning, Revision 13
RP-AA-401-1001, Special Instructions for Highly Radioactive In-core Components, Revision 0
RP-AA-460, Controls for High and Very High radiation Areas, Revision 17
RP-AA-461, Controls for Contaminated Water Diving Operations, Revision 3
RP-AA-462, Control for Radiographic Operation
RP-AB-460, TIP Area Access Controls, Revision 2
RP-HC-4003, Reactor Start-up and Shutdown radiological Controls, Revision 1
HC-RP-TI.XX-001(Q), Primary Containment Drywell Entry, Revision 30
HC.RP-TI.XX-0003(Q), Reactor Cavity, Fuel Pool and Drywell Special Evolution, Revision 25
NF-AA-390, Spent Fuel Pool Material Control, Revision 5
WC-AA-105, Work Activity Risk Management, Revision 3

Documents

Current Source Term
Collective Dose Report
Corrective Action Documents (various)

Section 2RS2: Occupational ALARA Planning and Controls

Procedures
CY-AB-120-1200, Chemistry Refuel Outage Readiness, Revision 3
RP-AA-400, ALARA Program, Revision 6
RP-AA-401, Operational ALARA Planning, Revision 13
WC-AA-105, Work Activity Risk Management, Revision 3
RP-AA-401-1001, Special Instructions for Highly Radioactive In-core Components, Revision 0
RP-AA-460, Controls for High and Very High radiation Areas, Revision 17
RP-AA-461, Controls for Contaminated Water Diving Operations, Revision 3
RP-AB-460, TIP Area Access Controls, Revision 2
RP-HC-4003, Reactor Start-up and Shutdown Radiological Controls, Revision 1
HC-RP-TI.XX-001(Q), Primary Containment Drywell Entry, Revision 30
HC.RP-TI.XX-0003(Q), Reactor Cavity, Fuel Pool and Drywell Special Evolution, Revision 25

Documents
Hope Creek Outage Task Breakdown and Dose Estimates
Various ALARA Plans (RWPs 8, 10, 12, 17)
Daily Occupational Dose Reports

Section 2RS3: In-plant Airborne Radioactivity Control and Mitigation

Procedures
RP-AA-303, Personnel Air Sampling, Revision 1
RP-AA-13, Respiratory Protection Program Description
RP-AA-440, Respiratory protection Program, Revision 10
RP-AA-376, Radiological Posting, Labeling, and Marking, Revision 7
RP-AA-300, Radiological Survey Program, Revision 5
RP-AA-220, Bioassay Program, Revision 9
RP-AA-825, Maintenance Care and Inspection of Respiratory Protection Equipment, Revision 5
HC.RP-RW. KG-0001(Q), Respiratory Protective Equipment Cleaning, Revision 10
RP-AA-443, Quantitative Respiratory Fit Testing, Revision 7
WC-AA-105, Work Activity Risk Management, Revision 3

Documents
Current Source Term
Collective Dose Report
Hope Creek Outage Task Breakdown and Dose Estimates
Daily Occupational Dose Reports
Air Sample Results
Breathing Air Test Data

Section 2RS4: Occupational Dose Assessment

Procedures
RP-AA-210, Dosimetry Issuance, Usage, and Control, Revision 13
RP-AA-401-1002, Instructions for Establishing Electronic Dosimeter Alarm Set-points, Revision 2
RP-AA-401-1001, Special Instructions for Highly Radioactive In-core Components, Revision 0
RP-AA-460, Controls for High and Very High radiation Areas, Revision 17
RP-AA-461, Controls for Contaminated Water Diving Operations, Revision 3
WC-AA-105, Work Activity Risk Management, Revision 3

RP-AA-303, Personnel Air Sampling, Revision 1
RP-AA-440, Respiratory Protection Program, Revision 10
RP-AA-300, Radiological Survey Program, Revision 5
RP-AA-220, Bioassay Program, Revision 9

Documents
Current Source Term
Collective Dose Report
Hope Creek Outage Task Breakdown and Dose Estimates
Various ALARA Plans (RWPs 8, 10, 12, 17)
Daily Occupational Dose Reports
Corrective Action Documents (various)

Section 2RS5: Radiation Monitoring Instrumentation

Procedures
HC.RP-TI.SP-0008(Q), Operation of North Plant Vent Skid, Revision 21
HC.RP-TI.SP-0018(Q), Sampling of the North Plant Vent Skid, Revision 8
HC.OP-AR.SP-001(Q), Radiation Monitoring System Operation, Revision 26
NC.RS-TI.ZZ-0594(Q), Generic Calibration of New and Non-Standard Instrumentation, Revision 3
CY-AA-130-205, Radiochemistry Quality Control, Revision 0
HC.CH-RC.ZZ-0025(Q), Gamma Spectroscopy Analysis, Revision 0

Documents
Supervisory Directive GEM 5
Instrument Calibration data
Process radiation Monitoring System Health Reports (2014, 2015)
Calibration Reports (Ge-Li Detector 1)

Section 2RS6: Radioactive Gaseous and Liquid Effluent Treatment

Procedures
EN-AA-170-200, Radioactive Effluent Control Program, Revision 0
HC.RP-ST.ZZ-004(Q), Gaseous Effluent Surveillance, Revision 2
EN-AA-170-400, Radiological Groundwater Protection Program Implementation, Revision 0
EN-AA-170-200, Annual Radioactive Effluent Release Report, Revision 0
CY-AA-130-205, Radiochemistry Quality Control, Revision 0
CY-AA-130-150, Chemistry Quality Assurance, Revision 0
EN-AA-170-0100, Job Familiarization Guide for REMP, MET, RGPP and REC Program, Revision 2
HC.CH-RC.ZZ-0025(Q), Gamma Spectroscopy Analysis, Revision 0
HC.RP-ST.SP-0027(Q), Liquid Effluent Surveillance with All Monitors Operable, Revision 4
HC.RW-SO.HB-0033®, Floor Drain Sample Tank B Discharge to Cooling Tower Blowdown with the Rad Monitor Operable, Revision 1
HC.CH-TI.ZZ-0015(Q), Radioactive Liquid Effluent Permits, Revision 36
HC.CH-TI.ZZ-0005(Q), Radioactive Gaseous Effluent Permits, Revision 21
HC.CH-TI.ZZ-0009, (Q), E-Bar Determinations, Revision 9

Documents
Audit NOSA Radiation Protection (HPC-13-08)
Audit NOSA Chemistry, Radioactive Waste, Effluent and Environmental Monitoring (HPC-14-04)
FASA Effluent Controls (20678026)
FASA Radiochemistry Quality Control (70151714)
Hope Creek performance Assessment Report (September 1 –December 31, 2014)
Hope Creek NOS Station Status Report (March 2015)
Sampling Plans (storm drains)
2013 and 2014 PSEG Hope Creek Effluent and Environmental Annual Reports
MES Report, Review of Gaseous Release Points and Dispersion Modeling Assumptions at Salem and Hope Creek Stations
Meteorological Data Inter-comparisons (2014)
Land Use Survey (August 2014)
APEX Quality Control testing (inter-comparisons APEX/CAS)
Laboratory Cross Check data (Inter and Intra)
10 CFR 50.59 Screenings
2014 Ground Water protection Program (RGPP) Report
Radioactive Release Analyses (liquid and gaseous discharges)
Effluent Alarm Set-point Determinations (South Plant Vent, Filtered Recirculation Ventilation System, Liquid Radioactive Waste, Cooling Tower Blow-down)

Section 4OA1: Performance Indicator Verification

Procedures
HC.CH-SA.RC-0002, Operation of the Reactor Building/RHR Sample Stations, Revision 18
HC.OP-DL.ZZ-0026, Surveillance Log, Revision 139
HC.OP-DL.ZZ-0026, Surveillance Log, Revision 140
HC.OP-DL.ZZ-0026, Surveillance Log, Revision 141
HC.OP-DL.ZZ-0026, Surveillance Log, Revision 142
HC.OP-DL.ZZ-0026, Surveillance Log, Revision 143
HC.OP-DL.ZZ-0026, Surveillance Log, Revision 144
HC.OP-DL.ZZ-0026, Surveillance Log, Revision 145
LS-AA-2090, Monthly Data Elements for NRC Reactor Coolant System Activity, Revision 5
LS-AA-2100, Monthly Data Elements for NRC Reactor Coolant System Leakage, Revision 6
LS-HC-1000-1001, Hope Creek Generating Station Surveillance Frequency Control Program List of Surveillance Frequencies, Revision 6
NC.CH-RC.ZZ-2525, Gamma Spectroscopy Analysis Using CAS, Revision 5

Documents
Daily Surveillance Log Data
Daily Dose Equivalent Iodine-131 Sample Data, April, 2014 – March, 2015
Monthly Data Elements for NRC Reactor Coolant System Leakage Data Sheets, April, 2014 – March, 2015

Section 4OA2: Problem Identification and Resolution

Procedures
MA-AA-734-461 Torque Procedure
HC.OP-IS.BE-0001, A & C Core Spray Pumps – AP206 and CP206 – In-service Test, Revision
MA-AA-716-010-1000,
MA-AA-716-006, Control of Lubricants Program, Revision 11
MA-AA-716-230-1001, Oil Analysis Interpretation Guideline, Revision 11

MA-AA-716-230-1004, Lubricant Sampling Guideline, Revision 3
SH-PBD-AMP.XI.M39, Lubricating Oil Analysis (Program Basis Document), Revision 3
HC.MD-PM.ZZ-0007, Missile Resistant and Watertight Doors P.M., Revision 10

Notifications

20173609	20173662	20425229	20490746	20542141	20543080
20593219	20617384	20670717	20670878	20670879	20670880
20671116	20674632	20675460	20680046	20683700	20683879
20684833	20685468	20685594	30185777		

Maintenance Orders/Work Orders

60120553	70036674	70156986	70159998	70171837	70174237
70175101	70175889	80103518			

Miscellaneous
OP-HC-108_115-1001 Technical Specification Action Statement Log- Verification of Operability, Revision 0, dated 2/17/12
PSE-91281, Failure Analysis of Extension Spring from Control Device for HK Breaker, dated June 26, 2015

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

Procedures
HC.OP-IS.BE-0001, A and C Core Spray Pumps – AP206 and CP206 – In-service Test, Revision 49
OP-HC-108-102, Management of Operations with the Potential to Drain the Reactor Vessel, Revision 4

Notifications
20559547 20678572 20683700

Maintenance Orders/Work Orders
50173442 70175101

Miscellaneous
NRC Enforcement Guidance Memorandum on Dispositioning BWR Licensee Noncompliance with Technical Specification Containment Requirements During Operations with a Potential for Draining the Reactor Vessel, 11-003, Revision 3

Section 4OA5: Other Activities

Notifications
20687475

Documents
NUREG-1021, “Operator Licensing Examination Standards for Power Reactors,” Revision 9, Supplement 1

Forms
NRC Form 398, “Personal Qualification Statement – Licensee”

LIST OF ACRONYMS

10 CFR	Title 10 of *The Code of Federal Regulations*
AC	alternating current
ACE	Apparent Cause Evaluation
ADAMS	Agencywide Documents Access and Management System
ALARA	As Low As is Reasonably Achievable
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
BWRVIP	Boiling Water Reactor Vessel and Internals Project
CAM	Continuous Air Monitor(ing)
CAP	Corrective Action Program
CIV	containment isolation valve
CS	Core Spray
DCP	design change packages
EDG	Emergency Diesel Generator
EGM	Enforcement Guidance Memorandum
EOC	extent of condition
EPD	Electronic Personal Dosimeter
EPRI	Electric Power Research Institute
FRVS	Filtration, Recirculation and Ventilation System
GFE	Generic Fundamentals Examination
GPI	Groundwater Protection Initiative
HCGS	Hope Creek Generating Station
HPCI	high pressure coolant injection
HRA	High Radiation Area
IMC	Inspection Manual Chapter
IRM	Intermediate Range Monitor
ISI	in-service inspection
IST	in-service test(ing)
IVVI	In Vessel Visual Inspection
kV	kilovolt
LAR	license amendment request
LER	licensee event report
LHRA	Locked High Radiation Area
LOCA	Loss of Coolant Accident
LOP	Loss of Power
NCV	Non-Cited Violation
NDE	Nondestructive Examination
NEI	Nuclear Energy Institute
NOTF	notification
NRC	Nuclear Regulatory Commission
NVLAP	National Voluntary Laboratory Accreditation Program
ODCM	Offsite Dose Calculation Manual
OPDRV	operations with the potential to drain the reactor vessel
PD	Performance Deficiency
PI	Performance Indicators
PI&R	Problem Identification and Resolution
PSEG	Public Service Enterprise Group Nuclear, LLC
RCIC	Reactor Core Isolation Cooling

RCS	Reactor Coolant System
RG	Regulatory Guide
RHR	residual heat removal
RMS	radiation monitoring system
ROP	Reactor Oversight Process
RP	Radiation Protection
RRP	Reactor Recirculation Pump
RTP	rated thermal power
RWCU	reactor water cleanup
RWP	Radiation Work Permit
SACS	safety auxiliaries cooling system
SDP	Significance Determination Process
SRM	Source Range Monitor
SRO	Senior Reactor Operator
SRV	Safety Relief Valve
SSC	structure, system, or component
SSW	station service water
STS	Standard Technical Specifications
TS	Technical Specifications
TWCU	Torus Water Cleanup
UFSAR	Updated Final Safety Analysis Report
UT	ultrasonic testing
VHRA	Very High Radiation Area
WO	work order
WT	watertight