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

August 10, 2012

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

SUBJECT: HOPE CREEK GENERATING STATION UNIT 1 - NRC INTEGRATED INSPECTION REPORT 05000354/2012003

Dear Mr. Joyce:

On June 30, 2012, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at the Hope Creek Generating Station. The enclosed inspection report documents the inspection results which were discussed on July 19, 2012, with Mr. D. Lewis, Plant Manager 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.

The report documents two findings of very low safety significance (Green). Both of these findings were determined to involve violations of NRC requirements. Additionally, a licensee-identified violation, which was determined to be of very low safety significance, is listed in this report. However, because of their very low safety significance and because they were entered into your corrective action program (CAP), the NRC is treating these findings as non-cited violations (NCVs) consistent with Section 2.3.2 of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Hope Creek Generating Station. In addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region I, and the NRC Resident Inspector at the Hope Creek Generating Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's Agencywide Document Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Arthur L. Burritt, Chief
Reactor Projects Branch 3
Division of Reactor Projects

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

Enclosure: Inspection Report 05000354/2012003
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServ

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-354
License No:	NPF-57
Report No:	05000354/2012003
Licensee:	PSEG Nuclear LLC (PSEG)
Facility:	Hope Creek Generating Station
Location:	P.O. Box 236 Hancocks Bridge, NJ 08038
Dates:	April 1, 2012 through June 30, 2012
Inspectors:	F. Bower, Senior Resident Inspector A. Patel, Resident Inspector C. Williams, Resident Inspector E. H. Gray, Senior Reactor Inspector T. Burns, Reactor Inspector R. Nimitz, Senior Health Physicist
Approved By:	Arthur L. Burritt, Chief Reactor Projects Branch 3 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000354/2012003; 04/01/2012 - 06/30/2012; Hope Creek Generating Station; Maintenance Effectiveness, Follow-Up of Events and Notices of Enforcement Discretion.

This report covered a three-month period of inspection by resident inspectors and announced inspections performed by a Senior Reactor Inspector, a Senior Health Physicist, and a Reactor Inspector. The inspectors identified two findings of very low safety significance (Green). The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, “Significance Determination Process” (SDP). The cross-cutting aspects for the findings are determined using IMC 0310, “Components Within Cross-Cutting Areas.” Findings for which the SDP does not apply may be Green, or be assigned a severity level after NRC management review. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 4, dated December 2006.

Cornerstone: Barrier Integrity

- Green. The inspectors identified a non-cited violation (NCV) of very low safety significance of 10 CFR 50, Appendix B, Criterion XI, “Test Control,” because PSEG conducted unacceptable preconditioning of the reactor building to torus vacuum relief valve. Specifically, PSEG’s surveillance test procedure for these valves cycled the valve (H1GS-1GSPSV-5032) prior to recording the as-found opening setpoint required to meet Technical Specification (TS) Surveillance Requirement (SR) 4.6.4.2.b.2.a. PSEG’s immediate corrective actions included revising the surveillance test procedure to record the as-found setpoint before cycling the valve manually. The violation was entered into the corrective action program (CAP) as notification 20554080.

  The performance deficiency was more than minor because it was associated with the procedure quality attribute of the Barrier Integrity Cornerstone and affected the cornerstone objective of providing reasonable assurance that physical design barriers (containment) protect the public from radionuclide releases caused by accidents or events. Specifically, preconditioning of the reactor building to torus vacuum relief opening setpoint could mask its actual as-found condition and result in an inability to verify its operability and potentially make it difficult to determine whether the vacuum breaker would perform its intended safety function during an event. The inspectors evaluated the finding using IMC 0609, Attachment 4, “Initial Screening and Characterization of Findings,” and determined the finding was of very low safety significance (Green) because it was not a degradation of the radiological barrier function provided for the control room, auxiliary building, spent fuel pool, or standby gas treatment system, did not represent a degradation of the barrier function of the control room against smoke or toxic atmosphere, did not represent an actual open pathway in the physical integrity of reactor containment and heat removal components, and did not involve an actual reduction in function of hydrogen igniters in the reactor containment. The finding had a cross-cutting aspect in the area of problem identification and resolution, corrective action component, because PSEG did not thoroughly evaluate a prior problem such that the problem resolution addressed the extent of condition. Specifically, PSEG’s extent of condition for notification 20370021, Potential Preconditioning BJHV-F004, did not go beyond operations’ procedures and review maintenance procedures for unacceptable preconditioning. Therefore, PSEG did not identify the unacceptable preconditioning of the reactor building to torus vacuum relief valve opening setpoint because the surveillance test was in a maintenance procedure. (P.1(c)) (Section 1R12)

Green. The inspectors identified an NCV of very low safety significance of TSs 3.3.1 and 6.8.1 because PSEG's written procedure (HC.IC-CC.SE-0032) was not adequately established and implemented for performing the weekly channel test and calibration of the flow biased APRMs that input into the simulated thermal power upscale RPS trip. Specifically, the procedure provided inadequate instructions for calculating total reactor recirculation drive flow while in single loop operations (SLO). PSEG's corrective actions included revision of the appropriate procedures and development of a schedule template (including required surveillances) for entry into and return from SLO. The violation was entered into the CAP as notification 20549760.

The performance deficiency was more than minor because it was associated with the procedure quality attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, incorrect calibration of the APRM flow units resulted in the APRM flow biased setpoint being non-conservative and exceeding the associated TS limiting safety system setpoint (LSSS) allowable value for a period of time that was considered a condition prohibited by TS. The inspectors performed a Phase I screening of the finding using IMC 0609, Attachment 0609.04, Table 4a, Mitigating Systems cornerstone and determined the issue was of very low safety significance (Green) because the finding was not a design or qualification deficiency, did not result in an actual loss of safety function, and was not potentially risk significant for external events. The finding had a cross-cutting aspect in the area of human performance, resources component, because PSEG did not ensure that a TS-required RPS calibration procedure was complete, accurate, and adequate to assure nuclear safety. Specifically, the formula provided in the APRM flow unit summer procedure that calculated the drive flow was incorrect. The formula provided in the procedure was for dual loop operation, not for SLO. (H.2(c)) (Section 4OA3.2)

Other Findings

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

REPORT DETAILS

Summary of Plant Status

The Hope Creek Generating Station began the inspection period at approximately 98 percent of rated thermal power (RTP) in end-of-cycle coastdown where it generally remained until the unit was manually shutdown on April 13, 2012, to start Hope Creek's planned 17th refueling outage (H1R17). On May 7, 2012, the reactor mode switch was placed in start-up, criticality was reached on May 8, 2012, and the unit was synchronized to the grid on May 9, 2012. On May 12, 2012, the unit was returned to full power and remained at or near for the duration of the inspection period except for brief periods to support planned testing and rod pattern adjustments.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01 - 2 samples)

.1 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 switchyard summer readiness letter, and walking down portions of the offsite and alternate AC power systems including the 500 kilovolt and 13.8 kilovolt switchyards. Documents reviewed for each section of this inspection report are listed in the Attachment.

.2 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 emergency diesel generators (EDGs), circulating water, and service water (SW). The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR) and TSs 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.

b. Findings

No findings were identified.

1R04 Equipment Alignment

Partial System Walkdowns (71111.04Q - 4 samples)

a. Inspection Scope

The inspectors performed partial walkdowns of the following systems:

- A and C EDGs while B EDG out-of-service on April 17, 2012
- B residual heat removal (RHR) shutdown cooling while A RHR shutdown cooling out-of-service on April 27, 2012
- D SW pump while B SW out-of-service on May 15, 2012
- Reactor core isolation cooling (RCIC) while high pressure coolant injection (HPCI) out-of-service on June 5, 2012

The inspectors selected these systems based on their risk-significance for the current plant configuration or following realignment. The inspectors reviewed applicable procedures, system diagrams, the UFSAR, TSs, work orders, condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have impacted system performance of their intended safety functions. The inspectors also performed field walkdowns of accessible portions of the systems to verify system components and support equipment were aligned correctly and were operable.

b. Findings

No findings were identified.

1R05 Fire Protection

Resident Inspector Quarterly Walkdowns (71111.05Q - 5 samples)

a. Inspection Scope

The inspectors conducted tours of the areas listed below to assess the material condition and operational status of fire protection features. The inspectors verified that PSEG controlled combustible materials and ignition sources 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-413, HPCI pump & turbine room
- FRH-II-433, A & C safety auxiliaries cooling system (SACS) rooms
- FRH-II-432, B & D SACS rooms
- FRH-II-423, A RHR heat exchanger room
- FRH-II-442, Containment instrument gas compressor rooms, filtration recirculation and ventilation system (FRVS) unit areas, and steam vent and equipment area

b. Findings

No findings were identified.

1R08 Inservice Inspection (ISI) (71111.08 - 1 sample)

a. Inspection Scope

The purpose of this inspection was to assess the effectiveness of PSEG's ISI activities for monitoring degradation of reactor pressure vessel (RPV) internals, reactor coolant system boundary, risk-significant piping system boundaries, and the containment boundary. The inspectors assessed the ISI activities using requirements and acceptance criteria for component examination specified in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, and applicable NRC regulatory requirements.

The inspectors selected a sample of nondestructive examination (NDE) activities and reviewed the inspection reports developed from performance of those examinations to verify the test activities comply with the requirements of ASME Section XI and applicable regulatory requirements. The sample selection was based on the inspection procedure objectives and risk priority of those components and systems where degradation could result in a significant increase in risk of core damage in the event of loss of structural integrity or pressure retaining capability.

The inspectors verified by documentation review that test procedures and examiner qualifications were current and in accordance with the ASME Code requirements.

Also, the inspectors reviewed examiner qualifications for use of the performance demonstration initiative manual ultrasonic test (UT) procedures. The inspectors selected a sample of notifications and corrective actions for review of PSEG's effectiveness in the identification and resolution of relevant indications discovered during ISI activities. The inspectors' review of selected samples of nondestructive testing included the following:

Manual UT examination of carbon steel pipe to elbow butt weld in the core spray (CS) system using UT procedure GEH-PDI-UT-1, Version 8. The inspectors verified the examination was performed in accordance with ASME Section XI and 100 percent weld coverage was achieved. The inspectors confirmed the examination was performed with Work Order 50137098 and the results documented in Report No. UT-12-025. No recordable indications were identified.
Magnetic particle test (MT) of integral attached lug, item C-C/C3.20, welded to carbon steel pipe in the HPCI system, component 1-FD-20HBB-9LG (1-8) using MT procedure

OU-AA-335-003, Revision 2. The inspectors confirmed the examination results were documented in Report No. MT-12-003. No recordable indications were identified.

Liquid Penetrant test (PT) of B recirculation loop outside radius at one inch instrument line to nozzle weld of component 1-BB-1CCA-225-1 using PT procedure OU-AA-335-002, Revision 2. The inspectors confirmed the examination results were documented in Report No. 12-006. No indications were identified.

Radiographic test (RT) examination of carbon steel grove butt weld, item HC-1-P-BD-203-13 in the RCIC piping system using RT procedure OU-AA-335-005, Revision 1. No recordable indications were identified.

The inspectors reviewed visual test (VT)-1 and VT-3 examination using NDE-VT-003, Revision 9, and VT-005, Revision 8, of the RPV internals consisting of jet pump main wedges, auxiliary wedges, steam dryer, shroud baffle support plate, and additional structural members. Various welds of in-vessel CS piping were also visually inspected. Indications noted in the weld of the shroud support plate to the inside diameter of the RPV were identified and recorded for characterization and disposition. The indications were documented in notification 20556380.

The inspectors selected two ASME Section XI repair/replacement plans for review where welding was performed. The review was performed to confirm that appropriately qualified weld procedures and welders were assigned this work and that essential welding parameters were indicated as “hold points” on the weld traveler. The inspectors noted these “hold point” attributes were examined by inspection personnel and documented on the weld traveler. The inspectors reviewed base materials and weld filler metal specifications to verify they were in accordance with ASME Code requirements. Also, the inspectors reviewed documentation that the completed weld examinations were performed in accordance with the ASME Section XI code requirements. The two ASME Section XI repair/replacement activities reviewed were:

Work Order 60088924: This work order governs replacement of two valves and associated piping in the Main Steam system (system AB). This replacement consisted of performing an ASME Section XI replacement of valves 1-AB-V063 and V064. The inspectors verified replacement activity was governed by ASME Section XI, Safety Class 1 and Seismic Class 1. The inspectors verified the welding was performed by qualified welders using qualified welding procedures and weld filler materials meeting the requirements of ASME Section XI. The inspectors verified final acceptance of the replacement welds was based on satisfactory liquid PT, pressure test, and visual surface examination. No recordable indications were identified and no leakage was noted.

Work Order 60101260: This work order governs the installation of a portion of RCIC piping ASME Class 2 and Seismic Class 1. The inspectors reviewed the four welds that were made to replace the existing portion of failed carbon steel suction piping. The inspectors verified the replacement welding was governed by the requirements of ASME Section XI with final acceptance as specified in ASME Section III. The inspectors confirmed the final acceptance of these replacement welds was based on satisfactory nondestructive testing (PT) and system pressure test (VT-2). The inspectors verified that appropriate verification of weld “hold points” was established on the replacement work instruction. No recordable indications were identified and no leakage was noted.

The inspectors performed a walkdown to view portions of the primary containment and additional structural members attached to the liner for assessment of the condition of the protective coating. The inspectors performed this visual assessment of limited locations at the equipment hatch entrance elevation. The assessment included the extent of any peeling, blistering, coating loss or other damage or degradation as a result of corrosion, foreign material impact, or lack of maintenance. Also, the inspectors evaluated coating integrity at accessible locations where the primary containment liner intersects the containment floor. The evaluation was consistent with the requirements provided in ASME Section XI.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11Q - 1 sample)

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

a. Inspection Scope

On April 13, 2012, the inspectors observed the planned power reduction for Hope Creek refueling outage 17 and subsequent operational placement of the mode switch from run to shutdown. During these control room observations, the inspectors assessed the adequacy of: procedure use, crew communications, human performance tool use, supervisory oversight, and coordination of activities between work groups to verify that PSEG's established expectations and standards were met.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12 - 1 samples)

a. Inspection Scope

The inspectors reviewed the samples listed below to assess the effectiveness of maintenance activities on structure, system, and component (SSC) performance and reliability. The inspectors reviewed CAP documents, maintenance work orders, and maintenance rule program 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.

- Reactor building to torus vacuum relief valves (Notification 20554080)

b. Findings

Introduction. The inspectors identified a Green NCV of 10 CFR 50, Appendix B, Criterion XI, “Test Control,” because PSEG conducted unacceptable preconditioning of the reactor building to torus vacuum relief valve. Specifically, PSEG’s surveillance test procedure for these valves cycled the valve (H1GS-1GSPSV-5032) prior to recording the as-found opening setpoint required to meet TS SR 4.6.4.2.b.2.a.

Description. The function of the reactor building to torus vacuum relief valves is to relieve vacuum when primary containment depressurizes below reactor building pressure. PSEG’s surveillance test HC.MD-ST.GS-0002, “Reactor Building to Torus Vacuum Relief Valve 18 Month Testing,” Revision 8, is used to satisfy TS SR 4.6.4.2.b.2.a. This surveillance requirement verifies that each vacuum relief valve opens at a setpoint of less than or equal to 0.25 psid.

On April 10, during a surveillance test review, the inspectors identified what they believed to be unacceptable preconditioning during performance of surveillance test procedure HC.MD-ST.GS-0002. Specifically, procedure section 5.2, “Vacuum Relief Valve Visual Inspection,” step 5.2.3, states, “Check pallet operation for interference or binding of hinge by manually operating pallet several times.” While the next section, section 5.3, “Vacuum Relief Valve As-Found Setpoint Checks,” step 5.3.1, “VERIFY Vacuum Relief Valve As-Found Setpoint,” records the as-found opening setpoint of the relief valve and is used to verify TS SR 4.6.4.2.b.2.a.

The inspectors reviewed regulatory positions and guidance regarding preconditioning, including NRC IMC Part 9900: Technical Guidance, “Maintenance-Preconditioning of Structures, Systems, and Components before Determining Operability,” and PSEG’s procedure PP-AA-3001, “Position Paper on Preconditioning.” IMC Part 9900 and PP-AA-3001 states, in part, that unacceptable preconditioning is defined as the alteration, variation, manipulation, or adjustment of the physical condition of a SSC before or during a TS surveillance that will alter one or more SSCs operational parameters, which results in acceptable test results. Such changes could mask the actual as-found condition of the SSC and possibly result in an inability to verify the operability of the SSC. In addition, unacceptable preconditioning could make it difficult to determine whether the SSC would perform its intended function during an event in which the SSC might be needed.

Based on the review of the technical guidance, the inspectors determined that, as written and implemented, HC.MD-ST.GS-0002, which included steps that obtained as-found data only after manually cycling the valve several times, was unacceptable preconditioning of the valves it tested. Specifically, manually cycling the valve several times before the actual test altered the physical condition of the valve and could have masked an unacceptable condition.

PSEG entered the issue into their CAP (Notification 20554080) and evaluated the inspectors concerns. PSEG verified that the procedure contained unacceptable preconditioning steps and revised the testing sequence.

The inspectors reviewed PSEG procedure LS-AA-125, “Corrective Action Program,” which defines “extent of condition” as the extent to which the identified condition has the potential to impact other plant processes, equipment, or human performance in the same manner as identified in the condition report. The inspectors found that in November 2009 (Notification 20370021/70085313 operation 110), in response to a potential unacceptable preconditioning of the HPCI condensate storage suction valve, PSEG's extent of condition reviewed operations department procedures to determine unacceptable preconditioning. Specifically, notification 20370021/70085313 operation 110 stated that PSEG "performed the expanded extent of condition review for all quarterly inservice test (IST) valve surveillances and additionally performed a review of all cold shutdown IST valve surveillances and all quarterly pump IST surveillances, and a sample population of over 50 percent of the operations department surveillance test procedures." However, this extent of condition review narrowly focused on IST procedures and operations' procedures used to test TS SR. Therefore, the extent of condition review did not identify the impact of unacceptable preconditioning in other plant processes such as maintenance procedures that are used to test TS SR, and specifically test procedure HC.MD-ST.GS-0002 that tests the reactor building to torus vacuum relief valve opening setpoint in accordance with TS SR 4.6.4.2.b.2.a.

Analysis. The inspectors determined that PSEG's performance of unacceptable preconditioning prior to recording the as-found setpoint of the reactor building to torus vacuum relief valve opening setpoint was a performance deficiency. The performance deficiency was more than minor because it was associated with the procedure quality attribute of the Barrier Integrity Cornerstone and affected the cornerstone objective of providing reasonable assurance that physical design barriers (containment) protect the public from radionuclide releases caused by accidents or events. Specifically, preconditioning of the reactor building to torus vacuum relief valve's opening setpoint could mask its actual as-found condition and result in an inability to verify its operability and potentially make it difficult to determine whether the vacuum breaker would perform its intended safety function during an event. The inspectors evaluated the finding using IMC 0609, Attachment 4, "Initial Screening and Characterization of Findings," and determined the finding was of very low safety significance (Green) because all of the containment barrier questions in Table 4a were answered no. Specifically, the finding was not a degradation of the radiological barrier function provided for the control room, or auxiliary building, or spent fuel pool, or standby gas treatment system, did not represent a degradation of the barrier function of the control room against smoke or toxic atmosphere, did not represent an actual open pathway in the physical integrity of reactor containment and heat removal components, and did not involve an actual reduction in function of hydrogen igniters in the reactor containment.

The finding had a cross-cutting aspect in the area of problem identification and resolution, corrective action component, because PSEG did not thoroughly evaluate a prior problem such that the problem resolution addressed the extent of condition. Specifically, PSEG's extent of condition for notification 20370021, Potential Preconditioning BJHV-F004, did not go beyond operations' procedures and review maintenance procedures for unacceptable preconditioning. Therefore, PSEG did not identify the unacceptable preconditioning of the reactor building to torus vacuum relief valves' opening setpoint because the surveillance test was in a maintenance procedure. (P.1(c))

Enforcement. 10 CFR 50, Appendix B, Criterion XI, "Test Control," requires, in part, that a test program shall be established to assure that all testing required to demonstrate that SSCs will perform satisfactorily in service, is identified and performed in accordance with written test procedures, and incorporate the requirements and acceptable limits contained in applicable design documents. Contrary to the above, on April 10, 2012, PSEG did not establish a test program that assured that all testing required to demonstrate that the reactor building to torus vacuum relief valves will perform satisfactorily in service was identified and performed in accordance with written test procedures, and incorporated the requirements and acceptable limits contained in applicable design documents. Specifically, due to an inadequate test sequence that resulted in an unacceptable preconditioning of the reactor building to torus vacuum relief valves, the testing performed in accordance with PSEG's written test procedures did not determine the as-found condition of the valves. This issue was entered into the CAP as notification 20554080, and PSEG's immediate corrective actions included revising the surveillance test procedure to record the as-found setpoint before cycling the valve manually. Because the violation was of very low safety significance (Green) and has been entered into the CAP, this violation is being treated as an NCV, consistent with Section 2.3.2.a of the NRC Enforcement Policy. **(NCV 05000354/2012003-01, Preconditioning of the Reactor Building to Torus Vacuum Relief Valves)**

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 - 4 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. As applicable for each activity, the inspectors verified that PSEG personnel performed risk assessments as required by 10 CFR 60.65(a)(4) and applicable station procedures, 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 to verify plant conditions were consistent with the risk assessment. The inspectors also reviewed the TS requirements and inspected portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met.

- A torus spray out-of-service for the torus spray valve failure to close on April 4, 2012 (Order 60102294)
- A RPS voltage regulator failure resulting in a half-scram on April 6, 2012 (Notification 20553616)
- Risk assessment associated with the emergent recovery of GE 14i fuel bundle parts including potential for damage of adjacent spent fuel bundles from foreign material, potential for overexposure during recovery activities and the potential effects on the spent fuel pool on April 22, 2012 (Order 60097577-0550)
- B & D EDGs out-of-service for preventive maintenance (Orders 80102248 and 60097819)

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:

- A control room ventilation degraded temperature controller (Order 70135888)
- HPCI with unsecured restraint chain falls (Order 80106266)
- Technical evaluation of the dose impacts of loaded isotope rod cask basket and dropped rod segment during H1R17 outage (Order 80105996-0040)
- J safety relief valve with failed tailpipe temperature indicator (Notification 20559654)
- Secondary containment with FRVS controller setpoint low (Notifications 20559654 and 20563290)
- Rising trend in drywell floor drain flow (Notification 20559476)

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 TS 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 - 4 samples)

.1 Temporary Modifications

a. Inspection Scope

The inspectors reviewed the temporary configuration change package (TCCP) listed below to determine whether the temporary modification affected the safety functions of systems that are important to safety. The inspectors reviewed 10 CFR 50.59 documentation and post-modification testing results of the modification to verify that the temporary modification did not degrade the design bases, licensing bases, and performance capability of the affected systems.

TCCP 4HT12-007 - East Fuel Prep Machine Grapple Operate Zone Setpoint Change (NUCP Order 80106352)

b. Findings

No findings were identified.

2 Permanent Modifications

a. Inspection Scope

The inspectors evaluated the below listed modifications to determine whether these permanent modifications affected the safety functions of systems that are important to safety. The inspectors verified that the design bases, licensing bases, and performance capability of the affected systems were not degraded by the modifications. As applicable, the inspectors also reviewed revisions to the drawings, interviewed engineering personnel, and performed a walkdown of the completed modification to ensure the modifications were installed as designed.

Engineering change package 80102248, “B Emergency Diesel Generator Governor Control System Replacement.” The inspectors reviewed a selected sample of the modification documents associated with the replacement of B EDG governor control system with an upgraded digital model and with the installation of a new magnetic speed pickup device.

Engineering change package 80103199, “Removal of N2 Pressure and Floating Roofs from Safety and Turbine Auxiliary Cooling System (STACS) Accumulators.” The inspectors reviewed a selected sample of the modification documents associated with the removal of the STACS accumulator floating roofs, removal of STACS 2522E/F isolation valves, and removal of the nitrogen supply to the STACS system.
Design change package 80106279, Revision 1, “Installation of Alignment Tools on HPCI Main Pump, Booster Pump and Gearbox.” The inspectors reviewed a selected sample of the modification documents associated with the design change to permanently install new alignment devices, including technical evaluations of piping stress due to displacement of the HPCI skid mounted equipment in relation to the attached piping.

b. Findings

No findings were identified.

1R19 <u>Post-Maintenance Testing</u> (71111.19 - 8 samples)

a. <u>Inspection Scope</u>

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.

- HPCI after electronic governor remote replacement on March 17, 2012 (Order 60101966)
- BD411 1E 125 volt battery after bank replacement on April 22, 2012 (Order 50136300)
- B standby liquid control pump after squib valve replacement on April 30, 2012 (Order 50135996)
- HPCI after HPCI turbine overhaul on May 7, 2012 (Orders 30200513 and 30097435)
- B SW pump after pump packing replacement on May 17, 2012 (Order 60101423)
- New A RHR heat exchanger supply side vent valve (1-BC-V631) after installation from May 6 to May 31, 2012 (Order 60099665-0420)
- B EDG room recirculation fan (1B-V-412) after associated breaker replacement from June 11 - 20, 2012 (Order 60103000)
- B primary containment instrument gas compressor after preventive maintenance from June 25 - 26 2012 (Order 30203239)

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 Hope Creek's 17th refueling outage (H1R17), which was conducted April 13 through May 9, 2012. 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
- Identification and resolution of problems related to refueling outage activities

PSEG reported the use of EGM 11-003 in licensee event report (LER) 05000354/2012-003-00. This LER and PSEG's use of EGM 11-003 will be reviewed and dispositioned in a subsequent inspection report.

Additionally, the H1R17 refueling outage included activities associated with PSEG's pilot project to produce the isotope Cobalt-60 in the reactor for commercial and medical purposes. Therefore, the inspectors reviewed a selected sample of the applicable procedures and work orders and observed a sample of the associated fuel handling activities in the spent fuel pool to ensure PSEG and PSEG's contractors were performing these first time evolutions safely and in accordance with their procedures.

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-ST.BC-0004, A low pressure coolant injection time response test on April 3, 2012
- HC.IC-FT.SE-0034, Nuclear Instrumentation System, Channel A Rod Block Monitor, Single Loop Flow Operation; from June 2 - June 20, 2012
- HC.OP-ST.BC-0009, B RHR heat exchanger flow measurement test on April 11, 2012
- HC.OP-LR.AB-0001, 2, 3 & 4, Containment Isolation Valve Type C Leak Rate Test - CIVs 1ABHV-F022A, B, C, & D and 1ABHV-F028A, B, C, & D - Penetration P1A, B, C, & D: A, B, C, & D Main Steam Line; from April 18 - 24, 2012
- HC.OP-ST.KJ-0006, B EDG simulated loss of offsite power and loss of coolant accident test on April 26, 2012
- HC.OP-IS.JE-0003, B EDG fuel oil transfer pump inservice test on May 16, 2012
- NWS-T-25, NWS test procedure for Public Service Electric & Gas - Hope Creek Nuclear Station Target Rock 7567F 2 stage main steam safety relief valves; from April 23 to May 7, 2012

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstone: Radiation Safety - Public and Occupational

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

a. Inspection Scope

The inspectors reviewed selected activities and associated documentation in the areas below. The evaluation of PSEG's performance was against criteria contained in 10 CFR Part 20, applicable TSs, and applicable station procedures.

Inspection Planning

The inspectors reviewed performance indicators for the Occupational Exposure Cornerstone. The inspectors also reviewed the results of recent radiation protection program audits and assessments, as available, and any reports of operational occurrences related to occupational radiation safety since the last inspection.

Radiological Hazard Assessment

The inspectors discussed plant operations during the outage to identify any significant new radiological hazards for onsite workers or members of the public. The inspectors assessed the potential impact of the changes and monitoring, as appropriate, to detect and quantify the radiological hazards.

The inspectors toured and conducted walkdowns of radiological controlled areas (RCAs) and reviewed radiological surveys from selected plant areas (e.g., refueling floor, reactor cavity, reactor building, turbine buildings, condenser areas, drywell, and suppression pool) to verify that the thoroughness and frequency of the surveys were appropriate for the given radiological hazard. The inspectors also evaluated material conditions and potential radiological conditions. The inspectors made independent radiation measurements to verify radiological conditions.

The inspectors selected various radiological risk-significant work activities (e.g., reactor cavity work, in-vessel work activities, drywell work activities, condenser work, reactor cavity platform work, turbine work, and suppression pool work) that involved exposure to radiation to verify that appropriate pre-work surveys were performed to identify and quantify the radiological hazards and to establish adequate protective measures. The evaluation included, as applicable: identification of discrete particles; the presence of alpha emitters; the potential for airborne radioactive materials; potential changes in radiological conditions; and non-uniform exposures of the body.

The inspectors selectively reviewed and discussed air sample survey records associated with various radiological work activities to verify that samples were representative of breathing zone and collected and counted in accordance with procedures.

The inspectors reviewed ongoing radiological work activities to evaluate methods used to update workers on changes in radiological conditions.

Instructions to Workers

The inspectors toured the RCAs, including H1R17 refueling outage work areas, and reviewed labeling of containers of radioactive materials to verify labeling was consistent with requirements and was informative to workers.

The inspectors reviewed various radiation work permits (RWPs), as low as reasonably achievable (ALARA) reviews, ALARA work-in-progress reviews, and radiological surveys used to access high radiation areas (HRAs) to identify work control instructions or control barriers specified, use of stay times or permissible dose, and appropriate electronic personal dosimeter (EPD) alarm setpoints were in conformance with survey indications. The inspectors evaluated changes to EPD setpoints for specified conditions and updating of RWPs. The inspectors reviewed ongoing remote monitoring via teledosimetry.

Contamination and Radioactive Material Control

The inspectors observed locations where PSEG monitors potentially contaminated material leaving the RCA and inspected the methods used for control, survey, and release from these areas. The inspectors observed the performance of personnel surveying and the releasing of material for unrestricted use to verify that it was performed in accordance with plant procedures and the procedures were sufficient to control the spread of contamination and prevent unintended release of radioactive materials from the site. The inspectors selectively evaluated the radiation monitoring instrumentation sensitivity for the type(s) of radiation present.

The inspectors reviewed PSEG's criteria for the survey and release of potentially contaminated material. The inspectors verified that there was guidance on how to respond to an alarm that indicates the presence of radioactive material.

The inspectors reviewed PSEG's procedures and records to verify that the radiation detection instrumentation was used at its typical sensitivity level based on appropriate counting parameters including application of alarm setpoints based on the instrument's typical sensitivity. The inspectors also discussed alarm setpoints and typical detection capabilities with cognizant PSEG personnel.

Radiological Hazards Control and Work Coverage

The inspectors toured the facility and reviewed ongoing work and evaluated ambient radiological conditions (e.g., radiation levels or potential radiation levels). The inspectors verified the existing conditions were consistent with posted surveys, RWPs, and worker briefings. Areas toured by the inspectors included the drywell, reactor building, refueling floor, and turbine condenser areas.

The inspectors observed ongoing work activities and verified the adequacy of radiological controls, such as required surveys (including system breach radiation, contamination and airborne surveys, and surveys of radiation dose rate gradients), radiation protection job coverage (including audio and visual surveillance for remote job coverage), and contamination controls. The inspectors selectively evaluated PSEG's means of using EPDs in high noise areas as HRA monitoring devices (e.g., use of teledosimetry).

The inspectors verified that thermoluminescent dosimeters were placed on the individual's body consistent with the method that PSEG is employing to monitor dose from external radiation sources. The inspectors conducted direct observations of selected work to verify that the dosimeters were placed in the location of highest expected dose. The inspectors reviewed for HRAs with significant dose rate gradients the use of dosimetry to effectively monitor exposure to personnel. The inspectors evaluated implementation of external effective dose equivalent measurement (EDEX).

The inspectors selectively reviewed RWPs for work within areas with the potential for individual worker internal exposures. The inspectors evaluated airborne radioactive controls and monitoring, including potentials for significant airborne levels. The inspectors directly observed welding and grinding activities, including use of local ventilation system and respiratory protection equipment, to minimize airborne radioactive exposure. The inspectors reviewed contamination system breach survey results. The inspectors reviewed control rod drive replacement activities.

The inspectors observed ongoing work activities within flooded pools (e.g., reactor cavity) and selectively reviewed physical and programmatic controls for highly activated or contaminated materials (non-fuel) stored within storage pools. The inspectors evaluated controls to preclude inadvertent removal of these materials from the pool.

The inspectors conducted selective inspection of postings and physical controls for HRAs and very high radiation areas (VHRAs) to verify conformance with the Occupational performance indicator. The inspectors evaluated down-posting of areas from HRAs.

Risk-Significant HRA and VHRA Controls

The inspectors selectively discussed with the Radiation Protection Manager, supervisors, and technicians the controls and procedures for high-risk HRAs and VHRAs and any procedural changes since the last inspection. The inspectors discussed methods employed by PSEG to provide control of VHRA access including potential reduction in the effectiveness and level of worker protection (e.g., use of lock boxes).

The inspectors discussed, with health physics supervisors, controls for special areas that had the potential to become VHRAs during certain plant operations including controls to ensure that an individual was not able to gain unauthorized access to the VHRA.

The inspectors conducted a locked HRA key inventory and discussed locked HRA key control and issuance with health physics staff.

Radiation Worker Performance

The inspectors toured RCAs and observed radiation worker performance with respect to stated radiation protection work requirements to determine if performance reflected the level of radiological hazards present. The inspectors interviewed numerous workers conducting work activities in the RCA to determine if workers were aware of the radiological conditions in their workplace and the RWP controls/limits in place.

The inspectors selectively reviewed radiological problem reports since the last inspection to identify human performance errors and determine if there were any observable patterns. The inspectors discussed corrective actions for identified concerns with PSEG personnel.

Radiation Protection Technician Proficiency

The inspectors toured RCAs and observed the performance of radiation protection technicians with respect to radiation protection work requirements to determine if technicians were aware of the radiological conditions in their workplace and the RWP controls/limits and it their performance was consistent with their training and qualifications with respect to the radiological hazards and work activities.

The inspectors selectively reviewed outage radiological problem reports to identify those that indicate the cause of the events due to radiation protection technician error and to evaluate corrective action approach taken by PSEG to resolve the reported problems.

Problem Identification and Resolution

The inspectors determined if problems associated with radiation monitoring and exposure control were being identified by PSEG at an appropriate threshold and were properly addressed for resolution in their CAP. The inspectors discussed corrective actions for identified concerns.

b. Findings

No findings were identified.

2RS2 Occupational ALARA Planning and Controls (71124.02)

a. Inspection Scope

Inspection Planning

The inspectors reviewed pertinent information regarding plant collective exposure history, current exposure trends, and ongoing or planned activities in order to assess current performance and exposure challenges. The inspectors reviewed the plant's three-year rolling average collective exposure.

The inspectors evaluated and determined the site-specific trends in collective exposures using various methods such as plant historical data, including outage work activity dose, evaluation of ALARA data, and source term data.

The inspectors reviewed site-specific procedures associated with maintaining occupational exposures ALARA, including the processes used to estimate and track exposures from specific work activities.

Radiological Work Planning

The inspectors obtained from PSEG a list of work activities ranked by actual or estimated exposure that were planned for H1R17 refueling outage and selected work activities of the highest exposure significance. These included reactor disassembly, reactor cavity decontamination, scaffolding, in-service inspection, control rod drive work, and valve work.

The inspectors reviewed ALARA work activity plans and evaluations, exposure estimates, and exposure mitigation requirements. The inspectors determined if PSEG reasonably grouped the radiological work into work activities based on historical precedence, industry norms, and/or special circumstances.

The inspectors determined if PSEG's planning identified appropriate dose mitigation features, considered commensurate with the risk of the work activity, alternate mitigation features, and defined reasonable dose goals. As applicable, the inspectors evaluated verified use of respiratory protective devices from an ALARA perspective.

The inspectors determined if work planning considered the use of remote technologies (such as teledosimetry, remote visual monitoring, and robotics) as a means to reduce dose and the use of dose reduction insights from industry operating experience and plant-specific lessons learned. The inspectors verified the integration of ALARA requirements into work procedure and RWP documents.

The inspectors selectively compared accrued results achieved (dose rate reductions, person-rem used) with the intended dose established in PSEG's ALARA planning for these work activities including person-hour estimates. The inspectors determined the reasons for inconsistencies between intended and actual work activity doses, as necessary. During the H1R17 refueling outage, the inspectors selectively evaluated reasons for increased doses for work as compared to original estimates. As part of this review, the inspectors reviewed ongoing ALARA work-in-progress reviews.

Verification of Dose Estimates and Exposure Tracking Systems

The inspectors selected various ALARA work packages and reviewed the assumptions and bases for the collective exposure estimate for reasonable accuracy. The inspectors reviewed applicable procedures to determine the methodology for estimating exposures for specific work activities and the intended dose outcome. The inspectors also reviewed approvals by the station ALARA committee as applicable.

The inspectors verified, for the selected work activities, that PSEG established measures to track, trend, and if necessary to reduce, occupational doses for ongoing work activities including criteria to prompt additional reviews and/or controls.

During the H1R17 refueling outage, the inspectors selectively evaluated the methods used to adjust exposure estimates, replanning work due to emergent work, and changes in work scope when identified, as well as variations in expected radiation dose rates.

Source Term Reduction and Control

The inspectors used PSEG records to determine the historical trends and current status of significant tracked plant source term known to contribute to elevated facility aggregate exposure. The inspectors discussed the Chemistry Plan and long term plans for source term reduction (e.g., Cobalt reduction). The inspectors discussed contingency plans for potential changes in the source term as the result of changes in plant fuel performance issues or changes in plant primary chemistry. The inspectors discussed source term reduction efforts including system flushing and use of additional demineralization and filtration systems.

Radiation Worker and Radiation Protection Technician Performance

The inspectors observed both radiation workers' and radiation protection technicians' performance during work activities being performed in radiation areas, HRAs, and airborne radioactivity areas. The inspectors determined if workers demonstrated the ALARA philosophy in practice and whether there were any procedure compliance issues. The inspectors observed performance to determine whether the training and skill level were sufficient with respect to the radiological hazards and the work involved.

Problem Identification and Resolution

The inspectors determined if problems associated with ALARA planning and controls were being identified by PSEG at an appropriate threshold and were properly addressed for resolution in their CAP. The inspectors discussed corrective actions for identified ALARA concerns with the health physics staff.

b. Findings

No findings were identified.

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

a. Inspection Scope

Inspection Planning

The inspectors selectively reviewed the plant UFSAR to identify areas of the plant designed as potential airborne radiation areas and any associated ventilation systems or airborne monitoring instrumentation. The inspectors also reviewed the UFSAR for overview of the respiratory protection program and a description of the types of devices used.

The inspectors reviewed procedures for maintenance, inspection, and use of respiratory protection equipment including procedures for air quality maintenance. The inspectors also reviewed and directly observed the use of respiratory protection equipment during ongoing work activities.

The inspectors reviewed the reported performance indicators to identify any related to unintended dose resulting from personnel intakes of radioactive materials.

Engineering Controls

The inspectors evaluated the use of selected ventilation systems as to control airborne radioactivity. The inspectors discussed controls and procedural guidance for use of installed plant systems to verify system use, to the extent practicable, during high-risk activities. The inspectors discussed verification of plant ventilation systems during reactor cavity work.

The inspectors reviewed selected installed ventilation systems used to mitigate the potential for airborne radioactivity. The inspectors discussed use of installed systems during work activities with health physics staff.

The inspectors selected various temporary ventilation system setups (high efficiency particulate air filters) to support work in contaminated areas. The inspectors discussed the use of these systems with regard to procedural guidance and ALARA with health physics staff.

The inspectors selected various installed systems to monitor and warn of changing airborne concentrations in the plant. The inspectors evaluated the alarms and setpoints used to prompt PSEG/worker action to ensure that doses are maintained within the limits of 10 CFR Part 20 and ALARA.

The inspectors evaluated PSEG's use of decision criteria for evaluating levels of hard-to-detect airborne radionuclides.

Use of Respiratory Protection Devices

The inspectors evaluated PSEG's use of respiratory protective devices to maintain occupational doses ALARA.

The inspectors evaluated the use of certified respiratory protection devices to limit the intake of radioactive materials and evaluated that the devices were used consistent with their National Institute for Occupational Safety and Health/Mine Safety and Health Administration certification or conditions of NRC approval.

The inspectors reviewed air quality test records for use of air supplied respiratory protection devices.

Problem Identification and Resolution

The inspectors reviewed and discussed problems associated with the control and mitigation of in-plant airborne radioactivity to evaluate PSEG's identification and resolution in their CAP.

b. Findings

No findings were identified.

2RS4 Occupational Dose Assessment (71124.04)

a. Inspection Scope

Inspection Planning

The inspectors reviewed PSEG procedures associated with dosimetry operations, including issuance/use of external dosimetry (routine, multi-badging, extremity, neutron, etc.), assessment of internal dose (operation of whole body counter, assignment of dose based on derived air concentration-hours, urinalysis, etc.), and evaluation of and dose assessment for radiological incidents. The inspectors evaluated implementation of dose determination by use of EDEX. The inspectors evaluated procedure guidance for personnel monitoring.

External Dosimetry

The inspectors evaluated the use of personnel dosimeters that require processing, to verify National Voluntary Laboratory Accreditation Program (NVLAP) accreditation. The inspectors determined if PSEG uses a “correction factor” to address the response of the electronic dosimeter as compared to its NVLAP accredited dosimeter for situations when the electronic dosimeter must be used to assign dose.

Internal Dosimetry

The inspectors selectively evaluated the routine whole body counting program, including use of passive monitoring provided, for detection and measurement of intakes of radioactive materials.

The inspectors evaluated the minimum detectable activity of PSEG’s instrumentation used for passive whole body counting to determine if the minimum detectable activity was adequate to determine the potential for internally deposited radionuclides sufficient to prompt additional investigation.

Special Dosimetric Situations

The inspectors reviewed PSEG’s methodology for monitoring external dose in situations in which non-uniform fields are expected or large dose gradients could exist. The inspectors selectively reviewed use of multi-badging.

Problem Identification and Resolution

The inspectors selectively reviewed corrective action documents to verify that problems associated with occupational dose assessment were being identified by PSEG at an appropriate threshold and were properly addressed for resolution in their CAP.

b. Findings

No findings were identified.

2RS5 Radiation Monitoring Instrumentation (71124.05)

a. Inspection Scope

Inspection Planning

The inspectors reviewed the plant UFSAR to identify radiation instruments associated with monitoring area radiological conditions including airborne radioactivity, process streams, effluents, materials/articles, and workers.

Walkdowns and Observations

The inspectors selected various portable survey instruments in use for risk-significant radiological work or available for issuance and checked calibration and source check stickers for currency and to assess instrument material condition and operability.

The inspectors walked down portable area radiation monitors and continuous air monitors to determine whether they were appropriately positioned relative to the radiation source(s) or area(s) they were intended to monitor. The inspectors selectively compared monitor response (via local or remote indication) with actual area conditions for consistency. The inspectors evaluated instrumentation in-place on the refueling bridge and work platforms.

The inspectors selected personnel contamination monitors, portal monitors, and small article monitors and verified that the periodic source checks were performed in accordance with PSEG procedures.

Calibration and Testing Program

The inspectors reviewed alarm setpoint data for various personnel and equipment monitors at RCA exits to verify that the alarm setpoint values were reasonable under the circumstances to ensure that licensed material was not released from the site.

Calibration and Check Sources

The inspectors selectively reviewed PSEG's latest 10 CFR Part 61 waste stream report to determine if the calibration sources used were representative of the types and energies of radiation encountered in the plant.

Problem Identification and Resolution

The inspectors selectively reviewed corrective action documents associated with radiation monitoring instrumentation to determine if PSEG identified issues at an appropriate threshold and placed the issues in their CAP for resolution.

2RS6 Radioactive Gaseous and Liquid Effluent Treatment (71124.06)

a. Inspection Scope

The inspectors selectively reviewed aspects of PSEG's gaseous and liquid effluent control program in the below listed areas.

Inspection Planning and In-Office Inspection

The inspectors reviewed the Radiological Effluent Release Reports, issued since the last inspection, to determine if the reports were submitted as required by the Offsite Dose Calculation Manual (ODCM)/TSs. The inspectors reviewed the reports for any anomalous results, unexpected trends, or abnormal releases identified by PSEG for further inspection.

The inspectors reviewed the reports to identify radioactive effluent monitor operability issues reported by PSEG as provided in effluent release reports.

The inspectors also reviewed groundwater remediation reports.

ODCM and UFSAR Reviews

The inspectors reviewed the UFSAR descriptions of the radioactive effluent monitoring systems, treatment systems, and effluent flow paths to verify during inspection walkdowns.

b. Findings

No findings were identified.

2RS7 Radiological Environmental Monitoring Program (REMP) (71124.07)

a. Inspection Scope

Inspection Planning

The inspectors reviewed the annual radiological environmental operating reports, since the last inspection, to verify that the REMP was implemented in accordance with the TS and ODCM. The inspectors reviewed the report for changes to the ODCM with respect to environmental monitoring, commitments in terms of sampling locations, monitoring and measurement frequencies, land use census, inter-laboratory comparison program, and analysis of data.

The inspectors reviewed the ODCM and the UFSAR to identify locations of environmental monitoring stations and to review for information regarding the environmental monitoring program and meteorological monitoring instrumentation.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

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 CAP at an appropriate threshold, gave adequate attention to timely corrective actions, and identified and addressed adverse trends. In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed regular screening of items entered into the CAP and periodically attended management review committee meetings.

b. Findings

No findings were identified.

.2 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a semi-annual review of site issues, as required by 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 period from November 2011 to June 2012 to assess the notifications written as well as individual issues identified during the NRC's daily condition report review (Section 4OA2.1). The inspectors reviewed the Hope Creek station performance improvement integrated matrix (PIIM) for the 3rd cycle of 2011, conducted under procedure LS-AA-125-1006, "Performance Improvement Integrated Matrix," to verify that PSEG personnel were appropriately evaluating and trending adverse conditions in accordance with applicable procedures.

b. Findings and Observations

No findings were identified.

The inspectors noted that an adverse trend of main steam safety/relief valve as-found setpoint test failures existed. In 2011, PSEG performed a root cause evaluation (Order 70128407) of this chronic trend of setpoint drift beyond the tolerances allowed by the TS and the ASME Code. This problem recurred in 2012, when 6 of the 14 safety/relief valves that were removed during the refueling outage, failed testing due to setpoint drift. PSEG evaluated this problem in a work group evaluation (Order 70138789).

The inspectors also reviewed the results of the 2011 3rd cycle Hope Creek Station PIIM meeting and noted that PSEG identified the following fundamentals in variance: CAP oversight (Order 70111712); accountability for high standards (Order 70111714); housekeeping (Order 70125481); security (Order 70133952); leak management (Order 70134165); and, negative trend in industrial safety (Order 70136751). These efforts were identified for focused station effort to enhance future performance. Based on the overall review of the selected sample, the inspectors concluded that PSEG was appropriately identifying and entering issues into the CAP, adequately evaluating the identified issues, and acceptably identifying adverse trends before they became more safety significant problems.

.3 Annual Sample: Instrument Inaccuracies Unaccounted for in RHR Suppression Pool Cooling Procedure

a. Inspection Scope

The inspectors performed an in-depth review of PSEG's corrective actions for inadequate RHR suppression pool cooling flow rate in operation procedures documented in notifications 20525566, 20540094, and 20541537. The inspectors had identified that the procedure to establish RHR suppression pool cooling flow did not account for instrument inaccuracies.

The inspectors assessed PSEG's extent of condition review and the prioritization and timeliness of corrective actions to determine whether they were appropriately identifying, characterizing, and correcting problems associated with the RHR suppression pool cooling procedure which did not account for instrument inaccuracies. In addition, the inspectors interviewed station personnel and reviewed selected extent of condition evaluations that were completed to assess the effectiveness of PSEG's corrective actions. The inspectors reviewed relevant procedures, corrective action notifications, and engineering evaluation related documents to verify PSEG addressed any instrument inaccuracies issues in abnormal procedures.

b. Findings and Observations

No findings were identified.

The inspectors determined that PSEG's overall response to the issue was commensurate with the safety significance, was timely, and included appropriate corrective actions such as increasing the RHR suppression pool cooling flow rate in the abnormal procedures to include instrument inaccuracies (Order 80105705). Additionally, the inspectors determined that the actions taken were reasonable to resolve the issue and that PSEG had appropriately evaluated the extent of condition. Further, the inspectors determined, based upon review of a technical evaluation of RHR suppression pool cooling flow (Order 70133354), PSEG appropriately performed an instrument uncertainty and design margin analysis. The inspectors concluded these actions were adequate.

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

.1 Plant Events

a. Inspection Scope

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

- J Safety/Relief Valve Position Indicators - Special Report (Notification 20559654)
- Technical support center out of service for planned maintenance (Event # 48010)

b. Findings

No findings were identified.

.2 (Closed) Licensee Event Report (LER) 05000354/2012-001-00: Average Power Range Monitor Flow Unit Summers out of Tech Spec Tolerance

a. Inspection Scope

On March 1, 2012, the B reactor recirculation pump (RRP) tripped causing entry into single loop operations (SLO). During SLO the APRM flow unit summers were adjusted on March 4, 2012, during the performance of a weekly channel calibration procedure. After restoring the B RRP to service and power ascension, several alarms were received indicating that the D APRM flow unit was upscale. PSEG's investigation discovered that all four APRM flow units were out of tolerance. All of the APRMs were declared inoperable and TS 3.3.1, action b was entered for the number of operable channels being less than the required minimum operable channels for both trip systems. Subsequently, the channel calibration procedure HC.IC-CC.SE-0032, "Average Power Range Monitor Flow Unit Summers," was completed satisfactorily and the APRM flow units were declared operable.

This event was reported under 10 CFR 50.36(c)(1)(ii)(A) for exceeding the allowable value for a LSSS; 50.73(a)(2)(i)(B) for a condition prohibited by TS; and, 50.73(a)(2)(vii) for a common-cause inoperability of independent trains or channels. The inspectors reviewed PSEG's LER, apparent cause evaluation (ACE), and supporting documentation and interviewed several members of station staff and management regarding the event. A finding was identified and is discussed below. This LER is closed.

b. Findings

Introduction. The inspectors identified an NCV of very low safety significance (Green) of TSs 3.3.1 and 6.8.1 because PSEG's written procedure was not adequately established and implemented for completing the RPS test and calibration listed in TS table 4.3.1.1-1, function 2.b, footnote e. Specifically, procedure HC.IC-CC.SE-0032 for performing the weekly channel calibration of the flow biased APRMs that input into the simulated thermal power upscale RPS trip provided inadequate instructions for calculating total reactor recirculation drive flow while in single loop operations (SLO).

Description. On March 1, 2012, the B RRP unexpectedly tripped and caused entry into SLO. The plant remained in SLO to conduct repairs. On March 4, 2012, at 0025 hours, I&C technicians performed the weekly channel calibration procedure, HC.IC-CC.SE-0032, "Average Power Range Monitor Flow Unit Summers," to fulfill the weekly surveillance requirement listed in TS table 4.3.1.1-1, function 2.b, footnote e, for APRM flow biased simulated thermal power - upscale. The calibration procedure directs the I&C technician to obtain the value of total reactor recirculation drive flow from the plant process computer OD-3 and OD-3d reports to calculate the desired output voltage of the APRM flow unit. In SLO, the process computer is unable to accurately calculate the value of total reactor recirculation drive flow due to the unavailability of drive flow in the inactive loop. I&C technicians obtained the value for total recirculation drive flow from reactor engineering personnel. The reactor engineer calculated the recirculation drive flow using the formula provided in HC.IC-CC.SE-0032, but did not recognize that the formula could be incorrect under certain conditions during SLO. Using the flow value calculated by reactor engineering, I&C technicians calculated the desired output voltage. When the APRM Flow Units were found to be out of the desired voltage range in the surveillance procedure, I&C technicians made adjustments to bring all four flow units into the desired voltage range.

The plant was returned to dual loop operations on March 5, 2012, and operators began to raise reactor power. On March 6, 2012, with the reactor at 92 percent, the Control Room received an Overhead Alarm C6-D1 "APRM/RBM FLOW REF OFF NORMAL" and "ROD OUT MOTION BLOCK" with computer point C028. In addition, the D APRM flow unit upscale light was illuminated. The Control Room entered the appropriate abnormal procedure and performed actions to bypass the D Flow Unit. Operators discovered that all of the Flow Units were reading abnormally high for the current plant condition. To validate the condition, operators performed the TS daily channel check for upscale APRM flow biased simulated thermal power and concluded three of six RPS APRM had failed the channel check in a non-conservative direction. Initially, operators entered TS 3.3.1 (action a) for the number of operable channels being less than the required minimum operable channels for one trip system.

The I&C Department promptly started calibration procedure HC.IC-CC.SE-0032 and discovered that all four APRM flow units were out of tolerance. Operators then entered TS 3.3.1 (action b) for the number of operable channels being less than the required minimum operable channels for both trip systems. Subsequently, I&C technicians completed HC.IC-CC.SE-0032 to restore all APRM flow units within calibration tolerances. All required testing was completed satisfactorily, the APRM flow units were declared operable, and the LCO and abnormal procedure were exited at 1524 hours on March 6, 2012. The inspectors noted that the APRM flow units had been non-conservatively calibrated for approximately 63 hours.

The inspectors reviewed the event and questioned why PSEG was not performing calculations to determine if the LSSS allowable value for the APRM Flow Biased Simulated Thermal Power-Upscale, as listed in TS Table 2.2.1–1, “Reactor Protection System Instrumentation Setpoints,” had been exceeded. The inspectors also reminded PSEG that exceeding a LSSS allowable value such that the automatic safety system does not function as required would be reportable in accordance with 10 CFR 50.36(c)(1)(ii)(A). Subsequently, PSEG determined that the LSSS allowable value had been exceeded for approximately 12 hours.

The inspectors reviewed PSEG’s apparent cause evaluation (ACE) for this event. The ACE determined that the cause of the APRM flow units’ setpoints being non-conservative was that the procedure did not provide I&C technicians and reactor engineers with the proper method for determining drive flow values for the surveillance. The formula provided in the procedure (HC.IC-CC.SE-0032) that calculated the drive flow was incorrect because the formula was for dual loop operation, not for SLO. Corrective actions included revising the appropriate procedures and developing a schedule template (including required surveillances) for entry into and return from SLO.

The inspectors determined that PSEG’s inadequate procedural guidance and instructions for calculating the percent recirculation drive flow values in the TS channel calibration procedure for the APRM flow unit summers, during SLO, was a performance deficiency. In addition, although the condition where the APRM Flow Unit setpoints was self-revealed by overhead alarms in the main control room, in accordance with IMC 0612, the inspectors considered this finding NRC-identified because the inspectors’ review of this issue added significant value. Specifically, the inspectors’ questions led PSEG to perform the calculations necessary to determine that LSSS allowable values had been exceeded; therefore, this issue was reportable in accordance with 10 CFR 50.36(c)(1)(ii)(A).

Analysis. The inspectors determined that PSEG’s inadequate procedural guidance and instructions for calculating the percent recirculation recirculation drive flow values in the TS channel calibration procedure for the APRM flow unit summers during SLO, was a performance deficiency. Specifically, the formula provided in the procedure for calculating drive flow was for dual loop operation, not for SLO. The performance deficiency was more than minor because it was associated with the procedure quality attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, incorrect calibration of the APRM flow units resulted in the APRM flow biased setpoint being non-conservative and exceeding the associated TS LSSS allowable value for a period of time that was considered a condition prohibited by TS. The inspectors performed a Phase I screening of the finding using IMC 0609, Attachment 0609.04, Table 4a, Mitigating Systems cornerstone and determined the issue was of very low safety significance (Green) because the finding was not a design or qualification deficiency, did not result in an actual loss of safety function, and was not potentially risk significant for external events.

The finding had a cross-cutting aspect in the area of human performance, resources component, because PSEG did not ensures that a TS-required RPS calibration procedure was complete, accurate and adequate to assure nuclear safety. Specifically, the formula provided in the APRM flow unit summer procedure that calculated the drive flow was incorrect. The formula provided in the procedure was for dual loop operation, not for SLO. H.2(c)

Enforcement. Technical Specification (TS) 3.3.1, "Reactor Protection System Instrumentation," requires that as a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be operable with the applicability shown in Table 3.3.1-1. Table 3.3.1-1, Function 2.b, "Average Power Range Monitor, Flow Biased Simulated Thermal Power – Upscale," is applicable in Mode 1 and requires a minimum of two (2) operable channels per trip system. The associated Limiting Condition for Operation Action Statement 3.7.1.2(a) states that with the number of operable channels less than required by the minimum operable channels per trip system requirement for both trip systems, place at least one trip system** in the tripped condition within one hour and take the action required by Table 3.3.1-1. Action 4 from Table 3.3.1-1 states that the mode switch be placed in at least startup within 6 hours.

TS 6.8.1 requires, that written procedures shall be established, implemented, and maintained covering the activities in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Section 8.b(2)(l) of Appendix A of Regulatory Guide 1.33 states, in part, that specific implementing procedures are required to be for each surveillance test or calibration listed in the technical specifications, including those for Reactor Protection System Tests and Calibrations.

Contrary to the above, between 0025 hours on March 4, 2012, and 1524 hours on March 6, 2012, PSEG's written procedure (HC.IC-CC.SE-0032) was not adequately established and implemented for completing a reactor protection system test and calibration listed in the technical specification table 4.3.1.1-1, function 2.b, footnote e. Specifically, the procedure for performing the weekly channel calibration of the APRMs that input into the flow biased simulated thermal power upscale reactor protection system trip, HC.IC-CC.SE-0032, provided inadequate instructions for calculating reactor recirculation drive flow while in single loop operations. This rendered both trip systems for the Average Power Range Monitor, Flow Biased Simulated Thermal Power – Upscale inoperable for greater than six (6) hours and the mode switch was not taken to STARTUP. This issue was entered into CAP as notification 20549760, and PSEG's corrective actions included revision of the appropriate procedures and development of a schedule template (including required surveillances) for entry into and return from SLO. Because the violation was of very low safety significance (Green) and has been entered into the CAP this violation is being treated as an NCV, consistent with Section 2.3.2.a of the NRC Enforcement Policy. **(NCV 05000354/2012-003-02, Average Power Range Monitor Flow Unit Summers out of Tech Spec Tolerance)**

4OA5 Other Activities (OA)

Inspection Procedure 71003, License Renewal Condition Numbers 26 and 27 Regarding the Drywell Air Gap Drainage Capability and Monitoring

a. Inspection Scope

An inspection, performed during April 23 - 25, 2012, evaluated activities related to the renewed license condition numbers 2.C(26) and 2.C(27) applicable to the drywell air gap drainage capability and monitoring at the Hope Creek Generating Station. During the 2012 refueling outage, the plant staff was in the process of establishing drainage from the drywell air gap region and conducting the work items specified in license condition number 2.C(26).

License condition number 2.C(26) requires that until drainage is established from the drywell air gap region, boroscope examinations be made in the bottom of the drywell air gap, ultrasonic thickness measurements be performed of the drywell shell at specific locations, the penetration sleeve J13 be monitored for water leakage when the drywell reactor cavity is flooded up, the torus room be monitored for leakage from other penetrations, and a report be submitted to NRC within 90 days after each refueling outage summarizing the results.

License condition number 2.C(27) provides the requirements to be met after drainage is established from the drywell air gap region.

The inspectors walked down portions of the outside of the drywell and the torus to confirm the acceptance of a sample of visual examinations was in accordance with site procedures and ASME Code IWE requirements. External portions of the containment boundary were also observed at the location of the J13 penetration and the 4" diameter drain lines from the air gap between the drywell steel shell and concrete to the torus room floor. The inspectors went into the tunnels in the concrete surrounding the drywell at the 135 and 270 degree locations to view the drywell outer surface and concrete conditions in the lower portion of the air gap. The performance of boroscope examinations by qualified visual examiners through two penetration sleeves of the lower air gap to drywell locations was observed. Additionally, the inspectors reviewed video records of the air gap region and the results of ultrasonic thickness measurements that were made within the scope of license condition number 2.C(26).

During an inspection, performed July 26, 2012, followup was done to review the concrete to drywell surface condition under the drywell. This was to determine if the concrete under the drywell was sufficiently dry to prevent drywell underside lower head corrosion by moisture. The inspector met with licensee engineering staff to review their work on this topic.

The applicable drawing is C-0935-0, Rev 11, Drywell Construction Sequence which along with specifications 10855-C-102, Rev 17 (Concrete Mix Designs), C-101(Q), Rev 13 (Concrete), C-152 (Q) Rev 10 &12 (Inorganic Zinc coating) and the ACI 506R-05, Standard on Shotcrete provide a basis for understanding the condition of the drywell underside. The engineering staff presented the conditions that the concrete placement sequence including Shotcrete application to the underside of the drywell and drying times with openings for venting did not allow for an excess of water buildup under the drywell. The zinc coating on the drywell underside provides an additional means of protection to the drywell material.

b. <u>Findings</u>

No findings were identified.

4OA6 Meetings, including Exit

On July 19, 2012, the inspectors presented inspection results to with Mr. D. Lewis and other members of his staff. The inspectors asked PSEG whether any materials examined during the inspection were proprietary. No proprietary information was identified.

4OA7 Licensee-Identified Violations

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

- 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," states, in part, that activities affecting quality shall be prescribed by documented instructions or procedures of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions or procedures. PSEG procedure OP-HC-108-115-1001, "Operability Assessment and Equipment Control Program," step 5.3.4.2.D states that if an inoperable SSC will impact secondary containment integrity during fuel handling and core alterations, then develop a contingency plan for sealing secondary containment penetrations for each inoperable penetration. Contrary to OP-HC-108-115-1001, a contingency plan for sealing an inoperable secondary containment penetration was not established on April 18 and 19, 2012, during core alterations, while a rupture disk (H1EA-1EAPSE-2210B) in the B station SW piping (secondary containment penetration) was removed for replacement. The inspectors evaluated the finding using IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process," and Appendix G, Attachment 1, "Shutdown Operations Significance Determination Process Phase 1 Operational Checklists for Both PWRs and BWRs." Specifically, Checklist 7, BWR Refueling Operation with RCS Level >23', was reviewed and the finding was determined to be of very low safety significance (Green) because it challenged the containment control guidelines, but did not meet the criteria that would require phase 2 or phase 3 analyses. PSEG documented the issue in Notification 20555753.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

PSEG Personnel

J. Perry, Hope Creek Site Vice President
D. Lewis, Hope Creek Plant Manager
E. Carr, Operations Director
K. Knaide, Work Management Director
W. Kopchick, Engineering Director
F. Mooney, Maintenance Director
P. Duca, Senior Engineer, Regulatory Assurance
M. Gaffney, Regulatory Assurance Manager
P. Bonnett, Senior Compliance Engineer
H. Trimble, Radiation Protection Manager
D. Boyle, Operations Support Manager
J. Krall, Reactor Engineering Manager
B. Brammeier, ISI Program Engineer
A. Enilo, Mechanical Design Engineer
G. Holoman, Project Manager (Drywell)
E. Maloney, Principal Nuclear Engineer (ISI)
R. Schmidt, Principal Nuclear Engineer (IVVI)
F. Leeser, Chemistry Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened/Closed

05000354/2012003-01	NCV	Preconditioning of the Reactor Building to Torus Vacuum Relief Valves (Section 1R12)
05000354/2012003-02	NCV	Average Power Range Monitor Flow Unit Summers Out of Tech Spec Tolerance (Section 4OA3.2)

Closed

05000354/2012-001-00	LER	Average Power Range Monitor Flow Unit Summers Out of Tech Spec Tolerance (Section 4OA3.2)

LIST OF DOCUMENTS REVIEWED

In addition to the documents identified in the body of this report, the inspectors reviewed the following documents and records:

Hope Creek Generating Station UFSAR
Hope Creek Generating Station TS
Technical Specification Action Statement Log
Hope Creek Generating Station Nuclear Control Operator (NCO) Narrative Logs

Section 1R01: Adverse Weather Protection

Procedures
WC-AA-107, Seasonal Readiness, Revision 11
HC.OP-AB.BOP-0004, Grid Disturbances, Revision 20
HC.OP-GP.ZZ-0003, Section 5.2, Securing the Plant from Winter Operations, Revision 26
HC.OP-IO.ZZ-0006, Power Changes During Operation, Revision 54
OP-AA-102-101, Unit Load Changes, Revision 6
OP-AA-108-107-1001, Electric System Emergency Operations and Electric Systems Operator Interface, Revision 3
OP-AA-108-111-1001, Severe Weather and Natural Disaster Guidelines, Revision 7

Other Documents
PSEG Transmission and Distribution Department Letter, from Thomas J. Fries regarding, "Hope Creek Switchyard Readiness for 2012 Summer Period," dated 5/16/2012
System Vulnerability Review Report, Hope Creek Transformers and Switchyard, dated December 2011
2012 Hope Creek Summer Readiness Affirmation Certification Letter
2012 Hope Creek Summer Readiness Plant System Readiness Review Summary
2011 Summer Readiness Hope Creek Critique

Orders
30209086, Secure from Winter Operations
60101981 Remove 'C' Circ Water Pump for Impeller Inspection
70139407, Evaluate 'A' Reactor Recirc M/G Set Lube Oil Cooler Performance
70139055, Summer Season Readiness Gaps
80105687, 2012 Hope Creek Summer Readiness

Notifications
20546593, Summer Readiness Contingency Order
20546739, Recirc Pump Trip Summer Readiness
20559718, Summer Seasonal Readiness Gaps
20564332, Open Summer Readiness Action Items
20563089, Review of PB Grid Disturbance Guidance

Section 1R04: Equipment Alignment

Procedures
HC.OP-SO.KJ-0001, Emergency Diesel Generator, Revision 63
HC.OP-SO.BC-0001, Residual Heat Removal System, Revision 51

HC.OP-IO.ZZ-0005, Cold Shutdown to Refueling, Revision 34
HC.OP-SO.BC-0002, Decay Heat Removal Operation, Revision 27
HC.OP-SO.EA-0001, Service Water System Operation, Revision 35
HC.OP-SO.BD-0001, Reactor Core Isolation Cooling System Operation, Revision 40

Drawings
M-30-1, Diesel Engine Auxiliary Systems Starting Air and Lube Oil, Revision 19
M-51-1, Residual Heat Removal, Revision 41
M-10-1, Service Water, Revision 54
M-49-1, Reactor Core Isolation Cooling, Revision 29

Orders
60101423, H1EA-1B-P-502, Repack Pump/RPLC Packing

Other Documents
HC RF17 Protected Equipment Log

Section 1R05: Fire Protection Measures

Procedures
FRH-II-413, HPCI Pump & Turbine Room, Revision 3
FRH-II-433, A SACS Heat Exchanger & Pump Room, Revision 4
FRH-II-432, B SACS Heat Exchanger & Pump Room, Revision 3
FRH-II-423, RHR Heat Exchanger Room, RACS Pumps & Heat Exchanger, Revision 4
FRH-II-442, Inert Gases Compressor Rooms, FRVS Re-Circulating Unit Area, Steam Vent and Equipment Area, Revision 4
MA-AA-716-010-1000, Maintenance Planning, Revision 4
FP-AA-011, Control of Transient Combustible Material, Revision 2

Notifications (*NRC-identified)
20552093*, NRC identified questions
20552970*, TP&L and TCPS
20552784*, NRC Questions on Transient Comb Permits
20554363, Material Staged w/o Combustible Permit

Other Documents
HTC-12 RB1-003, Transient Combustible Permit Reactor Bldg RM 4111
HTC-12 RB2-003, Transient Combustible Permit Reactor Bldg RM 4408
HTC-12 RB2-004, Transient Combustible Permit Reactor Bldg RM 4410

Section 1R08: Inservice Inspection (ISI)

NDT Examination Procedures
OU-AA-335-018, VT-1 and VT-3 Visual Examination of ASME Class MC and CC Surfaces, Revision 5
OU-AA-335-003, Magnetic Particle Examination, Revision 2
OU-AA-335-002, Liquid Penetrant Examination, Revision 2
OU-AA-335-014, VT-1 Visual Examination, Revision 2
OU-AA-335-016, VT-3 Visual Examination of Component Supports and Integral Attachments, Revision 2
OU-AA-335-005, Radiographic Examination, Revision 1

GEH-PDI-UT-1, PDI Generic Procedure for the Ultrasonic Examination of Ferritic Pipe Welds, Revision 8
GEH-PDI-UT-2, PDI Generic Procedure for Ultrasonic Examination of Austenitic Pipe Welds, Revision 6
EPRI-WOL-PA-1, Procedure for Manual Phased Array Ultrasonic Examination of Weld Overlaid Similar and Dissimilar Metal Welds, Revision 2
GEH-VT-204, Procedure for In-Vessel Visual Inspection of BWR4 RPV Internals, Version 14

NDT Examination Reports
MT-12-004, Magnetic Particle Test Results of Reactor Recirculation Welds
MT-12-003, Magnetic Particle Test Results HPCI Turbine Steam System Welds
PT-12-006, Liquid Penetrant Test Results of Nuclear Boiler & Recirc System
UT-12-005, Ultrasonic Test Results of Core Spray Pipe to Elbow Welds

Orders
60088924, Penetrant Exam of Vent Valve Leak Main Steam System (AB)
50137098, Ultrasonic Exam of Core Spray System (BE)
60101260, Piping Replacement (four field welds) RCIC System

Other Documents
Section XI Repair or Replacement
WPS NDWP-13, Manual Gas Tungsten (GTAW) and Shielded Metal (SMAW) Welding of Carbon Steel Group 1, Revision 11
WPS-NDWP-26, GTAW and SMAW Welding of Carbon Steel Group 2, Revision 0
PQ-4, Weld Procedure Qualification Supporting WDS-NDWP-13 (0.154 coupon)
PQ-10, Weld Procedure Qualification Supporting WDS-NDWP-13 (0.432 coupon)
PQ-114, Weld Procedure Qualification Supporting WDS-NDWP-26 Group 1
PQ-115, Weld Procedure Qualification Supporting WDS-NDWP-26 Group 2

Section 1R11: Licensed Operator Requalification Program

Procedures
HC.OP-IO.ZZ-0006, Power Changes During Operation, Revision 52

Notifications
20544829, Dual Rod Select - Entered AB.IC-0001

Other Documents
Main Control Room Operator Narrative Logs for Day Shift, dated 1/28/2012

Section 1R12: Maintenance Effectiveness

Procedures
HC.MD-ST.GS-0002, Reactor Building to Torus Vacuum Relief Valve 18 Month Testing, Revision 5 and Revision 8
PP-AA-3001, Position Paper on Preconditioning, Revision 0
HC.MD-ST.AB-0003, Safety Relief Valve Discharge Piping Vacuum Breaker in-place Setpoint Test, Revision 1
HC.OP-IS.GS-0101, Containment Atmosphere Control System Valves - Inservice Test, Revision 45

ER-AB-331-1006, BWR Reactor Coolant System Leakage Monitoring and Action Plan, Revision 0
HC.OP-AB.CONT-0006, Drywell Leakage, Revision 6
HC.OP-GP.ZZ-0005, Drywell Leakage Source Detection, Revision 9

Notifications (*NRC identified)
20554080*, NRC Inspector Preconditioning Question
20554343*, Preconditioning during Surveillance Test
20370021*, Potential Preconditioning BJHV-F004
20554159*, Possible Preconditioning of Equipment
20562300*, NRC Challenge of Extent of Condition Review
20373812, Develop HCGS Position on Preconditioning
20408678, Potential Preconditioning of MSIVs-RF15
20553979, Valve Setpoint Exceeded Tech Spec Value
20555930, Vacuum Breaker 1ABPSV-F037H Failed Setpoint
20553982, LLRT Above IST Limit
20563652, Drywell Floor Drain Flow Slowly Rising - 0.11 GPM
20564089, Industry Best Practices - DW Leak Procedure
20565829, New Procedure Request – MOV Backseating

Orders
70137157, Unacceptable Preconditioning of Reactor Building to Torus Relief Vacuum Breaker during 18 Month Surveillance Testing
70086624, Regulatory Analysis Paper Preconditioning of Structures, Systems, and Components (SSCs) Before Determining Operability
70085313, LCO 4.0.3 Evaluation for TS 3.0.5
50126411, ST 18M 1GSPSV-5030 RB-Torus Vacuum Breaker
50136807, ST 18M 1GSPSV-5032 RB-Torus Vacuum Breaker

Drawings
M-57-1, HCGS Containment Atmosphere Control, Revision 40

Other Documents
DITS 3.40, Design, Installation and Test Specification for Containment Atmosphere Control System for the HCGS, Revision 6
Adverse Condition Monitoring and Contingency Plan HC 12-010, Drywell Leakage, dated 6/19/2012

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Notifications (*NRC identified)
20553675*, NRC Question (PRA Risk Eval Form)
20553203*, NRC Resident Question Valve Operability
20561768*, WW-221 PRA Eval Form Typo
20553009, BC-HV-F027A Failed to Stroke Close
20553616, Received ‘A’ RPS Half Scram, Enter AB’s
20553684, NUMAC Failed to Reset After Loss of PWR
20555078, Spare Governor with Bad Shaft

Orders
60102294, BC-HV-F027A Failed to Stroke Close

60102342, Received ‘A’ RPS Half Scram, Enter AB’s
70136871, Received ‘A’ RPS Half Scram, Enter AB’s
80102248, B EDG Governor Replacement
60097819, Replace A3-11 Module in 1D-C-428

Drawings
M-51-1, HCGS Residual Heat Removal, Revision 39

Other Documents
HCGS PRA Risk Evaluation Form for Work Week 1214, 4/1/2012 - 4/8/2012, Revision 0
HC RF17 Protected Equipment Log

Section 1R15: Operability Evaluations

Procedures
CC-AA-320-011, Transient Loads, Revision 0
HC.OP-SO.GU-0001, Filtration, Recirculation and Ventilation System Operation, Revision 25
HC.OP-ST.GU-0002, Reactor Building Integrity Functional Test, Revision 15

Notifications (*NRC identified)
20552093*, NRC Identified Questions
20553200*, NRC Question on EDG Hoist Restraint
20562860*, FRVS DP Setpoint Low
20563290*, Rx Bldg to Atmosphere Alarm Setpoint
20552095, Operability Evaluation Required for A CR Vent TC
20550290, Operability Evaluation for ‘A’ 403 Heater Control Loop
20550995, Control Room Temp Control Problem
20555810, Thimble for Cask Basket Fell with Co-60
20550958, NRC Resident Questions Co-60 Activities and Cask
20559765, M SRV Tailpipe >200F at NOP/NOT
20560379, B SRV Tailpipe Temp >213 Deg F
20559654, J SRV Tailpipe Temp Indicator Failed Hi
20561616, G SRV Tailpipe Temp >200 Degs NOP/NOT
20562069, SRV Temp Recorder Media Trouble
20551812, Nuisance low D/P alarm with FRVS I/S
20110484, 1BV206 Building Pressure Controller Setpoint Low
20563939, OTDM for J SRV Temperature Indication
20557619, R SRV Wires Rolled

Orders
30097435, 10-S-211/Install Rigging/RMV Equipment
60101988, Tune H1GK-1GKTIC-9589A1
70135888, Operability Evaluation Required for A CR Vent TC
80106266, Impact of Unsecured Chain Falls on HPCI System Piping

Calculations
SC-GU-0065-1, Reactor Building/Atmosphere Diff. Pressure Control, Revision 4
GU-0030, Reactor Building DP Controller Setpoint, Revision 1
GU-0013, Filtration, Recirculation, and Ventilation System Exhaust Rate, Revision 4

Other Documents
Op Eval 12-004, A Control Room Ventilation Temperature Control
OTDM 12-008, J SRV Tailpipe Indication, dated 6/7/2012
ACM 12-009, J SRV Tailpipe Temperature Monitoring, dated 6/7/2012
Reactivity Maneuver Plan # 2012-0036, EOC17 Shutdown

Section 1R18: Plant Modifications

Notifications
20555941, Temporary Configuration Change for Fuel Prep Machine
20557799, Water Dripping from 132ft to 145ft Rx Bl
20557797, C SACS Pump LOOP B I/S Ind. Defective
20556783, B LOP/LOCA Fan Will Not Run
20555078, Spare Governor with Bad Shaft
20557004, Contingency to Remove 10-P-204 Piping
20557005, Contingency to Remove 10-P-217 Piping
20483121, HPCI Gearbox Requires Replacement
20554611, Speed Reducer As Found Alignment OOT
20558182, HPCI Potential Unanalyzed Condition
20558353, HPCI Evaluation Did Not Bound Alignment
20559255, Large Swings in SACS Head Tank Levels

Orders
80106352, TCCP 4HT12-007, East Prep Machine Grapple
80102066, GE 14i Reload 17 Inspection
60097577-0530, Adjust Fuel Prep Machine - TCCP 4HT12-007
80103199, Removal of N2 Pressure and Floating Roofs from STACS Accumulators
80102248, B Emergency Diesel Generator Governor Control System Replacement, Revision 1
80106344, HPCI Gearbox Requires Replacement
80106344, Operation 420, TE HPCI Pump Alignment Pipe Stress, Confirmation 9854522
80106344, Operation 620, TE HPCI Main Pump As-Found Alignment, Confirmation 9875140
80106344, Operation 660, TE NRC Questions on HPCI Alignment, Confirmation 9875774
60093018, Operation 135/0001, Engineering Walkdown, Confirmation 9864924

Other Documents
50.59 Screen HC-11-102, DCP 80103199/STACS Accumulator Modification, Revision 0
50.59 Screen HC-11-036, DCP 80102248, Revision 0
DCP 80106279, Installation of Alignment Tools on HPCI Main Pump, Booster Pump and Gearbox, Revisions 0 & 1
50.59 Screen HC-12-040, Order 80106279, Installation of Alignment Tools on HPCI Main Pump, Booster Pump and Gearbox, Revision 1
Design Analysis No. 6H4-4052, Alignment Tools for HPCI Pump/Gear Box, Revision 0
Document No. 80106344, Operation: 0420 (Records Management DEH120134), Pipe Stress Evaluation for HPCI Boost Pump Re-Alignment
Document No. 80106344, Operation: 0590 (Records Management DEH120149), Pipe Stress Evaluation for As-left HPCI Main Pump Re-Alignment
Document No. 80106344, Operation: 0620 (Records Management DEH120151), Pipe Stress Evaluation for As-found HPCI Main Pump Alignment
Document No. 80106344, Operation: 0660 (Records Management DEH120153), Response to NRC Questions Regarding HPCI Alignment During Plant Construction
ASME Section III, Subsection NB, Article 3672.8-1986, Cold Springing

Email from Sean Kababik (PSEG) to Peter Koppel (PSEG), HPCI As Found, dated 5/2/2012
Email from Sean Kababik (PSEG) to Peter Koppel (PSEG), et al, HPCI Final Alignment Numbers, dated 5/2/2012, with attachment Doc1.doc
Email from Scott Connelly (PSEG) to Fred Bower (USNRC), et al, HPCI Train Alignment, dated 5/3/2012

Section 1R19: Post-Maintenance Testing

Procedures
MA-AA-716-012, Post Maintenance Testing, Revision 18
HC.MD-CM.BJ-0001, HPCI Main Pump Overhaul, Revision 8
HC.MD-CM.BJ-0003, HPCI Gear Box Overhaul, Revision 6
HC.MD-CM.BJ-0002, HPCI Booster Pump Overhaul, Revision 12
HC.MD-CM.FD-0001, HPCI Turbine Overhaul, Revision 17
HC.MD-PM.KL-0002, Containment Instrument Gas Compressor Preventive Maintenance, Revision 14

Completed Surveillances
HC.OP-ST.BH-0002, SLC Flow Test - 18 Months, Revision 28
HC.OP-IS.BH-0003, Standby Liquid Control Pump - Inservice Test
HC.OP-ST.BJ-0002, HPCI System Functional Test, dated 3/17/2012
HC.IC-LC.BJ-0002, HPCI Turbine Controller Tuneup, dated 3/17/2012
HC.IC-LC.FD-0001, HPCI Turbine Speed Control Test, dated 3/17/2012
HC.MD-ST.PK-0002, 125 Volt Quarterly Battery Surveillance, dated 4/22/2012
HC.MD-ST.PK-0007, 125 Volt Station Batteries 18 Month Service Test using BCT-2000 with Windows Software and Associated Surveillance Testing, dated 4/23/2012
HC.MD-GP.ZZ-0015, Battery Equalizing Charge, dated 4/23/2012
HC.OP-ST.BJ-0002, HPCI System Functional Test (Low Pressure) and HPCI System Response Time Test (High Pressure), dated 5/8/2012
HC.OP-IS.BJ-0001, HPCI Main and Booster Pump Set - Inservice Test, dated 5/9/2012
HC.OP-IS.EA-0002, B Service Water Pump - Inservice Test, dated 5/17/2012
HC.OP-FT.KL-0001, Primary Containment Instrument Gas System Compressor Capacity Test, dated 6/21/2012

Notifications (*NRC identified)
20557614*, HC.OP-ST.BH-0002 Revision Request
20552450*, NRC Question - HPCI Flow Gain Adjustment
20552448*, NRC Resident Question HPCI Time Response
20558155*, NRC Questions Regarding 1B-D-411 Battery
20551121, Chart Recorder Noise Obscures Speed Data
20551062, HPCI Governor Woodward Failure Analysis
20550811, HPCI EGR Oil Tubing Discrepancy
20551124, HC.OP-ST.BJ-0002, HPCI 18M Time Response
20551122, HPCI Procedure Revision
20556400, HPCI Stop Vlv Act Oring Deformed/Melted
20554999, Turbine Shaft TIR Numbers OOS
20555193, Indication in HPCI Turbine Rev Chamber
20555133, Discolored Grease HPCI Main Pump Coupling
20554891, Missing Lock Plates Inside HPCI Turbine
20554806, Need Engineering Eval on Rigging Pin
20554921, Need Alternate Gasket for Bypass Bodies

20554611, Speed Reducer As Found Alignment OOT
20554689, 1FDHV-F071 Failed LLRT HC.OP-LR.FD-0003
20558731, Low Flow Alarm on EC483 1B-V-412
20563722, H1GM-1B-V-412 Low Flow
20562518, H1GM-1B-V-412 Start Spiking Transformer Amps
20565085, Water Leak From PCIG Head Manifold Gasket

Orders
50135996, ST 18M HC.OP-ST.BH-0002 SLC Flow Surv
60101966, HPCI Gov Vlv FD-HV-4879 Open w/ 0% Demand
80106130, HPCI Availability Assessment Following Erratic Governor Valve Response during Auxiliary Oil Pump Start
80106171, HPCI Speed Recorder Erratic Indication
50136300, ST 18M/1B-D-411 Service Test
30208818, 15Y 1B-D-411 Battery Replacement
30097435, 12 Y PM HPCI Turbine Internal Inspection
30200513, HPCI Turbine Inspection PM
60101423, H1EA-1B-P-502: Repack Pump/RPLC Packing
30203239, H1KL-1B-K-202: B PCIG Compressor Clean and Inspect
30215862, H1KL-1B-K-202: B PCIG Compressor Reed Valve Replacement

Section 1R20: Refueling and Other Outage Activities

Procedures
HC.OP-GP.ZZ-0002, Primary Containment Closeout, Rev. 14
OP-AA-108-114, Post Transient Review, Revision 4
ER-AA-600-1043, Shutdown Risk Management, Revision 5
HC.OP-IO.ZZ-0001, Refueling to Cold Shutdown, Revision 27
HC.OP-IO.ZZ-0003, Startup from Cold Shutdown to Rated Power, Revision 100
HC.OP-IO.ZZ-0004, Shutdown from Rated Power to Cold Shutdown, Revision 94
HC.OP-IO.ZZ-0005, Cold Shutdown to Refueling, Revision 34
OP-HC-108-102, "Management of Operations with the Potential to Drain the Reactor Vessel," Revision 0
OP-HC-108-102, "Management of Operations with the Potential to Drain the Reactor Vessel," record of completed procedure dated April 23, 2012 (Order 80105570)

Notifications (*NRC identified)
20556701*, Drywell Air Gap Question from NRC Inspector
20556720*, LTA Notification Screening
20555745*, Lack of Full Thread Engagement
20555570*, Lack of Full Thread Engagement
20556483*, R17 Seismic Restraint – TIP/HTV Bottles
20558155*, NRC Questions Regarding 1B-D-411 Battery
20558784*, NRC Identified Issues in Reactor Building
20558521*, NRC Identified Issues in the Drywell
20558717*, HC.OP-AB.CONT-0003 Revision Request
20557614*, HC.OP-ST.BH-0002 Revision Request
20557272*, R17 1B-G-400 Missing Bolt on C/Guard
20553205*, NRC Questioned Scaffold Red Tag
20553213*, Airlock-4322A Rx Bld Truck Bay Malf
20554160*, NRC Identified Unsecured Scaffold Cart

20554161*, NRC Identified Questionable FME for SRVs
20559547, OPDRV LAR Submittal Tracking
20555753, B SSWS Rupture Disk Replacement
20554889, Questionable Use of Tagging Exception
20554930, Insufficient Tagging to Perform Task
20554964, Level 4 Clearance Event
20555203, CR Narrative Missing Information
20554631, Hope Creek Reactor Scram for RF17
20555810, Thimble for Cask Basket Fell with Co-60 Segment in It
20556785, Perform Evaluation on Venting of SDC Suction
20556883, Penetration Sleeve J-37 Leaks
20555913, Penetration Sleeve J-19 Leaks
20556922, FME on Fuel Assembly in Core Location 19-60
20556175, Paint Chip on SFP Fuel Assembly AC-02
20556761, Technical Evaluation Order Tracking in RF17
20550210, Potential TEEW for RE Product Approvals
20555386, HC.MD-CM.AB-0006, SRV Maintenance Procedure Revision Request
20555988, Residual Water Found in AE Line
205557175, A SACS SRV Lift & Head Tank Drained
20549401, Extent of Condition Inspection – A Recirc Pump Motor
20557107, Inverters Lost During 10A401 Bus Testing
20557211, Momentary Loss of FPCC – A Vital Bus PB-05
20557200, Aborted Battery Discharge Test As Directed
20555753, B SSWS Rupture Disk Replacement
20557107, Inverters Lost During 10A401 Bus Testing
20557211, Momentary Loss of FPCC-A Vital Bus PB-05
20557200, Aborted Battery Discharge Test as Directed
20558917, K SRV Tailpipe > 200F at NOP/NOT
20558916, E SRV Tailpipe > 200F at NOP/NOT
20558918, R SRV Tailpipe > 200F at NOP/NOT
20558919, J SRV Tailpipe > 200F at NOP/NOT
20559112, R17 – SRV As-Found Test Results
20556305, SOVs For SRVs A, F, M Fail Leakage Testing
20559532, E and K SRV Tailpipe Temperature High
20559255, Large Swings in SACS Head Tank Levels
20559523, Drywell Floor Drain Sump Pump Malfunction
20559105, Voltage Regulator Placed in Manual
20555263, Residual Water Coming From SRV Pilot
20556629, Unexpected Start of 00K-107 Due to Work
20556622, Repeat “As Found” LLRT Failure in 1R16/1R17
20556488, FME Reconciliation in RPV Not Performed
20556487, Debris Found in Control Cell 46-39
20554896, Drywell Leak During Cavity Flood Up
20555265, Insulation Cover Missing on D Main Steam
20555266, A MSIV Limit Switch Conduit Disconnected
20555267, D MSIV Limit Switch Conduit Disconnected
20558125, LPRM 48-33 Found Leaking at 1005 psig
20557982, LPRM 40-49 Found Leaking at 1005 psig
20558779, HPCI Vibration Probes Not Functional
20557676, Reactor Engineer ST Not Coded to LCO
20557533, Repair #9 Turbine BPV in RF18

20558963, R17 HPCI Control Room Indications Flow Fluctuations
20559077, HPCI Turbine Vibration Point T2
20559094, Replace HPCI Magnetic Speed Pickup Connector
20559260, High Vibration on HP Turbine at Low Load
20559267, HPCI Inboard Bearing Axial Vibration > Alert Value
20559430, Entered AB.IC-001 for Stuck Rod 34-35
20559434, Unable to Establish Elevated CRD D/P with PCV
20559524, Reactor Engineer Evaluate Data For HC.OP-FT.BB-0001
20559528, Turbine Bypass Valve #5 Indicates Not Closed
20556291, R17 Jet Pump 16 Wedge Movement
20556380, R17, RPV Shroud Support Weld Indication
20558521*, NRC Identified Issues in Drywell
20555705, Fabricate Steel Catch Tray for SFP
Notification 20538142, Create Order for OPDRV Procedure Revisions
Notification 20536886, Integrate Rigging Requirements for OPDRVs
Notification 20529254, NRC Enforcement Guidance (ML11251A230)
Notification 20559547, OPDRV LAR Submittal Tracking

Orders
80106344-0150, Bolt Thread Engagement at Torus Isolation Valve 1GSV-028 and 1GSHV-4964
80106344-0160, Bolt Thread Engagement at Torus Isolation Valve 1GSV-201 and 1GHSV-11541
Order 70138857, OPDRV LAR Submittal Tracking

Calculations
AB-54, Pressure Drop Across SRV Accumulator, Revision 0
H-1-SN-MDC-0327, EPU SRV Actuation under Station Blackout Conditions, Revision 0

Other Documents
Hope Creek Narrative Log for dayshift on April 22, 2012
Hope Creek Narrative Log for dayshift on April 23, 2012
Technical Evaluation 80105570-0010 (DEH120014), Reactor Vessel Drain Down Time During Control Rod Drive Maintenance Window of Refueling Outage of April 2012
Technical Evaluation 80105570-0020 (DEH120017), Reactor Vessel Drain Down Time During LPRM Maintenance Window of Refueling Outage of April 2012

Section 1R22: Surveillance Testing

Procedures
HC.OP-IO.ZZ-0006, Power Changes During Operation, Revision 53
HU-AA-104-101, Procedure Use and Adherence
HC.OP-DL.ZZ-00026, Surveillance Log, Attachment 3v, Single Loop Operation (SLO) T/S 3.4.1.1 Action a, Revision 131
NF.HC-701-1003, Reactor Engineering Guidance for Single Loop Operation, Revision 3
HC.OP-SO.SF-0002, Rod Block Monitor Operation, Revision 6

Completed Surveillances
HC.OP-ST.BC-0004, LPCI Subsystem A ECCS Time Response Functional Test, dated 4/3/2012
HC.IC-FT.SE-0028, Nuclear Instrumentation System, Divisions 1 - Channel A Average Power Range Monitor, Single Loop Flow Operation, dated 3/1/2012

HC.IC-FT.SE-0029, Nuclear Instrumentation System, Divisions 2 - Channel B Average Power Range Monitor, Single Loop Flow Operation, dated 3/2/2012
HC.IC-FT.SE-0030, Nuclear Instrumentation System, Divisions 3 - Channel C Average Power Range Monitor, Single Loop Flow Operation, dated 3/2/2012
HC.IC-FT.SE-0031, Nuclear Instrumentation System, Divisions 4 - Channel D Average Power Range Monitor, Single Loop Flow Operation, dated 3/2/2012
HC.IC-FT.SE-0032, Nuclear Instrumentation System, Divisions 1 & 3 - Channel E Average Power Range Monitor, Single Loop Flow Operation, dated 3/2/2012
HC.IC-FT.SE-0033, Nuclear Instrumentation System, Divisions 2 & 4 - Channel F Average Power Range Monitor, Single Loop Flow Operation, dated 3/1/2012
HC.IC-FT.SE-0034, Nuclear Instrumentation System, Channel A Rod Block Monitor, Single Loop Flow Operation, dated 3/2/2012
HC.IC-FT.SE-0035, Nuclear Instrumentation System, Channel B Rod Block Monitor, Single Loop Flow Operation, dated 3/2/2012
HC.OP-DL.ZZ-0026, Surveillance Log; Attachment 1, Surveillance Log; Attachment 1a, Surveillance Log - Control Room; Attachment 3v, Single Loop Operation (SLO) T/S 3.4.1.1 Action a; and, Attachment 5, T/S Surveillance and Planned Evolution AOT Tracking Log; dated 3/1/2012
HC.OP-ST.BC-0009, B Residual Heat Removal Heat Exchanger Flow Measurement Test, dated 4/11/2012
HC.OP-LR.AB-0001, Containment Isolation Valve Type C Leak Rate Test - CIVs 1ABHV-F022A and 1ABHV-F028A - Penetration P1A: A Main Steam Line, dated 4/18/2012
HC.OP-LR.AB-0002, Containment Isolation Valve Type C Leak Rate Test - CIVs 1ABHV-F022B and 1ABHV-F028B - Penetration P1B: B Main Steam Line, dated 4/18/2012
HC.OP-LR.AB-0003, Containment Isolation Valve Type C Leak Rate Test - CIVs 1ABHV-F022C and 1ABHV-F028C - Penetration P1C: C Main Steam Line, dated 4/18/2012
HC.OP-LR.AB-0004, Containment Isolation Valve Type C Leak Rate Test - CIVs 1ABHV-F022D and 1ABHV-F028D - Penetration P1D: D Main Steam Line, dated 4/18/2012
HC.OP-ST.KJ-0006, Integrated Emergency Diesel Generator 1BG400 Test - 18 Months, dated 4/26/2012
HC.OP-IS.JE-0003, C Diesel Fuel Oil Transfer Pump Inservice Test, dated 5/16/2012
HC.IC-FT.SE-0034, Nuclear Instrumentation System, Channel A Rod Block Monitor, Single Loop Flow Operation, dated 3/6/2012
HC.IC-CC.SE-0019, Nuclear Instrumentation System - Non-divisional Channel A Rod Block Monitor, dated 6/1/2012 (Order 50144925)
HC.IC-CC.SE-0020, Nuclear Instrumentation System - Non-divisional Channel B Rod Block Monitor, dated 6/13/2012 (Order 50146287)

Notifications (*NRC identified)
20551955*, Single Loop Surveillance Test
20551964*, HU-AA-104-101 Violations in Procedure Steps
20552754, APRM Procedure Not Revised
20549232, HC.IC-FT.SE-0028 Revision Request
20549254, OTSC for HC.IC-FT.SE-0028
20545239, Clarification on Intent of HU-AA-104-101
20554120, F048B Leak-By Exceeds 250 GPM - RF17
20479941, Traveling Water Screen Fan Did Not Start
20556783, B LOP/LOCA Fan Will Not Run
20559112, RF17 - SRV As-Found Test Results
20551616, RF17 - SRV As-Found Test Results
20553891, HU-AA-104-101 Step 4.7 is Confusing

20552754, APRM Procedure Not Revised
20554207, NRC Question on RBM

Orders
50122117, B Residual Heat Removal Heat Exchanger Flow Measurement Test
50135563, LPCI Subsystem A ECCS Time Response Functional Test
70136792, APRM Procedure Not Revised
60101778, Reactor Recirc Pump Single Loop Operations
80102563, Traveling Screen Motor Room Fan 0BV-558
80106344, Performance of H1RF17 'B' LOP/LOCA Test with Unavailable Loads
80102248, B EDG Governor Replacement
50136366, 18M ST OP-ST.KJ-0006 B EDG Integrated ST
50147397, 3MO ST: OP-IS-JE-0003 B EDG FO Trans PMP
70137297, NRC Question on RBM

Calculations
SC-SE-0002-2, Average Power Range Monitor (APRM) Channels A - F & Rod Block Monitors (RBM) Channels A & B, Revision 9

Safety Valve Test Data Travelers
#12-141, A SRV, dated 5/8/2012
#12-149, B SRV, dated 5/7/2012
#12-143, C SRV, dated 4/23/2012
#12-152, D SRV, dated 5/9/2012
#12-148, E SRV, dated 4/23/2012
#12-139, F SRV, dated 5/10/2012
#12-145, G SRV, dated 5/9/2012
#12-150, H SRV, dated 5/9/2012
#12-140, J SRV, dated 5/11/2012
#12-146, K SRV, dated 5/11/2012
#12-142, L SRV, dated 5/10/2012
#12-147, M SRV, dated 5/10/2012
#12-144, P SRV, dated 5/10/2012
#12-151, R SRV, dated 5/9/2012

Other Documents
PSEG Vendor Technical Document Number 324450, NWS Test Procedure For Public Service Electric & Gas - Hope Creek Nuclear Station Target Rock 7567F 2 Stage Main Steam Safety Relief Valves, Revision 5, dated 2/9/2009

Section 2RS1: Radiological Hazard Assessment and Exposure Controls

Procedures
RP-AA-376, Radiological Posting, Labeling and Marking, Revision 6
RP-AA-403, Administration of the Radiation Work Permit Program, Revision 3
RP-AA-460, Control for High and Very High Radiation Areas, Revision 15
RP-AA-463, High Radiation Area Key Control, Revision 3
RP-AA-503, Unconditional Release Survey Method, Revision 7

Other Documents
RWP and ALARA Plans - 1/1000, 1/4225, 7, 4799, 1, 4703

Hope Creek BRAC Point/Source Term Table
Corrective Action Notifications – 20554898, 20554895, 20555918, 20550771, 20550764, 20550508, 20550315, 20556714, 20556677

Section 2RS2: Occupational ALARA Planning and Controls

Procedures
RP-AA-400, ALARA Program, Revision 6
RP-AA-401, Operational ALARA Planning and Control, Revision 11
RP-AA-1001, Establishing Collective Radiation Exposure Estimates and Goals, Revision 2
RP-AA-403, Administration of the Radiation Work Permit Program, Revision 3

Other Documents
RWP and ALARA Plans - 1/1000, 1/4225, 7/4799, 1/4703, 1/4223
Work-In-Progress Reviews – RWP – 1/6033, 1/6043, 1/6011, 1/4231, 1/4225, 1/4220, 6/4209
Corrective Action Documents (various)

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

Procedures
NC.RP-TI.ZZ-0403, Operation of Breathing Air System, Revision 3
NC.RP-TI.ZZ-0404, Testing and Evaluation of Compressed Air, Revision 1
RP-AA-825-1011, Inspection and Use of the Mururoa V4 MTH2 Air Supplied Suit, Revision 2
RP-AA-300-1002, Electron Capture Isotope Control, Revision 1

Other Documents
TC-19C-293 Instruction Manual
Compressed Air Breathing Analysis, 4/20/12
Breathing Air Radioactivity Test 4/17/12
Corrective Action Documents (various)

Section 2RS4: Occupational Dose Assessment

Procedures
NC.RP-TI.ZZ-0206, Dose Assessment for Airborne Radioactive Material Exposure, Revision 4
RP-AA-504, Routine Operation of the radiation Protection Gross Counting Facility, Revision 0
RP-AA-300, Radiological Survey Program, Revision 4

Other Documents
General Source Term Data
Radiological Survey data – various for risk important radiological work activities
Corrective Action Documents (various)

Section 2RS5: Radiation Monitoring Instrumentation

Other Documents
General Source Term Data
Calibration and source check records – instruments used for risk important radiological work activities
Corrective Action Documents (various)

Section 2RS6: Radioactive Gaseous and Liquid Effluent Treatment

Other Documents
Annual Effluent Release and Environmental Reports 2011
Offsite Dose Calculation Manual (Rev. 26) and changes
Reports (various) - Routine Groundwater
General Source Term Data

Section 2RS7: Radiological Environmental Monitoring Program

Other Documents
Annual Effluent Release and Environmental Reports 2011
Offsite Dose Calculation Manual (Rev. 26) and changes

Section 4OA2: Problem Identification and Resolution

Procedures
HC.OP-ST.BC-0009, Residual Heat Removal System RHR Heat Exchanger Flow Measurement - 18 Month, Revisions 13 and 14
HC.OP-IS.BC-0001, A Residual Heat Removal Pump In-Service Test, Revision 42
HC.OP-IS.BC-0003, B Residual Heat Removal Pump In-Service Test, Revision 43
HC.OP-AB.ZZ-0001, Transient Plant Conditions, Revisions 22, 23, 24, and 25

Notifications (*NRC identified)
20540094*, NRC Question on RHR Supp. Pool Clg Flow
20542366*, Assess Response Time in Resolving Issue
20541537*, Determine Optimum RHR Flow for SPC
20525566*, NRC Resident Question
20525050, Procedure Difference for SACS Pump Trip
20552558, HC PIIM Management Issue Industrial Safety
20545231, Leak Management Program Gaps to Excellence
20515895, Housekeeping
20468753, Ownership and Accountability for High Standards
20468751, Management Oversight and Monitoring of Corrective Actions
20544135, Security Chronic Yellow Rating by NOS
20564833, Create Action Plan for Mentoring – HC Ops PIIM

Orders
70132623, NRC Question on RHR Suppression Pool Cooling Flow
70133354, RHR Suppression Pool Cooling Optimum Flow-Rates Instrument Uncertainty & Design Margin
80105705, RHR Torus Cooling/Spray Flow
70128659, NRC Resident Question on RHR Suppression Pool Cooling
80091864, RHR Hydraulic Analysis
70136751, HC PIIM Management Issue Industrial Safety
70134165, Leak Management Program Gaps to Excellence
70125481, Housekeeping
70111714, Ownership and Accountability for High Standards
70111712, Management Oversight and Monitoring of Corrective Actions
70133952, Security Chronic Yellow Rating by NOS

Calculations
EG-0020, STACS Required Flows and Heat Loads - EPU, Revision 10
SC-BC-0071-1, RHR Loop Tolerance Calculation, Revision 7

Other Documents
NFS-0252, HCGS Nuclear Fuel Related Safety Analysis Information Report, Revision 0

Section 4OA3: Followup of Events and Notices of Enforcement Discretion

Procedures
OP-HC-108-102, Management of Operations with the Potential to Drain the Reactor Vessel, Revision 0
NF-HC-701-1003, Reactor Engineering Guidance for Single Loop Operation, Revision 3
HC.OP-SO.SB-0001, Reactor Protection System Operation, Revision 32

Notifications
20558919, J SRV Tailpipe Temperature Greater than 200°F
20559654, J SRV Tailpipe Temperature Indicator Failed High
20561933, J SRV Tailpipe Temperature Indicator Failed High
20563219, 8 Hour Report to NRC - RAL 11.7.1.C for TSC
20538142, Create Order for OPDRV Procedure Revisions
20536886, Integrate Rigging Requirements for OPDRVs
20529254, NRC Enforcement Guidance (ML11251A230)
20559547, OPDRV LAR Submittal Tracking
20549556, Recirc Flow Unit-D Upscale
20549639, D APRM Upscale Alarms
20549760, HC.IC-CC.SE-0032 Not Completed
20564678, Record Omission

Orders
70138857, OPDRV LAR Submittal Tracking

Other Documents
Letter (LR-N12-0160) from David P. Lewis (PSEG) to Document Control Desk (USNRC), regarding Special Report - Safety/Relief Valve Position Indicators, dated 5/31/2012 (ML12171A570)
Hope Creek Narrative Log for Dayshift, dated 4/22/2012
Hope Creek Narrative Log for Dayshift, dated 4/23/2012
OP-HC-108-102, Management of Operations with the Potential to Drain the Reactor Vessel, record of completed procedure, dated 4/23/2012 (Order 80105570)
Prompt Investigation, 20549760, HC.IC-CC.SE-0032 Not Completed, event date 3/6/2012
Apparent Cause Evaluation, 20549760/70135578, Non-conservative APRM Flow Unit Setpoints, event date 3/4/2012
HC.IC-CC.SE-0032, Nuclear Instrumentation System APRM Flow Unit Summers, Revision 20, record copy dated 2/26/2012 (Order 50147749)
HC.IC-CC.SE-0032, Nuclear Instrumentation System APRM Flow Unit Summers, Revision 20, record copy dated 3/4/2012 (Order 50147967)
HC.IC-CC.SE-0032, Nuclear Instrumentation System APRM Flow Unit Summers, Revision 20, record copy dated 3/6/2012 (Order 50148086)
HC.OP-ST.BB-0001, Recirculation Jet Pump Operability - Daily, Revision 49, record copy dated 3/5/2012

Section 4OA5: Other Activities

Drawings
P-6172-1, Bechtel Drywell Area
C-0791-1, Bechtel RB Drywell Shield Wall Sections and Details, Revision 17
C-0935-0, Bechtel RB Drywell Vessel Supports, Revision 11
C-0935-0, Drywell Construction Sequence, Rev 11

Notifications
20523591 20528217 20536777 20538337 20538467 20570312

Specifications
10855-C-102, Concrete Mix Designs, Rev 17
C-101(Q), Concrete, Rev13
C-152 (Q), Inorganic Zinc coating, rev 10 &12

Other Documents
Drywell Ultrasonic Thickness Measurements, for Order 50144280, Operation 0090, dated 4/17/2012 and 4/18/2012
Amendment No. 189, Renewed License No. NPF-57, Items 26 and 27, dated 7/20/2011
Order 50144280, Operation 0050, ST, Drywell Air Gap Boroscope Visual Examinations
Surveillance Log, Reactor Building, Drywell Areas, page 52 of 101, dated 4/22/2012
Standard ACI 506R-05, Shotcrete

Section 4OA7: Licensee-Identified Violations

Procedures
OP-HC-108-115-1001, Operability Assessment and Equipment Control Program, Revisions 18, 19, and 20

Notifications
20555753, B SSWS Rupture Disk Replacement

Orders
70137821, B SSWS Rupture Disk Replacement

Other Documents
Prompt Investigation (20555753), Inadequate Control of Secondary Containment Penetration, event date 4/19/2012
Work Group Evaluation (70137821), H1EA-1EAPSE-2210B Not Coded to the Secondary Containment LCO, event date 4/20/2012

LIST OF ACRONYMS

AC	Alternating Current
ACE	Apparent Cause Evaluation
ADAMS	Agency-wide Documents Access and Management System
ALARA	As Low As Reasonably Achievable
APRM	Average Power Range Monitor
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CRAC	Control Room Air Condition
CS	Core Spray
EDEX	Effective Dose Equivalent for External Exposure
EDG	Emergency Diesel Generator
EGM	Enforcement Guidance Memorandum
EPD	Electronic Personal Dosimeter
FRVS	Filtration Recirculation and Ventilation System
HPCI	High Pressure Coolant Injection
HRA	High Radiation Area
IMC	Inspection Manual Chapter
ISI	Inservice Inspection
IST	Inservice Test
LAR	License Amendment Request
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LPRM	Local Power Range Monitor
LSSS	Limiting Safety System Setpoint
MT	Magnetic Particle Test
NCV	Non-Cited Violation
NDE	Non-Destructive Examination
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
PIIM	Performance Improvement Integrated Matrix
PPC	Plant Process Computer
PSEG	Public Service Enterprise Group Nuclear LLC
PT	Penetrant Test
RCA	Radiological Controlled Area
RCIC	Reactor Core Isolation Cooling
REMP	Radiological Environmental Monitoring Program
RHR	Residual Heat Removal
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RRP	Reactor Recirculation Pump
RT	Radiographic Test
RTP	Rated Thermal Power
RWP	Radiation Work Permit
SACS	Safety Auxiliary Cooling System

SDP	Significance Determination Process
SLO	Single Loop Operation
SR	Surveillance Requirement
SSC	Structures, Systems, and Components
STACS	Safety and Turbine Auxiliary Cooling System
SW	Service Water
TCCP	Temporary Configuration Change Package
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
UT	Ultrasonic Test
VHRA	Very High Radiation Area
VT	Visual Test
WD	Drive Flow