January 19, 2007

Mr. William Levis
Senior Vice President and Chief Nuclear Officer
PSEG LLC - N09
P. O. Box 236
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION - NRC COMPONENT DESIGN BASES INSPECTION REPORT 05000354/2006015

Dear Mr. Levis:

On December 7, 2006, 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 December 7, 2006, with Mr. George Barnes 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. In conducting the inspection, the team examined the adequacy of selected components and operator actions to mitigate postulated transients, initiating events, and design basis accidents. The inspection also reviewed PSEG's response to selected operating experience issues. The inspection involved field walkdowns, examination of selected procedures, calculations and records, and interviews with station personnel.

This report documents two NRC-identified findings that were of very low safety significance (Green). One of these findings was determined to involve a violation of NRC requirements. However, because of the very low safety significance of the violation and because it was entered into your corrective action program, the NRC is treating the violation as a non-cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy. If you contest the 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, D.C. 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspectors at Hope Creek.

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 component of NRC's document 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/

Lawrence T. Doerflein, Chief
Engineering Branch 2
Division of Reactor Safety

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

Enclosure: Inspection Report 05000354/2006015

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No. 50-354

License No. NPF-57

Report No. 05000354/2006015

Licensee: Public Service Enterprise Group Nuclear LLC

Facility: Hope Creek Generating Station

Location: P.O. Box 236
Hancocks Bridge, NJ 08038

Dates: October 30 to December 7, 2006

Inspectors: S. Pindale, Senior Reactor Inspector (Team Leader)
D. Orr, Senior Reactor Inspector
K. Young, Senior Reactor Inspector
M. Snell, Reactor Inspector
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L. Casey, Nuclear Safety Professional Development Program (NSPDP) Participant (Observer)
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Approved by: Lawrence T. Doerflein, Chief
Engineering Branch 2
Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000354/2006015; 10/30/2006 - 12/07/2006; Hope Creek Generating Station; Component Design Bases Inspection.

This inspection covers the Component Design Bases Inspection, conducted by a team of four NRC inspectors and two NRC contractors. Two findings of very low safety significance (Green) were identified, one of which involved a violation of regulatory requirements and is considered to be a non-cited violation. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609, "Significance Determination Process" (SDP). 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 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events and Mitigating Systems

- Green. The team identified a non-cited violation of 10CFR50, Appendix B, Criterion III, "Design Control," for a failure to correctly translate the design basis into procedures in that measures were not established to verify the adequacy of the service water (SW) cooling water design flow. Specifically, the SW abnormal operating procedure allowed for continued SW pump operation (up to 12 hours) without declaring the pump inoperable, when SW strainer differential pressure exceeded the SW system hydraulic calculation assumptions. Operation in this condition did not ensure the design basis minimum flowrate would be provided to the safety auxiliaries cooling system, which in turn cools the emergency diesel generators and other safety-related equipment, under the most limiting conditions. PSEG performed a review of past operability, and initiated a notification to change the SW abnormal operating procedure to declare the SW pump inoperable when a SW strainer differential pressure exceeds the appropriate setpoint.

  The finding is more than minor because it is associated with the design control attribute of the mitigating systems cornerstone and affected the cornerstone's objective to ensure the availability and reliability of systems that respond to initiating events to prevent undesirable consequences. The finding also is associated with the initiating events cornerstone because unavailability of one service water pump increases the likelihood of loss of service water events. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the team conducted a Phase 1 SDP screening and determined a more detailed Phase 2 SDP evaluation was required to assess the safety significance because the finding affected two cornerstones. The finding was determined to be of very low safety significance (Green) based upon the Phase 2 SDP evaluation. The finding has a cross-cutting aspect in the area of human performance because the SW abnormal operating procedure was inappropriately revised in May 2006 to remove acceptance criteria for pump operability with elevated strainer differential pressure. (Section 1R21.2.1.2)

Cornerstone: Mitigating Systems

- Green. The team identified a finding for the failure to correctly translate the design basis of the containment ventilation backup nitrogen bottles into procedures. Specifically, the operator rounds log was revised to allow the two backup nitrogen bottles, which operate containment vent valve 1GSHV-4964, to decrease to 200 psig each. The nitrogen capacity calculation assumed a minimum of 800 psig per bottle to ensure sufficient nitrogen to stroke the containment vent valve as needed in beyond design basis events. Operation below 800 psig did not ensure the containment vent valve could be used according to emergency operating procedures to protect containment against overpressurization. PSEG raised the minimum backup nitrogen bottle pressure to 800 psig per bottle, performed a review of past bottle pressures, and initiated a notification to change the operator rounds log to allow a minimum of 800 psig per bottle.

  This finding is more than minor because it is associated with the design control attribute of the mitigating systems cornerstone, and affected the cornerstone's objective to ensure the availability and reliability of the containment vent to respond to initiating events to prevent undesirable consequences. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors conducted a Phase 1 SDP screening and determined the finding was of very low safety significance (Green) since it did not result in a loss of safety system function because the vent valve local hand-jack was available to open the containment vent. There were no violations of NRC requirements because the containment vent function is not covered by Technical Specifications, is not a part of Hope Creek's licensing basis, and is only credited in beyond design basis events. (Section 1R21.2.1.4)

B. Licensee-identified Violations.

None.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R21 Component Design Bases Inspection (IP 71111.21)

.1 Inspection Sample Selection Process

The team selected risk significant components and operator actions for review using information contained in the Hope Creek Probabilistic Risk Assessment (PRA) and the Nuclear Regulatory Commission's (NRC) Standardized Plant Analysis Risk (SPAR) model. Additionally, the Hope Creek Significance Determination Process (SDP) Phase 2 Notebook, Revision 2, was referenced in the selection of potential components and operator actions for review. In general, the selection process focused on components and operator actions that had a risk achievement worth (RAW) factor greater than 2.0 or a Risk Reduction Worth (RRW) factor greater than 1.005. The components selected were located within both safety related and non-safety related systems, and included a variety of components such as pumps, valves, diesel generators, transformers, and electrical buses.

The team initially compiled an extensive list of components based on the risk factors previously mentioned. The team performed a margin assessment to narrow the focus of the inspection to 20 components and five operator actions. The team's evaluation of possible low design margin considered original design issues, margin reductions due to modifications, or margin reductions identified as a result of material condition/equipment reliability issues. The margin assessment evaluated the impact of licensing basis changes that could reduce safety analysis margins, such as the extended power uprate (EPU), which was being reviewed by the NRC at the time of this inspection. The assessment also included items such as failed performance test results, corrective action history, repeated maintenance, maintenance rule (a)(1) status, operability reviews for degraded conditions, NRC resident inspector input of equipment problems, plant personnel input of equipment issues, system health reports and industry operating experience. The margin review of operator actions included complexity of the action, time to complete action, and extent of training on the action.

Consideration was also given to the uniqueness and complexity of the design and the available defense-in-depth margins. This inspection effort included walk-downs of selected components, a review of selected simulator scenarios, interviews with operators, system engineers and design engineers, and reviews of associated design documents and calculations to assess the adequacy of the components to meet both design bases and risk informed beyond design basis requirements. A summary of the reviews performed for each component, operator action, operating experience sample, and the specific inspection findings identified are discussed in the following sections of the report. Documents reviewed for this inspection are listed in the attachment.

.2 Results of Detailed Reviews

.2.1 Detailed Component Design Reviews (20 Samples)

.2.1.1 'B' Emergency Diesel Generator (mechanical)

a. Inspection Scope

The team reviewed three mechanical sub-components of the 'B' emergency diesel generator (EDG) to assess whether the EDG would function as required during accident conditions. These included the fuel oil, lube oil, and starting air systems. The fuel oil system was reviewed to ensure a sufficient supply of fuel oil would be available and the system would function as designed; the lube oil system was reviewed to ensure the system would provide a continuous supply of oil to components of the EDG; and the starting air system was inspected to ensure the system would supply sufficient compressed air to initiate an engine start. The team reviewed calculations, hydraulic analyses, system health reports, associated instrumentation and setpoints, and notifications to verify operation of the diesel was within its design basis. Surveillance test results were reviewed to verify fuel oil levels and coolant temperatures were within acceptable limits; and operating and test procedures were reviewed to assess whether component operation and alignments were consistent with design and licensing basis assumptions. Finally, the team discussed the design, operation and maintenance of the EDG and related equipment with design and system engineers and plant operators, and completed a walkdown of the EDG components.

b. Findings

No findings of significance were identified.

.2.1.2 'B' Service Water Pump

a. Inspection Scope

The team selected the 'B' service water (SW) pump as a representative sample of the Hope Creek service water pumps. The team reviewed design documents, including drawings, calculations, procedures, tests and modifications to evaluate the functional requirements of the 'B' SW pump. The team reviewed these documents to ensure the pump was capable of meeting design basis requirements, with consideration of allowable pump degradation and net positive suction head. The team also reviewed the design and operation of the SW pump to ensure EPU assumptions did not reduce or invalidate safety analysis margins. To assess the current condition of the pump, the team interviewed the system engineer, and reviewed system health and related condition reports. The team performed walkdowns of the service water pump house areas, which included the intake traveling screens. Surveillance test results were reviewed to determine whether pump performance margin was sufficient to assure design basis assumptions could be achieved.

b. Findings

Introduction: The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10 CFR 50, Appendix B, Criterion III, “Design Control,” in that measures had not been established to verify the adequacy of the SW design flow with elevated strainer differential pressure. Specifically, the SW abnormal operating procedure allowed for continued SW pump operation (up to 12 hours) without declaring the pump inoperable, when SW strainer differential pressure (d/p) exceeded the SW system hydraulic calculation assumptions. Operation in this condition did not ensure the design basis minimum flowrate would be provided to the safety auxiliaries cooling system (SACS), which cools the emergency diesel generators (EDG) and other safety-related equipment, under the most limiting conditions.

Description: The team reviewed SW abnormal operating procedure HC.OP-AB.COOL-0001, “Station Service Water,” and SW hydraulic analysis EA-0003, “Station Service Water System Hydraulic Analysis,” to ensure the SW pumps were capable of meeting their design basis requirements. The hydraulic analysis developed the minimum allowable SW pump flow under the most limiting conditions, including pump degradation.

The team noted that the SW abnormal operating procedure allowed SW pump operation with elevated strainer d/p, which was beyond the hydraulic analysis assumptions for system operation. The procedure stated that when the high-high d/p alarm was received at 138.55" (of water) across the SW strainers, the system could be operated in this condition for up to 12 hours before removing the strainer from service and initiating a notification to clean/inspect the strainer internals. The hydraulic calculation assumed a maximum of 138.55" across the strainer to support SW pump operation under the most limiting conditions. Consequently, the abnormal operating procedure permitted SW pump operation in an unanalyzed condition, where sufficient SW flow may not be available to meet design basis requirements.

The team found that the SW abnormal operating procedure was revised in May 2006 to allow continued operation at greater than 138.55" water for up to 12 hours. Previously, the procedure stated that the SW strainer should be declared inoperable for sustained strainer d/p greater than 138.55" water. The team noted the associated procedure change evaluation did not ensure the design basis minimum SW flow rate would be provided to SACS under all required scenarios. The team further noted that the procedure version in effect prior to May 2006 similarly contained weaknesses regarding operator actions for declaring SW pumps inoperable. As a result, on several occasions, operators did not declare SW pumps inoperable when the associated strainer exceeded 138.55" water. For example, on April 14, 2006, excessive river water grassing caused the in-service ‘C’ and ‘D’ SW strainers to experience off-scale high differential pressures (greater than 138.55" water) such that there was reduced SW pump flow. While operators properly declared the ‘D’ SW pump inoperable for 7 hours, they did not declare the ‘C’ SW pump inoperable.

PSEG subsequently performed a review of past operability, and concluded that no SW pump operability determinations were missed that would have caused a SW pump to exceed its technical specification allowed outage time. PSEG entered the issue into its corrective action program as notification 20305949 and intended to appropriately revise the abnormal operating procedure.

Analysis: The team determined this issue was a performance deficiency because PSEG failed to ensure that adequate design control measures existed to maintain sufficient SW flow under all conditions. Specifically, the abnormal operating procedure was changed in May 2006 such that adequate SW flowrate under the most limiting conditions was not ensured. Reduction in SW flowrates, due to elevated strainer d/p, could challenge SACS and EDG operation depending on loading and cooling water temperature conditions.

This issue was more than minor because it was associated with the design control attribute of the mitigating systems cornerstone and affected the cornerstone's objective to ensure the availability and reliability of systems that respond to initiating events to prevent undesirable consequences. This issue also impacted the initiating events cornerstone because unavailability of one service water pump increases the likelihood of loss-of-service water events. Traditional enforcement does not apply because the issue did not have any actual safety consequence or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the team conducted a Phase 1 SDP screening and determined a more detailed Phase 2 SDP evaluation was required to assess the safety significance because the finding affected two cornerstones. The team determined that the finding was of very low safety significance (Green) based upon the Phase 2 loss-of-service water worksheet SDP evaluation. Since the maximum allowed time of SW strainer operation at the higher d/p was 12 hours, an exposure time of less than three days was selected. The performance deficiency directly affected the probability of a loss-of-service water event, and therefore, the likelihood of a loss-of-service water event was increased by one order of magnitude. All of the mitigating equipment listed on the Phase 2 worksheet for a loss-of-service water event were unaffected by the finding and operator recovery actions were credited. The most predominant core damage sequence was an inadvertent/stuck-open relief valve with a failure of high pressure coolant injection and a failure of the operators to depressurize.

The finding has a cross-cutting aspect in the area of human performance because PSEG did not adequately ensure procedures were complete, accurate, and up-to-date to assure nuclear safety. Specifically, the SW abnormal operating procedure was inappropriately revised in May 2006 to remove acceptance criteria for pump operability with elevated strainer d/p, without proper assessment by design engineering of the strainer d/p design criteria.

Enforcement: 10 CFR 50 Appendix B, Criterion III, “Design Control,” requires, in part, that measures shall be established to assure that the design basis for structures, systems, and components are correctly translated into procedures. Contrary to the above, measures had not been established to ensure that the design basis minimum cooling water flowrate would be provided to SACS, under the most limiting conditions, with elevated SW strainer differential pressures. Because this violation is of very low safety significance and has been entered into PSEG’s corrective action program (Notification 20305949), this violation is being treated as a non-cited violation consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 05000354/2006015-01, Inadequate Strainer Differential Pressure Design Control to Ensure Adequate Service Water Flow)**

.2.1.3 Safety Relief Valve PSV-F013A

a. Inspection Scope

The 'A' safety relief valve (SRV) was inspected as a representative sample to verify its ability to meet its design basis requirements in response to transient and accident events, including automatic reactor depressurization. Accumulator volume calculations were evaluated to ensure that sufficient air would be provided to operate the valve under design basis specifications. The team verified that instrument setpoints were properly translated into system procedures and tests, and reviewed completed tests intended to demonstrate component operability. The team reviewed drawings, component calculations and system calculations to verify calculation inputs and assumptions were accurate and justified. The team reviewed the maintenance and functional history of the ‘A’ SRV by sampling corrective action reports, the system health report, and operating and surveillance test procedures.

b. Findings

No findings of significance were identified.

.2.1.4 Containment Vent Valve 1GSHV-4964

a. Inspection Scope

The team inspected valve 1GSHV-4964 to verify the capability of the valve to perform as required during beyond design bases accident conditions. The valve has an active safety function in the closed position to isolate the primary containment, and has a risk significant function in the open position to support primary containment pressure control as directed by the station emergency operating procedures (EOP). Backup nitrogen bottle volume calculations were evaluated to ensure that sufficient nitrogen would be available to operate the valve. The team verified that instrument setpoints were properly translated into system procedures and tests, and reviewed completed tests. The team reviewed drawings, component calculations, system calculations and design specifications to verify design inputs and assumptions were accurate and justified. The team also reviewed the design and operation of the containment vent valve to ensure

EPU assumptions did not reduce or invalidate safety analysis margins. The team reviewed the maintenance and functional history of the valve by sampling corrective action reports, work orders, system health reports, operator round sheets, and inservice test results. The team interviewed operators to understand when and how containment venting would be used, and the overall reliability of the valve.

b. Findings

Introduction: The team identified a finding of very low safety significance (Green) for a failure to translate the design of the containment vent backup nitrogen bottles into procedures. Specifically, an operator rounds log allowed the two backup nitrogen bottles associated with containment vent valve 1GSHV-4964 to decrease to 200 psig each. The nitrogen capacity calculation assumed a minimum of 800 psig per bottle. Operation below 800 psig did not ensure the minimum number of containment vent valve strokes could be achieved to protect containment against overpressurization. The containment vent valves are credited in PRA for beyond design basis events, and were originally installed in accordance with NRC Generic Letter 89-16, “Installation of a Hardened Wetwell Vent.”

Description: The team reviewed nitrogen capacity calculation H-1-KB-MDC-1007, “Backup Pneumatic Supply for 1GSHV-4964 and 1GSHV-11541," and operator rounds log HC.OP-DL.ZZ-0004, “Log 4 Reactor Building Data Log - Tuesday Night Shift,” to ensure the backup nitrogen bottles would supply sufficient nitrogen to vent containment for beyond design basis events. The nitrogen capacity calculation developed the basis for the minimum allowable nitrogen bottle pressure to ensure the containment vent valve could be operated as described in the EOP when instrument air was not available.

The team noted the operator rounds data sheet allowed the backup nitrogen bottle pressure to reduce to 200 psig before the bottles were replaced, yet the capacity calculation assumed a minimum of 800 psig would be maintained in each of the two nitrogen bottles to ensure sufficient valve operation. The team also noted the calculation may have been non-conservative in assuming a maximum of two valve strokes would be used since EOP HC.OP-EO.ZZ-0318, “Containment Venting,” directed operators to open 1GSHV-4964 as necessary for maintaining drywell pressure before reaching 65 psig.

The operator rounds sheets were changed in June 1998 to allow a minimum of 200 psig in the nitrogen bottles, instead of the necessary 900 psig (800 psig plus an additional 100 psig for conservatism). PSEG administrative procedure, NC.DM-AP.ZZ-0001, “Procedure Administrative Processes,” specifies design engineering be contacted when values are being changed in procedures, including operating bands. The evaluation performed in June 1998 did not address the design impact of lowering the nitrogen bottle operating band. The procedure change evaluation did not ensure the minimum nitrogen supply would be provided to operate the containment vent valve, for beyond design basis scenarios.

The team reviewed past nitrogen bottle pressure data from January 2003 to September 2005, and noted that on several occasions, at least one nitrogen bottle was less than

800 psig. Since these nitrogen bottles are not in the technical specifications, no operability determination was required when pressure was less than 800 psig.

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

Analysis: The team determined this issue was a performance deficiency because PSEG did not ensure that adequate design control measures existed to verify sufficient backup nitrogen supply to operate the containment vent valve. Specifically, operator rounds sheets were revised so that they did not ensure sufficient nitrogen supply to the containment vent valve, under the most limiting conditions. Insufficient nitrogen to operate the containment vent valve, when required for beyond design basis scenarios, could challenge containment heat removal.

This issue was more than minor because it was associated with the design control attribute of the mitigating systems cornerstone, and affected the cornerstone's objective to ensure the availability and reliability of systems that respond to initiating events to prevent undesirable consequences. Traditional enforcement does not apply because the issue did not have any actual safety consequence or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the team conducted a Phase 1 SDP screening and determined the finding was of very low safety significance (Green) because it did not result in a loss of safety system function as the vent valve local hand-jack function was available to open the containment vent.

Enforcement: This finding was not a violation of NRC requirements, in that the performance deficiency involved a beyond design basis event. Further, the containment vent function is not covered by technical specifications, and is not a part of Hope Creek's licensing basis. PSEG entered this problem into their corrective action program as notification 20305386. **(FIN 05000354/2006015-02, Inadequate Containment Vent Valve Backup Pneumatic Supply)**

.2.1.5 High Pressure Coolant Injection Steam Supply Valve, H1-FD-HV-F001

a. Inspection Scope

The team inspected the high pressure coolant injection (HPCI) steam supply valve, H1-FD-HV-F001, to verify that it was capable of meeting its design basis requirements. This direct current (DC) motor operated valve has a function to automatically open to supply steam to the HPCI pump turbine during postulated events. The review included system calculations and motor operated valve (MOV) calculations to verify that thrust and torque limits and actuator settings were correct. The team also reviewed the design and operation of HPCI to ensure EPU assumptions did not reduce or invalidate safety analysis margins. Inservice testing results were reviewed to verify that the stroke time acceptance criteria were in accordance with the design bases and accident analysis assumptions. Additionally, notifications related to the valve were reviewed to ensure conditions did not exist that would invalidate previous assumptions for the capability of the valve. The team verified the system operating conditions and terminal voltage values used in the valve analyses were bounding. The team also verified that this valve was not susceptible to thermal binding or pressure locking conditions.

b. Findings

No findings of significance were identified.

.2.1.6 Residual Heat Removal Heat Exchanger Bypass Valve, H1-BC-HV-F048A

a. Inspection Scope

The team inspected the residual heat removal (RHR) heat exchanger bypass valve, H1-BC-HV-F048A, to verify that it was capable of meeting its design basis requirements. This alternating current (AC) valve was required to close to ensure adequate RHR flow through the RHR heat exchangers under postulated accident conditions. The review included system calculations and motor operated valve calculations to verify appropriate actuator settings. Inservice testing results were reviewed to verify that the stroke times were in accordance with the acceptance criteria. The team verified the system operating conditions and terminal voltage values used in the valve analyses were bounding. The team also reviewed the control logic associated with this valve to verify it would perform its design basis function.

b. Findings

No findings of significance were identified.

.2.1.7 RHR Heat Exchanger SACS Isolation Valve, H1-EG-HV-2512A

a. Inspection Scope

The team inspected the RHR heat exchanger SACS Isolation Valve, H1-EG-HV-2512A, to verify that it is capable of meeting its design basis requirements. This AC valve was required to open to ensure adequate SACS flow through the RHR heat exchangers under postulated accident conditions. The review included system calculations and motor operated valve calculations to verify appropriate actuator settings. Inservice testing results were reviewed to verify that the stroke times were in accordance with the acceptance criteria. The team verified the system operating conditions and terminal voltage values used in the valve analyses were bounding. The team reviewed the control logic associated with this valve to verify it would perform its design basis function. The team also reviewed a temporary modification and a design change associated with throttling this valve.

b. Findings

No findings of significance were identified.

2.1.8 Reactor Core Isolation Cooling System Turbine-Driven Pump, BD-OP203

a. Inspection Scope

The team inspected the reactor core isolation cooling system (RCIC) turbine-driven pump, BD-OP203, to verify it was capable of meeting its design basis requirement of automatically providing high pressure cooling water to the reactor vessel under postulated transient conditions, including station blackout events. This review included various RCIC system calculations, instrument setpoint calculations, summaries of various in-service testing results, and notifications related to the RCIC pump and turbine. The team verified the capability of the RCIC pump to provide its design flowrate to the reactor vessel. The team also reviewed the design and operation of RCIC to ensure EPU assumptions did not reduce or invalidate safety analysis margins. In addition, the team verified the bases for the pump inservice testing acceptance criteria, the bases of various setpoints associated with the pump and turbine, and the availability of adequate net positive suction head during RCIC pump operation. Finally, the team reviewed the control logic associated with pump operation and the transfer of the suction source from the condensate storage tank to the suppression pool.

b. Findings

No findings of significance were identified.

.2.1.9 High Pressure Coolant Injection System Turbine-Driven Pump, BJ-OP204

a. Inspection Scope

The team inspected the high pressure coolant injection (HPCI) system turbine-driven pump, BJ-OP204, to verify it was capable of meeting its design basis requirement of automatically providing high pressure cooling water to the reactor vessel under postulated accident conditions. This review included various HPCI system calculations, instrument setpoint calculations, summaries of various inservice testing results, and notifications related to the pump and turbine. The team verified the capability of the HPCI pump to provide its design flowrate to the reactor vessel. In addition, the team verified the bases for the pump inservice testing acceptance criteria, the bases of various setpoints associated with the pump and turbine, and the availability of adequate net positive suction head during HPCI pump operation. Finally, the team reviewed the control logic associated with pump operation and the transfer of the suction source from the condensate storage tank to the suppression pool.

b. Findings

No findings of significance were identified.

.2.1.10 Safety Auxiliaries Cooling System ‘B’ Pump

a. Inspection Scope

The team inspected the ‘B’ Safety Auxiliaries Cooling System (SACS) pump, BP210, to verify it was capable of meeting its design basis requirement of providing cooling water to safety-related equipment under postulated accident conditions. This review included various SACS system calculations, instrument setpoint calculations, summaries of various in-service testing results, and notifications related to the pump. The team verified the capability of the SACS pump to provide its design flowrate, considering a single failure. The team also reviewed the design and operation of the SACS pump to ensure EPU assumptions did not reduce or invalidate safety analysis margins. In addition, the team verified the bases for the pump inservice testing acceptance criteria, the bases of various setpoints associated with the pump, the availability of adequate net positive suction head, and the control logic associated with pump operation.

b. Findings

No findings of significance were identified.

.2.1.11 Division A 250 Vdc Battery Charger 10D423

a. Inspection Scope

The team reviewed the 250 Vdc battery charger 10D423, the associated 250 Vdc station battery (10D421) and the 250 Vdc class 1E vital bus (10D450) to verify the components

were adequately sized to meet the operation requirement of the loads, and that the minimum voltage requirements at the loads were analyzed. Specifically, the evaluation focused on verifying that the battery charger, battery and vital bus were adequately sized to supply the design duty cycle of the 250 Vdc system for both the postulated loss-of-offsite power/loss-of-coolant accident and station blackout (SBO) loading scenarios, and that adequate voltage would remain available for individual load devices required to operate during a four-hour SBO coping duration. Loss of ventilation calculations and component data sheets were reviewed to ensure that room temperatures would not rise enough to impede operability of the selected components. Additionally, a walkdown was performed to determine physical and material condition of the selected components and confirm that the battery charger and battery room temperatures were within specified design temperature ranges. The team reviewed battery charger and battery surveillance test results to verify that applicable test criteria and test frequency requirements specified for the battery charger and battery were met. The cognizant design and system engineers were interviewed regarding design aspects and operating history for the battery charger and the associated battery.

b. Findings

No findings of significance were identified.

.2.1.12 Primary (H1ME-10-R-106) and Backup (H1ME-10-R-112) 125 Vdc Switchyard Batteries

a. Inspection Scope

The team reviewed the primary and backup station switchyard batteries as they supply indication and control power for the 500 kVac switchyard circuit breakers and protective relays. Specifically, the team reviewed an engineering evaluation detailing the battery sizing requirements and duration of the batteries under loss of AC power scenarios to verify that the batteries could provide sufficient power to operate the 500 kVac circuit breakers in the switchyard. Additionally, the team reviewed preventive maintenance procedures, completed surveillance procedures, and system health reports to ensure no significant issues existed for the switchyard batteries. A walkdown of the switchyard batteries and components was conducted to verify the material condition.

b. Findings

No findings of significance were identified.

.2.1.13 Gas Turbine Generator (Salem Unit 3)

a. Inspection Scope

The team reviewed the electrical design and operating procedures for the gas turbine generator to assess operating performance during a postulated SBO event at Hope Creek. While not credited for an SBO event at Hope Creek, the PRA and the recovery of AC power procedures allow the alignment of the gas turbine generator to Hope Creek.

The review included operating procedures, switchyard alignment, calculations and engineering analyses to determine if the gas turbine generator could be successfully aligned to provide AC power to Hope Creek. The team also reviewed routine surveillance tests, system health reports and notifications associated with the gas turbine generator. A walkdown of the gas turbine generator and associated systems/components was conducted to assess material condition.

b. Findings

No findings of significance were identified.

.2.1.14 High Pressure Coolant Injection Suppression Pool Level Transmitters

a. Inspection Scope

The team reviewed the design basis setpoint requirements, calibration procedures, calibration calculations and alarms for HPCI suppression pool level transmitters H1BJ-1BJLT-N062E-E41 (High) and H1BJLT-4805-1 (Low). The team reviewed the electrical supply and loop calibration of the transmitters to ensure the components would operate as designed during postulated accident conditions. The team also reviewed completed surveillance procedures and calibration data sheets to verify appropriate testing criteria had been established, results were within acceptable ranges and there were no repetitive degradations or failures.

b. Findings

No findings of significance were identified.

.2.1.15 Design and Engineering Aspect of Ability to Cross-tie ‘B’ and ‘D’ Battery Chargers (Supports Operator Action NR-XTIE-CHARGE)

a. Inspection Scope

The team reviewed the engineering aspects of the actions necessary to cross-tie power to the ‘B’ and ‘D’ 125 Vdc battery chargers. This review was conducted to support operator action NR-XTIE-CHARGE, which is one of the top ten most important operator actions on the Hope Creek PRA risk summary chart. The team’s review considered procedures that would allow operators to conduct the appropriate actions, operator training to conduct the necessary actions to cross-tie the battery chargers and a walkdown of staged hardware specifically for the associated operator actions. Additionally, the team reviewed battery sizing and charger calculations to verify they were appropriately sized. A walkdown of the chargers and their associated batteries was conducted to verify material condition of the components and to ensure that the chargers were energized and indicating appropriate voltage and current readings. The team also verified that room temperatures were appropriate to support charger and

battery operation. Recent completed surveillances and system health reports were reviewed to determine if Hope Creek had operating issues with the reviewed battery chargers and their associated batteries.

b. Findings

The team identified an issue relative to the ability to cross-tie power to the 'B' and 'D' 125 Vdc battery chargers that support operator action NR-XTIE-CHARGE. The issue was associated with a licensee PRA evaluation that indicated there were no procedures to accomplish the necessary operator actions. Additionally, the team determined that training was not provided to the operators to perform the necessary actions, and hardware (e.g., appropriate wire, tools, etc.) was not staged. Without procedural guidance, operator training and staged hardware, the team questioned the appropriateness of crediting this action in the plant's risk assessment model. This issue was determined not to involve a violation of regulatory requirements. Specifically, the battery charger cross-tie operator action was not part of the licensing bases, not covered by technical specification requirements, and was not credited in the station blackout analysis. However, the NRC determined that this issue warranted documentation because weaknesses in PRA risk models can impact risk assessments completed in accordance with Maintenance Rule requirements (10 CFR 50.65) as well as the implementation of the NRC's Reactor Oversight Process. The PRA model is also an important tool in the determination of the risk based performance indicators.

For postulated loss-of-offsite power (LOOP) events with an assumed concurrent failure of the 'B' and 'D' EDGs, the Hope Creek PRA model credits manual operator actions to cross-tie the 'B' and 'D' battery chargers to provide power to the safety relief valves (SRV) for reactor pressure control. For the action to be successful, the action has to be completed prior to the depletion of the batteries.

Several standards provide guidance on the technical adequacy of plant PRA results, including verifying or validating risk significant assumptions:

- NUREG/CR-4772, Accident Sequence Evaluation Program Human Reliability Analysis (HRA) Procedure;
- NRC Regulatory Guide 1.200, An Approach for Determining the Technical Adequacy of PRA Results for Risk-Informed Activities;
- NRC Standard Review Plan Chapter 19.1, Determining the Technical Adequacy of PRA Results for Risk-Informed Activities;
- ASME RA-S-2002, Standard for PRA for Nuclear Plant Applications
- NEI 2000-02, PRA Peer Review Process Guidance; and
- NEI Revision to NUMARC 93-01 Section 11, Assessment of Risk from Maintenance Activities.

These documents, as well as additional licensee procedures, such as ER-AA-600, "Risk Management," and the Hope Creek HRA specify the requirements and responsibilities of the risk management program. They state that the PRA model shall be maintained reasonably representative of the as-built, as-operated nuclear power unit which it

represents. Further, the Hope Creek HRA (Table 4.2-1) states that the human error probability (HEP) should be assigned 1.0 in the case where critical skill-based or rule-based post diagnosis actions are not described in written procedures. However, the current PRA calculated a HEP of 0.6 (60% chance of failure). Accordingly, the team determined that the Hope Creek PRA may not reflect a reasonable representation of the current plant design with respect to the ability to cross-tie the 'B' and 'D' battery chargers.

In response to the team's observation and PSEG's self-identification that there were no procedures to accomplish the battery charger cross-tie operator actions, PSEG revised procedure HC.OP-AM.TSC-0004, "Alternate Power Supply to 1E 125/250 Vdc," to include the necessary steps to crosstie the battery chargers.

The extent that the apparent incorrect HEP value affected the PRA accuracy and representation of plant risk is unresolved pending further NRC review, which may include SPAR model revision and evaluation. The NRC will also pursue the more generic question related to the extent that PRA quality issues are reviewed and evaluated during the component design bases inspection. Upon completion of this review, the NRC will characterize and resolve this issue. **(URI 05000354/2006015-03, Inspection of PRA Quality Issues, and NRC Review of HEP Assigned Value for Battery Charger Cross-tie Operator Action)**

.2.1.16 4160 Vac - Vital Bus 10A401

a. Inspection Scope

The team reviewed the electrical capabilities of the 4160 Vac Vital Bus 10A401 and the capabilities of the normal and alternate offsite power supply circuit breakers to verify that the system design basis requirements were adequately considered. The team reviewed selected load flow, short circuit, voltage drop, degraded voltage and protective relay coordination calculations to determine whether the 4160 Vac Vital Bus 10A401 and Supply Breakers 52-40101 and 52-40108 were capable of supplying the minimum voltage necessary to ensure proper operation of the connected equipment during normal and accident conditions. The team reviewed a sample of preventive circuit breaker maintenance and relay calibration test results to verify whether appropriate acceptance criteria were provided. Included in this review were the trip setpoint calculations of the 4160 Vac Bus 10A401 differential, overcurrent and loss-of-voltage relays to verify that adequate coordination was provided.

The team reviewed system health reports, design basis documents, and samples of corrective action notifications to assess equipment performance history. The team interviewed system and design engineers, and conducted field and control room walkdowns to observe whether the installed equipment configuration, circuit breaker position indicating lights and instrument meter readings were consistent with the plant design drawings and that the observable material condition was acceptable.

b. Findings

No findings of significance were identified.

.2.1.17 'A' Emergency Diesel Generator (electrical) and Output Circuit Breaker 52-40107

a. Inspection Scope

The team reviewed the electrical capabilities of the 4160 Vac 'A' EDG and that of its output circuit breaker, 52-40107. Specifically, the team reviewed: 1) load flow analysis and voltage drop calculations to verify that adequate voltage was provided to meet minimum voltage specifications for the safety-related electrical loads during worst case loading conditions; and 2) protective relay trip setpoint calculations to verify that the trip setpoints would not spuriously interfere with the equipment fulfilling its safety function, and secondarily, that adequate protection was provided. The team also reviewed vendor manuals, surveillance test results and relay calibration test records to verify that the calibrations were within the calculated limits and that excessive instrument drift did not occur.

The team reviewed the EDG output circuit breaker 52-40107 control logic circuit drawings to verify the proper functioning of close and trip functional requirements, including interlocks and protective device trip-output bypass circuits.

The team performed field and control room walkdowns to verify: 1) that the installed local and remote EDG and circuit breaker control switches and breaker position indicating lights were consistent with design drawings; 2) that the system alignments were consistent with design and licensing basis assumptions; and 3) that the observable material condition was acceptable. Finally, the team witnessed the performance of a monthly operability test on the 'A' EDG.

b. Findings

No findings of significance were identified.

.2.1.18 'A' Residual Heat Removal Pump Motor/Breaker

a. Inspection Scope

The team reviewed the electrical power supply calculations and samples of completed maintenance and functional performance test results to verify that the power supply for the 'A' RHR pump motor and the control logic design for the RHR breaker 52-40106 were adequate and properly maintained to meet the functional requirements during normal and accident conditions.

The team reviewed the electrical protective relay settings associated with the 'A' RHR pump motor circuit to verify that the trip setpoints would not spuriously interfere with the pump fulfilling its safety function, and secondarily, that adequate motor feeder circuit

protection was provided. The team reviewed the RHR motor feeder cable ampacity calculation to verify that the cable was adequately sized for the RHR pump motor load, and that adequate protection was provided. The breaker ratings were reviewed to verify that they were adequate for the short circuit duty requirements. The degraded voltage relay setpoints were reviewed to verify that adequate design considerations were provided to meet minimum voltage specifications for motor starting and running.

b. Findings

No findings of significance were identified.

.2.1.19 4160 Vac - Fast Bus Transfer

a. Inspection Scope

The team reviewed PSEG's commitments described in the UFSAR, technical specifications and design basis documents to determine the requirements for the automatic and manual power supply transfer to the 4160 Vac vital buses. The team reviewed calculations, procedures, drawings and completed power supply transfer functional surveillance test results to determine whether the capability of the automatic and manual power supply transfer circuits was adequately demonstrated through functional validation tests as required in technical specifications and the UFSAR.

The team reviewed the basis of the protective and auxiliary relay settings used in the 4160 Vac vital bus transfer breaker control circuits to verify the 4160 Vac bus transfer requirements were met. Included were the electrical interlocks to assure that the 4160 Vac vital buses are powered from one offsite source at a time and that the bus supply breakers are not closed onto a faulty bus.

b. Findings

No findings of significance were identified.

.2.1.20 Station Service Transformer 1AX501

a. Inspection Scope

The team reviewed calculations, drawings and completed maintenance test results to determine whether the 13.8/4.16 kV station service transformer (SST) 1AX501 was adequately designed to supply power from the offsite source to the 4.16 kV safety-related buses. The team reviewed the adequacy of SST design assumptions and calculations. The team reviewed the transformer nameplate design information, vendor test data, system short circuit and loading studies and protective relay coordination calculations to verify the adequacy of the protective relay trip settings and equipment ratings. The team also performed independent short circuit and relay setting calculations to verify that the trip setpoints would not spuriously interfere with transformer operation during normal and accident loading conditions. The team

reviewed samples of completed preventive maintenance work orders, recent design changes affecting the automatic load tap changer, relay calibration test records and system health reports. Finally, the team conducted a field walkdown of the station service transformers to verify the equipment nameplate data was consistent with that contained in plant design documents. The walkdown also served to determine whether the observable material condition was acceptable.

b. Findings

No findings of significance were identified.

.2.2 Detailed Operator Action Reviews (5 Samples)

The team assessed manual operator actions and selected a sample of five operator actions for detailed review based upon risk significance, time urgency, and factors affecting the likelihood of human error. The operator actions were selected from a PRA ranking of operator action importance based on RAW and RRW values. The non-PRA considerations in the selection process included the following factors:

- Margin between the time needed to complete the actions and the time available prior to adverse reactor consequences;
- Complexity of the actions;
- Reliability and/or redundancy of components associated with the actions;
- Extent of actions to be performed outside of the control room;
- Procedural guidance; and
- Training.

.2.2.1 AC Power Recovery

a. Inspection Scope

The team selected operator actions to recover AC power to the safety related buses via the preferred power supply (PPS) or the offsite transmission system. Loss-of-offsite power events is 52% of Hope Creek initiating events and the PPS contributes 37% to the Hope Creek core damage frequency, second only to the EDGs. The potential consequence of failure of this action is core damage at about 75 minutes after the high pressure coolant injection and reactor core isolation cooling system batteries are depleted. The incorporation of this action into site procedures, classroom training, and simulator training was reviewed. The team also walked down recovery of the PPS to the safety-related 4160 Vac buses with licensed and non-licensed operators to verify that PSEG could restore AC power within the SBO coping duration.

b. Findings

No findings of significance were identified.

.2.2.2 Containment Venting

a. Inspection Scope

The team selected the operator action to initiate containment venting when pressure cannot be maintained below the primary containment pressure limit of 65 psig. Safety relief valves are also expected to close at about 69 psig containment pressure. Failure to initiate containment venting has the highest RAW of all Hope Creek post-initiator operator actions, and emergency venting of the containment contributes 12% to the overall core damage frequency. The potential consequence of failure of this action is primary containment over-pressurization and reactor pressure vessel re-pressurization after the safety relief valves close. The incorporation of this action into site procedures, classroom training, and job performance measures was reviewed. The team also walked down procedures to remotely and locally operate the hardened torus vent valves with a licensed reactor operator. Finally, the team walked down the equipment alignment of the compressed gas system that supports the hardened torus vent valves.

b. Findings

No findings of significance were identified.

.2.2.3 Safety Auxiliaries Cooling System Heat Load Manipulation

a. Inspection Scope

The team selected the operator action to reduce SACS heat loads during various SACS or service water system malfunctions. Specifically, operators may need to reduce SACS flow or residual heat removal system flow during suppression pool cooling to prevent overheating other SACS cooled equipment, e.g., EDGs. The time window available for alignment is about 35 minutes with a required manipulation time of five minutes. The potential consequence of failure of this action is core damage after the EDGs overheat and the high pressure coolant injection and reactor core isolation cooling systems batteries are depleted. The team reviewed the incorporation of this action into abnormal and annunciator response procedures. The team interviewed control room operators and assessed the cues available to initiate operator response. Finally, the team reviewed operator guidance to prevent excessive cycling of motor operated valves.

b. Findings

No findings of significance were identified.

.2.2.4 ‘B’ Core Spray Loop Alignment to the Condensate Storage Tank

a. Inspection Scope

The team selected the operator action to align the ‘B’ core spray loop suction to the condensate storage tank. The condensate storage tank provides a source of cold

injection water through an installed test line to the 'B' core spray loop when net positive suction head is reduced due to saturated conditions or suction strainer blockages within the suppression pool. (The 'A' core spray loop has a similar test line but valves are inaccessible during accident conditions.) The Hope Creek PRA group listed this action as one of the most important operator actions. The potential consequence of failure of this action is core damage if emergency core cooling systems operate to the point of reaching a saturated suppression pool condition. The team reviewed the incorporation of this action into emergency operating procedures, abnormal operating procedures, job performance measures, and training. The team assessed the material condition of the normally locked closed manual suction valves and reviewed preventive maintenance measures to assure successful operation of the valves.

b. Findings

No findings of significance were identified.

.2.2.5 Diesel Fire Pump Alignment for Alternate Injection

a. Inspection Scope

The team selected the operator action to manually align the fire protection system to the low pressure coolant injection (LPCI) valves. Manually aligning alternate injection systems through the LPCI injection valves is one of the last resort methods of reactor pressure vessel injection and could be used during a sustained station blackout. The potential consequence of failure of this action is core damage after all emergency core cooling systems fail. The team verified that PSEG staged and inventoried all necessary equipment and tools in an appropriate location to expeditiously align the fire protection system. The team observed an equipment operator access the emergency equipment and tools and walk through the actions to initiate alternate injection. The incorporation of this action into site procedures, classroom training, and job performance measures were also reviewed.

b. Findings

No findings of significance were identified.

.3 Review of Industry Operating Experience (OE) and Generic Issues (5 Samples)

a. Inspection Scope

The team reviewed selected OE issues for applicability at Hope Creek. The team performed a detailed review of the OE issues listed below to verify that PSEG had appropriately assessed potential applicability to site equipment.

.3.1 NRC Information Notice (IN) 2006-09: Performance of NRC-Licensed Individuals While on Duty with Respect to Control Room Attentiveness

The team reviewed PSEG's disposition of IN 06-09. This notice emphasized the importance and professional responsibility of control room operators to be alert to prevent or mitigate any operational problems. The team verified that PSEG conducted training on IN 06-09 with the operations staff. The team also verified that PSEG reviewed administrative controls that establish acceptable standards of operation.

.3.2 NRC IN 2006-22, New Ultra-Low-Sulfur Diesel Fuel Oil Could Adversely Impact Diesel Engine Performance

The team reviewed PSEG's assessment of the potential impact on EDG operation with the use of the new ultra low sulfur fuel. The team reviewed fuel receipt forms, technical evaluations, and PSEG's assessment of potential impact on the EDGs. A sample of notifications associated with EDG issues was also reviewed to verify that the problems identified in IN 06-22 had not occurred.

.3.3 NRC Generic Letter (GL) 1995-07: Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves

The team reviewed the applicability and disposition of GL 95-07. The NRC issued this generic letter to request that licensee's perform or confirm that they previously performed (1) evaluations of operational configurations of safety-related, power-operated gate valves for susceptibility to pressure locking and thermal binding, and (2) further analyses and any needed corrective actions to ensure that those valves which were susceptible were capable of performing the safety functions within the current licensing bases of the facility. The team sampled HPCI Steam Supply Valve H1-FD-HV-F001. The team reviewed both the original GL 95-07 evaluation and the current MOV analyses. This review included verifying that the inputs and assumptions used in the original evaluation remained valid.

.3.4 NRC Inspection Report 0500034/2005016, Fermi Unit 2: Effects of Frequency Variations on Diesel Generator Loading Calculations

This is an NRC inspection finding concerning the licensee's failure to consider the effects of electrical frequency variation on EDG loading. Specifically, EDG loading calculations failed to account for the increased loading that could result from the allowable frequency variations above the nominal generator frequency of 60Hz. The team reviewed PSEG's response and actions documented in Notification 20287483/Order 70059019 and in calculation E-9 Standby Class 1E Diesel Generator Sizing, Rev. 8A; addressing the effects of electrical frequency variation on EDG operation. The areas reviewed included the effects of increased currents on existing relay setting and protective relay coordination requirements.

.3.5 NRC IN 2005-15: Three-Unit Trip and Loss of Offsite Power at Palo Verde Nuclear Generating Station

The team reviewed PSEG's assessment of this single failure susceptibility of transmission line protective relaying and circuit breaker control systems and the resultant loss of offsite power. The team reviewed PSEG's response and actions documented in Notification 20242345/Order 70048464 addressing the applicability of the issue identified in NRC IN 05-15, including circuit breaker trip coils, auxiliary tripping relays, primary and backup transmission line protective relaying schemes, breaker failure protection and equipment testing and maintenance design philosophy and practices.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Problem Identification and Resolution

a. Inspection Scope

The team reviewed a sample of problems that were identified by the licensee and entered into the corrective action program. The team reviewed these issues to verify an appropriate threshold for identifying issues, and to evaluate the effectiveness of corrective actions related to design or qualification issues. In addition, corrective action notifications written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problem into the corrective action system. The specific corrective action documents that were sampled and reviewed by the team are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

4AO6 Meetings, Including Exit

On December 7, 2006, the team presented the inspection results to Mr. G. Barnes, Site Vice President, Hope Creek, and other members of licensee staff. The team verified that no proprietary information is documented in the report.

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

J. Barstow, Regulatory Affairs/Compliance Engineer
T. MacEwen, Online Work Control Manager
E. Ortalan, Mechanical Design Manager
S. Seyedhossini, Design Engineer
M. Tadjalli, Senior Manager, Design Engineering
S. Afarian, System Engineer
A. Bready, PRA Engineer
K. Petroff, System Engineer
F. Ricart, Design Engineer
D. Schiller, System Engineer
B. Swartley, Design Engineer
V. Warren, PRA Engineer

NRC Personnel

G. Malone, Senior Resident Inspector
C. Cahill, Senior Risk Analyst

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000354/2006015-03	URI	Inspection of PRA Quality Issues, and NRC Review of HEP Assigned Value for Battery Charger Cross-tie Operator Action (Section 1R21.2.1.15)

Opened and Closed

05000354/2006015-01	NCV	Inadequate Strainer Differential Pressure Design Control to Ensure Adequate Service Water Flow (Section 1R21.2.1.2)
05000354/2006015-02	FIN	Inadequate Containment Vent Valve Backup Pneumatic Supply (Section 1R21.2.1.4)

LIST OF DOCUMENTS REVIEWED

Calculations

108-173-PSD-1, Evaluation of Valve 1FDHVF001 under Thermal Binding Conditions, Rev. 1
AP-0004, Condensate Storage Tank - Level Setpoints, Rev. 6
BC-0002, NPSH for RHR System Pumps, Rev. 5
BC-0056, RHR Hydraulic Analysis, Rev. 3
BD-0001, NPSH for RCIC System Pump, Rev. 3
BD-0003, RCIC Hydraulic Analysis, Rev. 5
BJ-0001, NPSH for HPCI System Pump (Suction from Suppression Chamber), Rev. 5
BJ-0002, NPSH for HPCI System Pump (Suction from Condensate Storage Tank), Rev. 5
BJ-0023, HPCI Full Flow Test Required, Rev. 5
E-1.3, Hope Creek Generating Station Short Circ. Study of 480V Systems, Rev. 3
E-7.4, Class 1E 4.16KV System Protective Relay Settings, Rev. 4
E-15.5, Hope Creek Fast Bus Transfer Analysis, Rev.4
E-15.1, Hope Creek Degraded Voltage Analysis, Rev. 7
E-7.6, Diesel Generator Protective Relaying, Rev. 0
E-9, Standby Class 1E Diesel Generator Sizing, Rev. 8A
E-1.4, Class 1E 125 & 250 Vdc Systems: Short Circuit & Voltage Drop Studies, Rev. 5
E-10, Medium Voltage Cable Ampacity, Rev. 4
E-18, Selection of Overload Heaters for AC MOVs and Continuous Duty Motors, Rev. 2
E-4.1, Class 1E 125 Vdc Station Battery & Charger Sizing, Rev. 14
E-45.001, 250 Vdc Battery Capacity Verification Calculation, Rev. 1
E-5.1, Class 1E 250 Vdc Station Battery & Charger Sizing, Rev. 7
E-7.9, 125 Vdc and 250 Vdc Class 1E System (Coordination), Rev. 3
E-9, Standby Class 1E Diesel Generator Sizing, Rev. 8
EA-0003, Station Service Water System Hydraulic Analysis, Rev. 6
EA-0030, Emergency Service Water Flow to the Reactor Vessel, Rev. 0
EG-0008, Process Set-points for STACS Temperature Controlling Devices, Rev. 2
EG-0020, STACS - Required Flows and Heat Loads, Rev. 8
EG-0026, TACS Isolation Valve Setting, Rev. 1
EG-0041, Reduced SACS Flow for RHR Shutdown Cooling, Rev. 0
EG-0046, STACS Operation, Rev. 7
EG-0047, HCGS Ultimate Heat Sink Temperature Limits, EPU, Rev. 5
ES-4.007, Salem Unit 3, Battery Sizing Calculation, Rev. 0
GR-0022, Loss of Ventilation During SBO, Rev. 1
H-1-AB-MDC-2024, Main Steam SRV Tailpipe Temperature Monitoring Criteria, Rev. 0
H-1-BC-MDC-0922 MOV Capability Assessment for 1BC-HV-F048A, Rev. 0
H-1-BJ-MDC-1997, HPCI Lube Oil System Analysis, Rev. 0
H-1-BJ-MDC-2004, HPCI Pump Assembly Hydraulic Model, Rev. 1
H-1-EG-MDC-0938, MOV Capability Assessment for 1EG-HV-2512A, Rev. 1
H-1-FD-MDC-0941, MOV Capability Assessment for 1FD-HV-F001, Rev. 1
H-1-GK-MDC-0734, Loss of Ventilation During SBO, Rev. 2
H-1-KB-MDC-1007, Backup Pneumatic Supply for 1GSHV-4964 and 1GSHV-11541, Rev. 0

H-1-SN-MDC-0327, SRV Accumulator Sizing for SBO, Rev. 1
H-1-ZZ-REE-1450, Impact of EPU on SBO Conditions/Remote Shutdown Capability, Rev. 0
JE-0015, Diesel Fuel Oil Storage Capacity Design Basis, Rev. 2
JE-9, Diesel Fuel Oil Pump Inlet Line, Rev. 1
KJ-0003, Skid Interconnecting Compressed Air Lines, Rev. 1
KJ-5, Instrument Gas Compressor and Receiver Sizing, Rev. 3
SC-AP-0001, CS Low Level Switch to HPCI & Tank 135,000 Gal. Reserve, Rev. 6
SC-AP-0003, CST Low Level to RCIC, Rev. 7
SC-BB-0210, Reactor Level CS/RHR/HPCI, Rev. 4
SC-BD-0017, Setpoint Calculation for a High RCIC Turbine Exhaust Pressure, Rev. 2
SC-BD-0023, Setpoint Calculation - RCIC Pump Suction Header Trip, Rev. 2
SC-BD-0026-1, Setpoint Calculation for RCIC Pump Discharge Pressure High, Rev. 2
SC-BJ-0003-1, LOOP Tolerance Calculation for HPCI Pump Discharge Line Pressure, Rev. 4
SC-BJ-0004, Set Point Calculation - HPCI (Suppression Pool Level High), 6/9/04
SC-BJ-0008-3, HPCI Suppression Chamber Level Low, 2/22/05
SC-BJ-0060, HPCI Pump Turbine Exhaust - High Pressure, Rev. 2
SC-EA-0001-1, SSW Across Strainer AF509, Rev. 1
SC-EA-0023, SACS Heat Exchanger Low Flow (Low d/p) Uncertainty Calculation, Rev. 3
SC-EA-0037, HC River Water Level Measurement, Rev. 0
SC-EG-0150-1, SACS Heat Exchanger Outlet Overtemperature Protection, Rev. 7
SC-EG-0152-1, SACS Pumps Runout Flow - Low Differential Pressure Trip, Rev. 5
SC-EG-0156-1, SACS Heat Exchanger Outlet Overtemperature Protection, Rev. 5
SC-EG-0346-1, SACS Expansion Tank BT205 Low Low Level Alarm, Rev. 6
SC-JE-0051-1, Diesel Fuel Oil Storage Tank A Level, Rev. 3
SC-JE-0059, Diesel Fuel Oil Day Tank Level, Rev. 6
SC-KJ-0192, Diesel Generator 'A-D' Lube Oil Temperature High from Diesel, Rev. 1

Completed Surveillance Test Procedures

HC.CH-AP.ZZ-0041, EDG Fuel Oil Analysis for Oil Deliveries via Tanker (5/02/06, 9/29/06)
HC.IC-DC.ZZ-0064, GE Ammeters and Voltmeters, Types AB and DB; and Westinghouse Ammeters and Voltmeters, Type KX-241, KA-241, and KC-241 (11/8/04)
HC.IC-GP.ZZ-0069, Dew Point Test (3/15/06)
HC.MD.PM-PB-0002, 4.16KV Breaker Time Response (11/9/04)
HC.MD.PM-PB-0001, 4.16KV Breaker Cleaning and PM (12/10/04, 11/10/05)
HC.MD.PM.ZZ-0001, Oil Filed Transformer PM (4/23/06)
HC.MD.PM.ZZ-0005, General Motor PM (8/12/05)
HC.MD.ST.KJ-0001, EDG TS Surveillance and PM (4/25/06)
HC.MD.ST.PB-0014, 4 KV Degraded Voltage Channel Calibration/Functional Test (4/24/06)
HC.MD.ST.PB-0015, 4 KV Degraded Voltage Channel Calibration/Functional Test (4/25/06)
HC.MD-GP.ZZ-0006, Electrical Component Meggering Procedure (11/12/04)
HC.MD-GP.ZZ-0015, Battery Equalizing Charge (4/5/02)
HC.MD-PM.ME-0001, Monthly Battery Surveillance -Switchyard (9/15/06)
HC.MD-PM.ME-0002, Switchyard 125 Vdc Quarterly Battery Surveillance (9/15/06)
HC.MD-PM.ME-0003, 500KV Switchyard Batteries Performance Discharge Test (8/10/04)

HC.MD-PM.PB-0001, 4.16 KV Breaker Cleaning and Preventive Maintenance, (7/22/05)
HC.MD-PM.ZZ-0003, Battery Charger Maintenance (11/13/04)
HC.MD-PM.ZZ-0004, Limitorque Valve Operator Inspection and Lubrication (9/17/02)
HC.MD-PM.ZZ-0006, General PM for Distribution Panels and Switchgear (11/12/04, 2/24/06)
HC.MD-ST.PJ-0006, PM of 250 Volt Battery Chargers and Associated Surveillance (4/8/06)
HC.MD-ST.PJ-0008, 250 Volt Station Batteries 18 Month Service Test (8/24/04)
HC.MD-ST.PK-0002, 250 Volt Quarterly Battery Surveillance (4/8/06, 8/15/06, 4/11/06)
HC.MD-ST.PK-008, 125 Volt Battery Chargers Service Test (11/4/05, 11/23/05, 3/26/06, 8/4/06)
HC.OP-ST.KJ-004, 'D' EDG Operability Test (11/28/06)
HC.OP-ST.PB-0002, AC Power Supply Transfer Function Test (10/28/04, 4/25/06)
HC.OP-ST.PB-0005, 4.16 KV Bus 10A401 UV Test (10/29/04, 4/30/06)
HC.OP-DL.ZZ-0003, Log 3 Control Console Log for SRV Tailpipe Temperatures, (11/8/06)
HC.OP-DL.ZZ-0004, Log 4 Reactor Building Data Log (11/7/06, 11/14/06)
HC.OP-GP.EA-0001, SW Emergency Makeup Deadleg Flushing, (4/21/06)
HC.OP-IS.BC-0001, 'A' RHR Pump Inservice Test, (10/10/06)
HC.OP-IS.BC-0101, RHR Subsystem 'A' Valves - Inservice Test (11/24/04, 10/11/06)
HC.OP-IS.BD-0001, RCIC Pump - Inservice Test (5/4/06, 7/6/06, 9/26/06)
HC.OP-IS.BJ-0001, HPCI Main/Booster Pump Set - Inservice Test, (5/4/06, 6/23/06, 10/11/06)
HC.OP-IS.BJ-0101, HPCI System Valves - Inservice Test (3/2/05, 10/11/06)
HC.OP-IS.EA-0002, 'B' SW Pump Inservice Test (3/23/06, 6/10/06, 9/2/06)
HC.OP-IS.EA-0102, SW Subsystem 'B' Valves - Inservice Test (10/7/06)
HC.OP-IS.EG-0002, 'B' SACS Pump - Inservice Test, (5/8/06, 7/6/06, 9/28/06)
HC.OP-IS.EG-0101, SACS System - Subsystem 'A' Valves - Inservice Test (6/27/04, 7/26/06)
HC.OP-IS.GS-0101, Containment Atmosphere Valves - Inservice Test (3/27/06)
HC.OP-ST.BD-0003, RCIC Functional Verification (11/10/06)
HC.OP-ST.BD-0004, RCIC Flow Verification (5/3/06)
HC.OP-ST.KJ-0001, 'A' EDG Operability Test (11/2006)
HC.OP-ST.KJ-0002, 'B' EDG Operability Test (9/15/06)
HC.OP-ST.KJ-0015, 'B' EDG 24 Hour Operability Run and Hot Restart Test (6/12/05)
HC.OP-ST.ZZ-0009, MOV Thermal Overload Protection Surveillance (4/30/06)
NC.MD-GP.ZZ-00052, Foreign Material Exclusion and Closure Control (11/10/04)
S3.OP-PT.JET-0001, Dead Bus Bootstrap Start Test (8/8/04)
S3.OP-PT.JET-0002, Gas Turbine Overspeed Test (12/19/03)
S3.OP-PT.JET-0003, 'A' and 'B' Engines #1 and #2 Fuel Shutoff Valves (12/7/03)
S3.OP-SO.JET-0001, Gas Turbine Operation (8/11/06)
SH.MD.GP.ZZ.0011, Meggering of Rotating Electrical Equipment (4/25/06)
SH.MD-EU.ZZ-0001, Crimping Instructions (11/16/04)
SH.MD-GP.ZZ-0012, Meggering Electrical Equipment (11/12/04)
SH.MD-PM.ZZ-0004, Limitorque Valve Operator Inspection and Lubrication (12/14/04)

Corrective Action Program - Notifications

20054992	20233833	20281837	20303253*	20305777*
20141697	20234211	20281840	20303319*	20305828*
20147230	20234331	20282700	20303362*	20305847*
20199628	20234701	20285857	20303365*	20305893
20210982	20240825	20286373	20303376*	20305907
20217121	20242655	20287480	20303557*	20305912*
20217152	20246789	20287483	20304167	20305949*
20217184	20247025	20287766	20304507*	20305950*
20217471	20249423	20289923	20304577*	20305963*
20217879	20256144	20290015	20304638	20306042
20219155	20260563	20292715	20304777*	20306084*
20221348	20265821	20293710	20304827*	20306165*
20224819	20272479	20294509	20304836*	20306190*
20225114	20273737	20297213*	20305252*	20306225*
20228263	20273739	20297338	20305263*	20306236*
20228504	20274187	20297489	20305382*	20306401*
20232452	20274616	20298147	20305386	20306402*
20232503	20276127	20300039	20305423*	20306475*
20232673	20276570	20300577	20305428*	20306536*
20232789	20279934	20302235	20305721	20306588
20233674	20281208	20303252*	20305727*	20306720*

* Notification written as a result of inspection effort

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30059381	50057842	50077549	70042891	70061678
30075420	50057906	50081117	70046024	70084745
30077629	50057911	50081767	70046270	80018155
30139635	50065245	50083548	70049542	80020757
30140272	50073466	50093903	70052110	80076344
30142146	50074091	50096410	70054745	80080350
40004860	50074092	60025002	70055579	80087728
40018281	50075806	70028860	70056477	80090629
50044583	50077513	70028866	70056600	

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01761770, Electrical Schematic Engine Control, Rev.10
249002A 1818-7, Switching Station AC Distribution One-Line Electrical System, Rev. 7
249004A 1818-5, Switching Station DC Distribution One-Line Electrical System, Rev. 5
249182A 1827-2, Switching Station 13KV Switchyard, Rev. 2
601701S1000-27, Salem/Hope Creek 500, 13.8, 4 KV, Elementary One Line Diagram, Rev. 27
E-0021-1, 480V MCC Tabulation Class 1E MCC - Reactor Area, Sh. 1, Rev. 18
E-2550-0, EDG Fuel Oil Transfer & Standby Pumps, Rev.2

E-6231-0, Electrical Schematic Diagram RHR Valve 1HV-F048A, Sh.2, Rev.5
E-0068-0, Electrical Schematic Diagram Class1E 4.16KV Station Power CB 52-40108, Rev. 10
E-0069-0, Electrical Schematic Diagram Class1E 4.16KV Station Power CB 52-40101, Rev. 8
E-0223-0, Electrical Schematic Diagram SACS Valve HV-2512A, Sh. 1, Rev. 5
E-0006-1, Single Line Diagram 4.16KV Class 1E Power System, Sh.1 & 2, Rev. 11
E-0008-1, Single Line Metering & Relay Diagram Diesel Generators, Rev. 4
E-1465-0, Non-Class1E 13.8KV Differential & Breaker Failure Relay Setting, Sh. 1, Rev.1
E-0001-0, Single Line Diagram Station Electrical, Rev. 24
E-0019-1, 480 Volt MCC Tabulation Class 1E - Auxiliary Building, Sh. 1, Rev. 12
E-0024-1-20, Single Line Meter & Relay Diagram 480 Volt Unit Substations, Rev. 20
E-0118-0, Schematic Meter & Relay Diagram 250 Volt DC System, Sh. 1, Rev. 7
E-1431-0-22, Lighting & Telephone Plan Control and D/G Area, Rev. 22
J-11-0, Logic Diagram - SACS, Sh. 4, Rev. 8
J-11-0, Logic Diagram - SACS, Sh. 5, Rev. 7
J-51-0, Logic Diagram - RHR, Sh. 7, Rev. 8
J-51-0, Logic Diagram - RHR, Sh. 7A, Rev. 1
J-55-0, Logic Diagram - HPCI, Sh. 4, Rev. 9
M-08-0, Condensate & Refueling Water Storage & Transfer, Sh. 1, Rev. 30
M-08-0, Condensate & Refueling Water Storage & Transfer, Sh. 2, Rev. 20
M-10-1-27, Service Water, Sh. 3, Rev. 13
M-10-1-36, Service Water, Sh. 2, Rev. 14
M-10-1-49, Service Water, Sh. 1, Rev. 17
M-11-1, SACS Reactor Building, Sh. 1, Rev. 29
M-11-1, SACS Reactor Building, Sh. 2, Rev. 39
M-11-1, SACS Reactor Building, Sh. 3, Rev. 25
M-12-1, SACS Auxiliary Building, Sh. 2, Rev. 1
M-12-1, SACS Auxiliary Building, Sh. 1, Rev. 31
M-15-0-06, Compressed Air (Instrument), Rev. 6
M-30-1-15, Diesel Engine Auxiliary Systems, Sh. 2, Rev. 10
M-30-1-18, Diesel Engine Auxiliary Systems Starting Air & Lube Oil, Rev. 12
M-30-1-26, Diesel Engine Auxiliary Systems Fuel Oil, Sh. 1, Rev. 19
M-49-1, Reactor Core Isolation Cooling, Rev. 28
M-50-1, RCIC Pump Turbine, Rev. 29
M-51-1, Residual Heat Removal, Sh. 1, Rev. 37
M-51-1, Residual Heat Removal, Sh. 2. Rev. 34
M-55-1, HPCI, Rev. 38
M-55-1-38, High Pressure Coolant Injection, Rev. 5
M-56-1, HPCI Pump Turbine, Rev. 31
M-57-1, Containment Atmosphere Control, Sh. 1, Rev. 40
M-59-1-30, Primary Containment Instrument Gas, Sh. 1, Rev. 30

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80073096, Modify HPCI Discharge Flow Orifice FO-5051 and FE-6813, Rev. 1
80076874, RCIC Turbine Speed Increase, Rev. 0
80090342, Modify EG-HV-2512B Logic to Allow Valve to be Throttled, Rev. 0

80090368, Modify EG-HV-2512A Logic to Allow Valve to be Throttled, Rev. 0
TMP No. 06-020, Jog Circuit for 1EG-HV-2512A, Rev. 0
DCP 4EC-3632, Diesel Fuel Oil Storage Tank Low Level, Rev. 0
DE-CB.EA/EP-0052, CBD for Station Service Water System, Rev. 2
DE-CB.EG-0054, CBD for Safety and Turbine Auxiliaries Cooling System, Rev. 2
DE-CB.KJ-0083, CBD Document for Emergency Diesel Generator System, Rev. 1
DE-CB.PJ-0061, CBD for 250 Vdc Power System, Rev. 1
DE-CB.PK-0062, CBD for 125 Vdc Control Power System, Rev. 1

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HC.IC-CC.FC-0014, RCIC - Division 2 E51-N656F - Turbine Exhaust Pressure, Rev. 5
HC.IC-LC.FD-0001, HPCI Turbine Speed Control System, Rev. 7
HC.OP-AB.COOL-0001, Station Service Water, Rev. 8
HC.OP-AB.COOL-0002, Safety/Turbine Auxiliaries Cooling System, Rev. 1
HC.OP-AB.MISC-0001, Acts of Nature, Rev. 8
HC.OP-AB.ZZ-0001, Transient Plant Conditions, Rev. 7
HC.OP-AB.ZZ-0135, SBO/Loss of Offsite Power/EDG Malfunction, Rev. 24 - 26
HC.OP-AB.ZZ-0155, Degraded ECCS Performance/Loss of NPSH, Rev. 4
HC.OP-AM.TSC-0001, Alternate Injection Using Service Water, Rev. 0
HC.OP-AM.TSC-0004, Alternate Power Supply to 1E 125/250 VDC, Rev. 2
HC.OP-AP.ZZ-0109, Equipment Operational Control, Rev. 14
HC.OP-AR.KJ-0001, Lube Oil Temperature High Alarm, Rev. 16
HC.OP-AR.KJ-0003, FOST No. 1 Level Low & Low Low Alarm, Day Tank Level Low, Rev. 15
HC.OP-AR.MH-0001, Switchyard Gas Breaker Local Panels, Rev. 6
HC.OP-AR.ZZ-0001, Overhead Annunciator Window Box A1, Rev. 17
HC.OP-AR.ZZ-0002, River Water Level Alarm, Rev. 17
HC.OP-AR.ZZ-0005, Overhead Annunciator Window Box A7, Rev. 18
HC.OP-AR.ZZ-0006, Overhead Annunciator Window Box B1, Rev. 20
HC.OP-AR.ZZ-0008, ADS or Safety Valve Not Reseated Alarm, Rev. 32
HC.OP-AR.ZZ-0010, Drywell Pressure Hi Alarm, Rev. 9
HC.OP-AR.ZZ-0017, Overhead Annunciator Window Box E4, Rev. 4
HC.OP-DL.ZZ-0004, Log 4 Reactor Building Data Log - Tuesday Night Shift, Rev. 34
HC.OP-DL.ZZ-0006, Log 6 Auxiliary Building Log, Rev. 43
HC.OP-DL.ZZ-0026, Control Room 2-Hour Readings, Rev. 60
HC.OP-EO.ZZ-0101A-FC, ATWS RPV Control, Rev. 6
HC.OP-EO.ZZ-0102-FC, Primary Containment Control, Rev. 11
HC.OP-EO.ZZ-0102A-FC, ATWS Primary Containment Control, Rev. 7
HC.OP-EO.ZZ-0202-FC, Emergency RPV Depressurization, Rev. 6
HC.OP-EO.ZZ-0308, Alternate Injection Using Service Water, Rev. 4
HC.OP-EO.ZZ-0309, Alternate Injection Using Condensate Transfer, Rev. 4
HC.OP-EO.ZZ-0310, Alternate Injection Using Fire Water, Rev. 5
HC.OP-EO.ZZ-0311, Defeating PCIG Isolation Interlocks, Rev. 6
HC.OP-EO.ZZ-0318, Containment Venting, Rev. 4
HC.OP-EO.ZZ-0319, Restoring Instrument Air in an Emergency, Rev. 1
HC.OP-EO-ZZ.0101-FC, Reactor Pressure Vessel Control, Rev. 10

HC.OP-FT.BD-0002, RCIC System Functional Test (Low Pressure), Rev. 5
HC.OP-IO.ZZ-0008, Shutdown From Outside the Control Room, Rev. 25
HC.OP-IS.AB-0102, Main Steam System Valves - Cold Shutdown - Inservice Test, Rev. 18
HC.OP-IS.EA-0102, Service Water Subsystem B Valves, Inservice Test, Rev. 48
HC.OP-IS.GS-0102, Containment Atmosphere Control System Valves, 18 Months, Rev. 5
HC.OP-IS.JE-0002, B Diesel Fuel Oil Transfer Pump BP401 - Inservice Test, Rev. 22
HC.OP-SO.EA-0001, Service Water System Operation, Rev. 29
HC.OP-SO.EG-0001, Safety/Turbine Auxiliaries Cooling Water System Operation, Rev. 37
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HC.OP-SO.MH-0001, 500kV System, Rev. 19
HC.OP-SO.PB-0001, 4.16kV System Operation, Rev. 21
HC.OP-SO.SN-0001, Nuclear Pressure Relief and ADS Operation, Rev. 6
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S3.OP-SO.JET-0001, Gas Turbine Operation (Salem 3), Rev. 23
S3.OP-SO.JET-0002, Dead Bus Operation - SBO, Rev. 8
S3.OP-SO.JET-0003, Gas Turbine Battery Charger 7 Inverter Operation, Rev. 8
SH.OP-DG.ZZ-0011, Station Seasonal Readiness Guide, Rev. 5

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324205, Station Blackout Analysis for Hope Creek Generating Station, Rev. 6
'B' EDG Starting Air Compressor Action Plan, 10/13/06
10CFR50.59 Screen, HC.OP-AB.COOL-0001 Procedure Change, 5/11/06
10CFR50.59 Screen, HC.OP-DL.ZZ-0004 Procedure Change, 6/29/98
AOV Program Document, Current Status, 3rd Quarter 2006
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DC-89-052, Station Blackout Technical Report, 4/14/89
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DRF-0000-0004-6923, Task T0606, SACS System, Rev. 2
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E41-4010, HPCI System Specification, Rev. 0
EPU Task Report T0604, Station Service Water System, Rev. 1
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GE-NE-0000-0005-8918-01, Task T0404, High Pressure Coolant Injection System, Rev. 1
GE-NE-0000-0031-7552-01, Task T0900, Transient Analysis, Rev. 0
HC PRA-005.07, System Information Notebook - RCIC, 6/11/03
HC-PRA-0004, Hope Creek Generating Station Human Reliability Analysis, Rev. 1
Hope Creek PRA System Information Notebook

Hope Creek UFSAR (various sections)
IST Basis Data Sheets - Torus Purge Exhaust Inboard Containment Isolation Valve, 10/17/97
IST Graphs for 'B' Service Water Pump, 6/27/03-2/26/06
IST Basis Data Sheets - ADS Accumulator Nitrogen Supply Check Valves, 10/17/97
IST Graphs for SSW to RHR Emergency Makeup Drain Valve, 10/13/03 - 10/7/06
IST Graphs for Sw Loop 'B' Emergency M/U Drain Valve, 6/17/04 - 10/7/06
IST Basis Data Sheets - Service Water Makeup Header Auto Drain Valve, 10/17/97
LR-N96179, Response to Request for Additional Information - Generic Letter 95-07, 7/10/96
MPR-2567, Impact of EPU on MOV/AOV Programs and NRC GL 96-06, Rev. 0
NEDO-33076, Safety Analysis Report for HC Constant Pressure Power Uprate, Rev. 2
NOH01EAC00-02, Class 1E AC Power Distribution Lesson Plan, 12/02/04
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NOH01OELERSC-04, OE Feedback - Lesson Plan, 3/16/06
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SG-322, Loss of Offsite Power Followed by a SBO Simulator Scenario Guide, Rev. 1
SRV Tailpipe Temperature Data Trends, Cycle 12 and 13, 6/2/03 - 11/22/06
Technical Specifications (various sections)
Technical Evaluation 70063828, LOOP with Coincident Failure of Two EDGs, Rev. 0
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VTD PE109Q-0185, 1E Electrical EQ - Indoor Metal Clad Switchgear, Rev. 2

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3rd Quarter 2006, 480 Vac Substation Power
3rd Quarter 2006, Containment,
3rd Quarter 2006, Diesel Generators
3rd Quarter 2006, RHR
3rd Quarter 2006, Service Water

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309292, Instruction Book - Westinghouse Sulfur-Hexafluoride Circuit Breakers, Rev. 3
10855-M18(Q) 506-2, Test Report, Emergency Diesel Generator Unit 700002B, Rev. 1
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11909606, Exchanger, Heat Lube Oil, ASME Section III, Rev. 4
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GE-NE-0000-0005-4365-01, HC EPU, NSSS TS Instrument Setpoints, Rev. 1
GE-NE-0000-0047-0285-01, HC EPU, SRV Actuations Under SBO Conditions, Rev. 1
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PM076Q-0044, Service Water Strainer Element Data, Rev. 1
309292, Vendor Manual, 500KV Yard Circuit Breaker, Rev. 6

LIST OF ACRONYMS USED

AC	Alternating Current
CDF	Core Damage Frequency
DC	Direct Current
d/p	Differential Pressure
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
EPU	Extended Power Uprate
GL	[NRC] Generic Letter
gpm	Gallons per Minute
HC	Hope Creek
HEP	Human Error Probability
HPCI	High Pressure Coolant Injection
HRA	Human Reliability Analysis
IMC	Inspection Manual Chapter
IN	[NRC] Information Notice
LOOP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
OE	Operating Experience
PRA	Probabilistic Risk Analysis
psig	Pounds per Square Inch (Gauge)
RAW	Risk Achievement Worth
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
ROP	Reactor Oversight Process
RRW	Risk Reduction Worth
SACS	Safety Auxiliary Cooling System
SBO	Station Blackout
SDP	Significance Determination Process
SPAR	Standardized Plant Analysis Risk
SRV	Safety Relief Valve
SST	Station Service Transformer

SW	Service Water
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
Vac	Volts Alternating Current
Vdc	Volts Direct Current