January 30, 2003

EA-03-018

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

SUBJECT: HOPE CREEK NUCLEAR GENERATING STATION - NRC INSPECTION REPORT 50-354/03-002

Dear Mr. Keiser:

On December 16, 2002, the NRC completed a team inspection at the Hope Creek Generating Station. The enclosed report documents the inspection findings which were discussed on December 16, 2002, with Mr. D. Garchow and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety system design and performance capability of the high pressure cooling injection system and the electrical power system including the emergency diesel generators and offsite power systems, compliance with the Commission's rules and regulations, and with the conditions of your license. The inspection consisted of system walkdowns; examination of selected procedures, drawings, modifications, calculations, surveillance tests and maintenance records; and interviews with site personnel.

Based on the results of this inspection, the team identified six findings, five of which were evaluated individually under the risk significance determination process as having very low safety significance (Green). Five of the six findings were determined to involve violations of NRC requirements. Four violations were entered into your corrective action program, and the NRC is treating these findings as non-cited violations in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny any of these non-cited violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, 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 facility.

One apparent violation involving the failure to conduct a Technical Specification required surveillance test of the emergency diesel generators, and the circumstances surrounding it, are described in detail in the subject inspection report. Regarding this apparent violation, your staff's review of needed actions was not thorough, the actions taken did not restore compliance, and when challenged as to why compliance was not restored, your staff developed a position on the issues with which the NRC disagreed. Subsequently, after the disagreement was made known to your staff, additional actions were taken to restore compliance. Enforcement action for this apparent violation will be handled by separate correspondence at a later date. No response to this apparent violation is required at this time.

We understand that subsequent to the inspection, you initiated an assessment of various activities regarding the inspection and wanted to meet with us to discuss the results. We look forward to meeting with you to discuss the results of your assessment and the inspection.

In accordance with 10CFR2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records component of the NRC's Agency Wide Document and Access 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/ R. V. Crlenjak for:

Wayne D. Lanning, Director
Division of Reactor Safety

Enclosures:
Inspection Report 50-354/03-002

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

cc w/encl:
M. Friedlander, Director - Business Support
J. Carlin, Vice President - Engineering
D. Garchow, Vice President - Projects/Licensing
G. Salamon, Manager - Licensing
R. Kankus, Joint Owner Affairs
J. J. Keenan, Esquire
Consumer Advocate, Office of Consumer Advocate
F. Pompper, Chief of Police and Emergency Management Coordinator
M. Wetterhahn, Esquire
N. Cohen, Coordinator - Unplug Salem Campaign
E. Gbur, Coordinator - Jersey Shore Nuclear Watch
E. Zobian, Coordinator - Jersey Shore Anti Nuclear Alliance
State of New Jersey
State of Delaware

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REGION I

Docket No:	50-354
License No:	NPF-57
Report No:	50-354/03-002
Licensee:	PSEG Nuclear LLC
Facility:	Hope Creek Nuclear Generating Station
Location:	P.O. Box 236 Hancocks Bridge, NJ 08038
Dates:	November 18 - December 16, 2002
Inspectors: Lead	W. Schmidt, Senior Reactor Inspector, Division of Reactor Safety (DRS) S. Pindale, Senior Reactor Inspector, DRS B. Norris, Senior Reactor Inspector, DRS K. Mangan, Reactor Inspector, DRS J. Jaxheimer, Reactor Inspector, DRS G. Skinner, NRC Electrical Contractor S. McCarver, Reactor Inspector, DRS (in training)
Approved By:	Lawrence T. Doerflein, Chief Systems Branch Division of Reactor Safety

TABLE OF CONTENTS

Summary of Findings

IR 05000354/03-002; Public Service Electric Gas Nuclear LLC; on 11/18 - 12/16/02; Hope Creek Generating Station; Safety System Design and Performance Capability.

The inspection was conducted by five region-based inspectors and one NRC contractor. Five findings of very low safety significance (Green) were identified: four were considered non-cited violations; and one was a finding with no violation of NRC requirements. Also, one apparent violation regarding emergency diesel generator surveillance testing was identified. 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 may 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.

Cornerstone: Mitigating Systems

TBD The team identified an apparent violation of Technical Specification (TS) 4.8.1.1.2.h.14 (a, b, and c) because of inadequate testing to verify that the emergency diesel generator (EDG) features associated with the 86R, 86B, and 86F lockout relays prevent EDG starting only when required. The licensee failed, in several cases, to test that the actual lockout features (i.e., lockout relay inputs) tripped the specific lockout relays as specified in the TS.

This issue was more than minor because a TS required test was not performed within the required periodicity (Question 1.C in Appendix E of NRC Manual Chapter 0612). There was no actual loss of the safety system function, and subsequent testing indicated that the lockout features would have been able to accomplish their design safety functions. Enforcement action for this apparent violation will be handled by separate correspondence at a later date. (Section 1R21.1/AV 50-354/2003-002-01)

Green A non-cited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III (Design Control) was identified for inadequate acceptance limits for HPCI quarterly operability surveillance testing requirements for developed pump flow. The licensee calculations that established the required test pressure and flows for the quarterly operability test were found to be non-conservative and no calculation was done to ensure that the system could meet design requirements.

This issue was more than minor because applying the non-conservative or unreviewed acceptance limits for the pump operability test did not assure the availability and reliability of the HPCI system. This issue is considered a very low safety significance finding, because while established acceptance limits may not have been correct, there was no loss of safety function. (Section 1R21.2.1/NCV 50-354/2003-002-04)

Green The team identified an NCV of 10 CFR Part 50, Appendix B, Criterion XVI (Corrective Actions), for not assuring that conditions adverse to quality concerning the high pressure coolant injection (HPCI) system lubricating oil (LO) temperature monitoring were promptly identified and corrected. The temperature alarm actuated during observed inservice testing after which the team identified several deficiencies with plant drawings and procedures and at least six notifications related to uncalibrated instruments, high temperature alarms, and defective temperature switches entered into the corrective action program during the past two years.

The finding was more than minor because the licensee failed to provide reliable indication in the control room potentially affecting the ability to monitor and assess equipment performance, which could affect the availability and reliability of HPCI. The issue was considered to be of very low safety significance because there was no loss of safety function and the actual oil temperature was below the technical manual temperature limit. (Section 1R21.2.3/ NCV 50-354/2003-002-06)

Green The team identified a TS violation dispositioned as an NCV for failure to demonstrate the HPCI system operability by, at least once per 31 days, verifying that each valve, manual or automatic, in the system flow path that is not locked, sealed or otherwise secured in position is in its correct position. The team identified that manual valve BJ-048 was not accounted for in the HPCI system valve lineup.

The finding is more than minor because a TS required valve position verification was not performed (Question 1.c in Appendix E Manual Chapter 0612), which had the potential to impact HPCI availability and reliability in reference to the configuration control attribute for operating equipment. Mis-positioning of this valve could result in damage due to inadequate LO cooling. The risk of this finding is determined to be of very low safety significance because there was no loss of safety function, and the valve was found to be in the proper positions during a subsequent valve line-up. (Section 1R21.2.4/ NCV 50-354/2003-002-07)

Green An NCV of 10CFR50.63 (a) (2) Station Blackout (SBO) was identified due to the lack of an operability determination or engineering evaluation for the multiple steam leaks in the HPCI pump room. The degraded plant material condition of elevated HPCI room temperatures and humidity were not evaluated for the direct impact on the station's ability to cope following an SBO.

The finding was more than minor because, for an SBO event, both the expected HPCI pump room temperatures and HPCI DC bus room temperatures would be above the evaluated temperature limits potentially affecting the availability and reliability of HPCI. The finding is of very low safety significance because there was no loss of safety function. Draft calculations and subsequent engineering review of the conservatism in the original calculation method provided evidence that the resulting elevated room temperatures were not likely to cause a short term failure of HPCI or a risk significant failure on the DC bus components during the four-hour coping period. (Section 1R21.2.5/ NCV 50-354/2003-002-08)

Green The team identified a finding concerning inadequate modification and system design controls with respect to the station anticipated transient without scram (ATWS) evaluation. The team identified two modifications as well as other configuration differences that could change the results for the ATWS evaluation of record.

There was no violation of NRC requirements because the licensee did evaluate design basis events such as loss of offsite power (LOP) and loss of coolant accident (LOCA) to verify design inputs and limits for the modifications. The finding is greater than minor because the condition if left uncorrected had the potential to affect the availability and reliability of HPCI. The finding is of very low safety significance because there was no loss of safety function and the reconciled and corrected ATWS evaluation demonstrated that suppression pool temperature remains below design limits and would provide adequate net positive suction head (NPSH) and LO cooling.
(Section 1R21.2.6/ FIN 50-354/2003-002-09)

REPORT DETAILS

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R21 Safety System Design and Performance Capability (71111.21)

a. Inspection Scope

The team reviewed the design and performance capability of: the electrical power system, including the onsite emergency diesel generators (EDGs) and offsite electrical power system, and the high pressure coolant injection (HPCI) system. From a risk perspective, the team focused inspection activities on components and procedures that would minimize a loss of offsite power (LOP) initiating event and mitigate the associated accident sequences.

Electrical Power System:

Electrical power to safety-related equipment is supplied by four vital electrical busses (vital busses). Each bus has two offsite power supplies, one each from the two station service transformers (SSTs) supplied from the 13.8KV ring bus, and a backup power supply from an EDG if offsite power was unavailable. Normally, offsite power is supplied to two of the four vital busses by one SST and the other two by the other SST. If the normal offsite source to a vital bus is lost or in a degraded voltage condition the bus will be fast transferred to the other SST source. These are required power sources per 10CFR50 Appendix A (general design criteria), Criterion 17, Electrical Power Systems.

Maintaining the electrical power to the vital busses is important since they supply the safety-related systems needed to respond to plant conditions. A LOP is an initiating event, which results in a reactor scram and the need for EDGs to start and power the vital busses. The EDG system has been credited as a mitigating system for several design basis events including LOP and the design basis loss of coolant accidents (LOCA).

The power supplying the SSTs was designed with two independent offsite power sources from the transmission network by two physically independent circuits. The offsite power lines feed two separate 500 KV buses which then feed the 13.8 KV yard ring bus via four step-down station power transformers (T1, T2, T3, and T4). The ring bus is separated so that power transformer T1 &T4 supply half of the ring bus while T2 and T3 supply the other half; with each half of the ring bus supplying one of the two SSTs.

The four EDGs fulfill the requirement for onsite standby alternating current (AC) power to each of the four vital buses. The mechanical and electrical systems are designed so that a single failure affects the operation of only one EDG. Each equipment train consists of a 12 cylinder diesel engine directly coupled to a vital AC electric generator.

Important auxiliary and support equipment for the electrical power system includes: the electrical relaying needed to monitor the offsite power supply voltage and vital bus voltage, and the associated transfer circuits, to minimize the frequency of a LOP and ensure operability of safety-related equipment; the EDG important skid mounted/auxiliary equipment including the EDG air starting, fuel oil, jacket water cooling, lubricating oil, AC and direct current (DC) power; the safety auxiliary cooling system (SACS) which cools the lubricating oil, jacket water, and combustion air; room ventilation; and circuit breakers, relaying, and control power needed to coordinate the operation of safety-related equipment.

<u>High Pressure Coolant Injection System (HPCI)</u>

HPCI is an emergency core cooling system (ECCS) primarily designed to maintain the reactor vessel inventory after a small break LOCA that does not depressurize the reactor vessel. The system also maintains the reactor vessel inventory following a reactor isolation and coincident failure of the non-ECCS reactor core isolation cooling (RCIC) system. The HPCI system has been credited as a mitigating system for all initiating events except a large LOCA, including: LOP, small and medium LOCAs, and for the beyond design basis events of SBO and anticipated transients without scram (ATWS).

The system consists of a booster and main centrifugal pump powered by a common (through a speed changer) steam turbine, and the necessary valving and instrumentation. Steam to the turbine is supplied from the reactor vessel upstream of the main steam isolation valves and exhausts below the water level in the suppression pool (torus). The pump takes a suction from either the condensate storage tank (CST) or suppression pool and discharges into the reactor vessel through both the feedwater sparger and core spray sparger. The system is initially aligned to the CST and will automatically transfer to the suppression pool when a high suppression pool water level alarm is actuated. The turbine/pump lubrication and control, and the speed changer lubrication are provided by a skid mounted lubricating oil (LO) system. The LO is cooled by water taken from the discharge and returned to the suction of the booster pump. Additionally, the system has an automatic flow controller to maintain flow into the reactor vessel over a wide range of pressures during an event.

Important auxiliary and support equipment includes: skid mounted LO system and associated cooling system, which provides control oil, bearing cooling oil and speed changer lubrication; DC power from the “A” 250Vdc battery bus, providing control power and power to operate valves and the auxiliary LO pump; SACS and AC power which provides cooling water and operate the fan motors for the HPCI pump room coolers; and the residual heat removal (RHR) system in the suppression pool cooling mode to support HPCI net positive suction head (NPSH) and LO temperature requirements.

For the electrical power and HPCI systems, the team verified that: (1) the system design bases were in accordance with the licensing commitments and regulatory requirements; and (2) the design documents, such as drawings and design calculations, were correct. The documents reviewed included engineering analyses, calculations, plant modifications, piping and instrumentation drawings (P&IDs), electrical schematics, instrumentation and control drawings, logic diagrams, and instrument setpoint documentation. The team interviewed site personnel including; operators, system engineers, design engineers, and work management personnel regarding the operation and performance health of the HPCI and EDG support systems.

The team reviewed the operating procedures and engineering design calculations for the HPCI and EDG support systems in order to verify that procedure actions match design analysis assumptions. The types of procedures reviewed included: system operating procedures, abnormal and emergency operating procedures (AOPs and EOPs), alarm responses, and engineering design control procedures.

A walkdown of accessible portions of the HPCI, EDG, and vital switchgear systems was performed to verify the physical installation of the system and to verify consistency with design documents, calculations, assumptions, and installation specifications. The team used the updated final safety analysis report (UFSAR), TS, piping and instrument diagrams (P&IDs), and isometric drawings as references to verify the physical installation was consistent with design bases assumptions for major components, including piping, piping supports, pumps, turbine, valves, generator and circuit breakers. During field walkdowns, the team examined the material condition of support systems, and physical line-up of major components, including pumps, valves, piping, supports, heat exchangers, instrumentation, and breakers. The team also walked down supporting systems including SACS and DC power.

The mechanical design review focused on the capability of the HPCI and EDGs systems including associated supporting pumps, piping, and valves under the design basis and transient conditions. Additionally, the current performance and test criteria for the HPCI pump and EDGs were reviewed to ensure consistency between allowable component performance and minimum allowable capabilities assumed in the accident analyses and associated design basis calculations.

The electrical design review focused on the capability of the offsite and onsite electrical power sources to supply the vital switchgear and the ability of associated actuation, control, and instrumentation systems to support the design basis. The team reviewed one-line diagrams, elementary diagrams, control schematics, calculations of equipment loading and fast transfer capability, load flow diagrams, and protective device setpoints. This included a review of related operating instructions, and surveillance and test procedures.

The team assessed the reliability and unavailability performance of the HPCI system and the EDGs by reviewing selected corrective and preventive maintenance work orders (WOs) over the past two years. The team also used the Maintenance Rule Program Quarterly Reports, system health reports, and discussions with the system engineers to review system reliability and availability. The team reviewed post-maintenance testing results for various WOs to verify the demonstrated capability of the components to perform their intended safety function.

The team reviewed HPCI and EDG TS required performance data acquired during surveillance testing (ST) activities to verify that the results demonstrated the functional capability and met the acceptance criteria. Selected component performance data was reviewed to verify that test results reflected design conditions. The team witnessed the performance of the HPCI quarterly full flow ST from the field and assessed test data to verify the functional capability and operational readiness of the system. The team also observed portions of the monthly EDG STs from both the control room and the field. Test acceptance criteria were reviewed and compared to design calculations, TS requirements, and the American Society of Mechanical Engineers (ASME) code, Section XI, inservice testing (IST) requirements. Surveillance test acceptance criteria, component online performance data, and chemistry sample results were compared with design limits to determine if the design margins were being maintained and components were properly monitored.

The team selected a sample of notifications associated with the HPCI system and the EDGs and the selected support systems including notifications written on unsatisfactory surveillance test results, to verify the licensee was identifying and correcting design issues at an appropriate threshold, and taking appropriate corrective actions to prevent recurrence after entering them in the corrective action program. The review included deficiencies associated with normal operations, and testing and maintenance activities.

The team reviewed operator actions in normal procedures and EOPs for operating, monitoring, and controlling the HPCI and electrical power systems. This included a review of the adequacy of HPCI suction sources including the CST and the suppression pool relative to suction temperature, and the swapover to the CST suction if suppression pool temperature increased due to LOP, SBO or ATWS conditions. The team verified that normal, abnormal, and EOPs were consistent with systems design bases. System interfaces (instrumentation, controls, and alarms) were reviewed to assess the support to operator decision making. The team also reviewed the ability to respond to anomalous conditions and complete recovery activities.

The team also reviewed significant modifications to HPCI and EDG support systems as well as significant changes to the license, TSs or plant design that could impact the functionality or reliability of HPCI or the EDGs. The Hope Creek license amendment to increase ultimate heat sink temperature (TS Amendment No. 120) and the change to a mixed vendor supplied reactor fuel load were changes reviewed in detail with respect to HPCI and EDG support system impact.

The team observed system environmental conditions during normal operation to verify plant conditions were bounded by the equipment qualification assumptions. The team reviewed recent plant experience with HPCI supply steam leaks to determine the plant impact with regard to risk significant functions and equipment reliability during risk significant accident sequences. Administrative controls on temporary modifications and on water tight doors and fire barriers were reviewed to assure physical system protection was maintained for external events such as earthquake, fire and flooding as described in design documentation.

b. Findings

.1 Electrical Power System

.1.1 Failure to Properly Implement Technical Specification Surveillance Requirement for Emergency Diesel Generator Lock-out Features

Introduction: The licensee failed to adequately verify that the EDG lockout features prevented EDG starting only when required as per TS 4.8.1.1.2.h.14 (a, b, and c). Subsequent testing verified that the EDG lockout features and the associated EDGs were operable and were capable of performing their intended function. Enforcement action for this apparent violation of technical specifications will be handled under separate correspondence.

Description: TS 4.8.1.1.2.h.14 (a, b, and c) require, at least once per 18 months, the verification that the EDG lockout features associated with the regular lockout, backup lockout, and breaker failure lockout relays (86R, 86B, and 86F, respectively) prevent EDG starting only when required. The specified features included: engine overspeed, generator differential, and low lube oil pressure associated with the 86R relay; backup generator differential, and generator overcurrent associated with the 86B relay; and generator ground, and lockout relays - regular, backup, and test associated with the 86F relay.

While reviewing surveillance procedures HC.OP-ST.KJ-0005(Q) through -0008(Q), the team noted that each of the three relays was independently tripped and verified to not allow the associated EDG to start during a manual start attempt. The team questioned if each of the individual features were tested to verify that they tripped the associated lockout relays. (i.e., if the inputs to each of the three relays (for all four EDGs) were verified to trip the lockout relay.)

In reviewing this issue, the licensee confirmed that some portions of the lockout relay inputs had been tested within the required 18-month frequency for all the EDGs. In particular, the engine overspeed input to the 86R relay had been satisfactorily tested in accordance with surveillance procedure HC.MD-ST.KJ-0001(Q), “Diesel Generator Technical Specification Surveillance and PM.” The remaining inputs were not tested completely; for example, the generator differential input to 86R had been tested and was within the 18 month frequency for two of the four EDGs; and there was no recent testing associated with the low lube oil pressure input to 86R. This appeared to be due to inappropriate changes made to preventive maintenance task frequencies from an 18 month to a 36 month frequency.

Following this discovery, the licensee declared all four EDGs inoperable at 1:07 p.m. on December 12, 2002, and entered the provisions of TS 4.0.3, which allowed 24 hours to complete the missed testing and restore compliance with TS. On the morning of December 13, 2002, the team recognized that the testing focused on the 86R relay inputs only. The team re-stated the concern that all three relays (86R, 86B, and 86F) appeared to have been inadequately tested. In response, the licensee reentered the provisions of TS 4.0.3 for the purpose of properly testing the 86B and 86F lockout relays on the EDGs where it had not been completed in the required frequency.

The team returned to the facility on December 16, 2002, to review the results of the licensee’s completed testing. The team found that the testing associated with the 86F lockout relay for the A and C EDGs was inadequate in that it tested only one of the four inputs (the 86B -backup - relay input). The team informed the licensee of the apparent inadequacy of the testing at the exit meeting on December 16, 2002.

On December 18, 2002, the NRC and the licensee discussed this issue via tele-conference. The licensee stated a revised position where they believed that the features of the individual lockout relay did not have to be tested. Rather, their testing of the relays and verification that the associated EDG would not start was an acceptable test to implement the requirements of the TS. The NRC identified and presented to the licensee existing guidance on this issue (NRC Inspection Manual, Part 9900 - Technical Guidance - Standard Technical Specifications). This technical guidance stated that the individual features needed to be tested to verify that they would prevent starting the EDG only when required and that conformance to TS requirements was not subject to the interpretations with regard to intent by subsequent change to the standard TSs. Based upon review of the specific TS, discussion with NRR personnel, and review of the technical guidance, the team concluded that the licensee did not comply with TS 4.8.1.1.2.h.14.c. Subsequent to this discussion, the licensee conducted additional testing to comply with TS 4.8.1.1.2.h.14.c. On December 19, 2002, the licensee stated that they satisfactorily completed all testing for the lockout relay features (inputs) listed in TS 4.8.1.1.2.h.14 (a, b, and c).

Analysis: The finding is more than minor because the TS required ST had not been performed within the required periodicity (Question 1.c in Appendix E of NRC Manual Chapter 0612). The condition could have affected the ability, reliability, and capability of the EDG system, which is a system designed to mitigate the consequences of postulated accidents. Specifically, the team assessed that the requirement to verify that the feature lock-out the EDG “only when required” intended to maximize EDG availability while ensuring an EDG lock-out and the associated operator manual actions to clear a lock-out, following a significant failure as determined by the specified features. However, there was no actual loss of the safety system function, as subsequent testing indicated that the lockout features would have been able to accomplish their design safety functions.

Enforcement: Technical Specification 4.8.1.1.2.h.14 requires that, once per 18-months, the licensee verify that the following diesel generator lockout features prevent diesel generator starting only when required:

a) Engine overspeed, generator differential, and low lube oil pressure (regular lockout relay, (1) 86R);
b) Backup generator differential and generator overcurrent (backup lockout relay, (1) 86B);
c) Generator ground and lockout relays - regular, backup and test, energized (breaker failure lockout relay, (1) 86F).

The team identified that the licensee failed to verify that all of the features described above would prevent diesel generator starting only when required within the 18 month frequency. Enforcement action for this apparent TS violation will be handled under separate correspondence at a later date. (AV 50-354/2003-002-01, Inadequate EDG Lockout Relay Testing Frequency)

.1.2 Degraded Voltage Time Delay Setting

Introduction: The team questioned the adequacy of the time delay setting of the offsite power undervoltage relays. Specifically the team was concerned that the existing time delay of 20 +15/-5 seconds (TS Table 3.3.3-2) from the detection of a sustained degraded voltage condition until the vital busses were transferred to the EDGs was longer than time allowed by the 10CRF50.46 Loss of Coolant Accident analysis sequential loading time of 15 seconds following receipt of a LOCA. This is an unresolved item pending further NRC review and evaluation of the licensee position to determine the adequacy of the existing setpoint.

Description: The team referenced Branch Technical Position PSB-1 Section B.1 which states that a second level of undervoltage protection should be provided with two separate time delays, the first time delay would be of short duration (i.e., longer than a motor starting transient), with a subsequent LOCA signal causing separation from the offsite source. The team believed that the degraded voltage scheme should be suitable to protect safety-related equipment if a LOCA initiated at the same time that a degraded voltage condition existed. In addition, the team reviewed an NRC letter dated June 2, 1977 (sent to all operating plant at that time) which stated that the allowable time delay for the degraded voltage protection scheme, including margin, "shall not exceed the maximum time delay that is assumed in the UFSAR accident analysis."

The licensee was unable to show that during a LOCA with degraded voltage the 15 second time delay limit cited for the availability of power from the diesel generators could be met. During this delay ECCS pumps may fail to start and MOVs may fail to move to their required positions.

The licensee's position was that the design basis of the plant was a LOCA concurrent with a LOP and that the assumption that a LOCA occurred during a degraded voltage condition or that LOCA loading of offsite power which leads to a degraded condition were outside the plant design basis as described in UFSAR Table 6.3-1 which describes a simultaneous LOP and LOCA.

The team believed that applying a potentially non-conservative acceptance limit for the time delay relay did not assure the availability of the vital buses. The undervoltage relay time delay setpoint requirements, to assure compliance with 10CFR50 General Design Criterion 17, is unresolved pending appropriate evaluations and resolution of the design and licensing basis by the NRC: URI 50-354/2003-002-02, Degraded Grid Time Delay Relay Setting Relative to LOCA Analysis Assumptions.

.1.3 Grid Separation Vulnerability

Introduction: The team identified an issue related to the technical specification requirement for the availability of two independent offsite power sources for the site. The licensee had not evaluated the bus voltage relay reset setpoints of the degraded voltage relays to include the effects of voltage transients, from fast transfer of buses or unit trips, to assure the design would prevent grid separation during these transients. This is an unresolved item pending further licencee calculation and analysis, and NRC review, to determine the adequacy of the existing undervoltage relay reset setpoints.

Description: The team questioned the adequacy of the reset setpoint of the degraded voltage relays and the minimum voltage required to maintain power to the vital safety buses following relay dropout. The relays are likely to drop out during certain plant transients, therefore it is important to set the relay reset setpoint as low as practical to minimize the likelihood of grid separation. The team found that TS Table 3.3.3-2 specified a trip (dropout) setpoint of greater than 110.2 volts with an allowable as-found value of greater than 109.0 volts; but a reset setpoint was not specified. The licensee also provided relay calibration procedure HC.MD-ST.PB-0014(Q) which specified the pickup (reset) setting upper limit of 115 volts.

The licensee stated that they had no analysis that evaluated the reset setting relative to design requirements of the electrical distribution system. Additionally, the licensee could not demonstrate the adequacy of the degraded voltage relay As-Left Trip setpoint of greater than 110.2 volts relative to the minimum allowable value of greater than 109.0 volts. The licensee did not account for relay or test equipment accuracy effects that could result in a lower bus voltage than analyzed and voltage calculations provided to the team used the as found setting for the degraded voltage relays as the minimum voltage for analysis.

The team reviewed a limited sample of recent calibration results for the degraded voltage relays to determine the actual field setpoints. It found that the average as-left reset setting of 112.03 volts. Although less than the procedure required setpoint of 115 volts it was approximately 1% higher than allowed by equipment capabilities. Consequently, the team concluded that the actual field reset setpoints were higher than necessary and questioned if both the allowable and actual setpoints would result in spurious grid separation.

While no specific calculations were presented, the team reviewed several other calculations and completed independent calculations indicating that the loss of one offsite source with maximum LOCA loading would very likely cause the loss of the remaining offsite power source even considering the non-conservative assumptions of maximum bus voltage and average relay setpoints. Specifically:

- Calculation E15-5 which analyzed the adequacy of sequencing control relays during a fast transfer, showed that during a fast transfer with LOCA loading, voltage first dips and then recovers to approximately 91% of its pre transfer voltage within 1 second. The team determined that the voltage dip was of sufficient magnitude and duration to trip the degraded voltage relays.

- Voltage transient calculation for normal loading during power operation showed a voltage drop of approximately 7.6% following a fast transfer.

- The licensee's electrical distribution voltage calculations did not consider the step voltage decrease that could occur upon a trip of the unit generator combined with the expected voltage dips on the safety buses resulting from the starting of large motors.

- No analysis was available for the most limiting case where all safety busses are supplied from a single offsite source during LOCA sequencing. The latest revision of the bus voltage calculation (E-15(Q), Revision 6) omitted the alternate alignment where all safety buses were supplied from a single source.

The team concluded that, with normal loading and the allowable relay setpoint ranges, loss of one offsite source could cause loss of the remaining source because it appeared that the voltage on the remaining source would not recover to the reset setpoint in order to prevent separation.

In response to the team's concern, the licensee:

- Stated that the loading used in Calculation E-15.5 was too high and that an alternative calculation would be provided that would show adequate system performance.

- Identified a case involving the outage of a 500KV transmission line and high VAR output by the station. The calculated voltage drop after a station trip indicated a step voltage decrease of 2.6% on the 500KV system feeding the switchyard and then the SSTs.

- Wrote notification 20123613 to address this omission and cited Calculation E-15(Q) Revision 5 as showing acceptable results. However, Revision 5 took credit for automatic load tap changer operation during LOCA sequencing. The team found that the load tap changer parameters, including target voltages and time delays, were not controlled or retrievable by the station, and have not been verified by periodic test since installation of the equipment several years ago. The licensee concluded that they could not take credit for automatic operation of the tap changers during transients to maintain offsite power. The team, consequently, found that the justification to use the previous revision of the calculation was inappropriate.

The team believed that applying a potentially non-conservative acceptance limit for the degraded voltage relay reset did not assure the availability of two independent offsite power supplies. This issue is unresolved pending appropriate evaluations and resolution of the adequacy of the undervoltage dropout and reset setpoints by the licensee and review by the NRC: URI 50-354/2003-002-03, Offsite Power Grid Separation Vulnerability - Adequacy of Calculation Assumptions.

.2 <u>High Pressure Coolant Injection</u>

.2.1 Inservice Testing Operability Limits

Introduction: The team identified an NCV of very low safety significance (Green) regarding the licensee's failure to ensure adequate limits for quarterly operability surveillance testing of HPCI concerning developed pump flow .

Description: The team identified several non-conservative assumptions while reviewing design calculation BJ-23 Revision 0 to verify that the appropriate limits had been established to ensure that the HPCI pump could meet its design requirements. Specifically the calculation did not:

- Account for errors in measuring the required 5600 gpm.

- Include the pressure drop due to the feed water sparger and core spray sparger nozzles in system head loss determination.

- Use the design basis pressure of 1135 psig (relief valve setpoint). The calculation was based on a reactor operating pressure of 1000 psig. The licensee did not have a calculation to show, based on the operating conditions of the test, the pump and turbine would be able to meet the actual design limits.

Analysis: The issue affected the objective of the mitigating systems cornerstone and was more than minor because applying non-conservative or unreviewed acceptance limit for the pump operability test did not assure the availability and reliability of HPCI to prevent core damage. The finding screened to very low safety significance finding (Green) in SDP Phase I, because, while established acceptance limits may not have been correct, there was no loss of safety function.

Enforcement: 10 CFR Part 50, Appendix B, Criterion XII requires, in part, that measures shall be established to assure the design basis are correctly translated into specifications, procedures and instructions and that a test program shall be established to assure that all testing required to demonstrate that systems will perform satisfactorily is performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in design documents. Contrary to this requirement, for the HPCI system: the calculation to establish the required flow, speed and pressure was non-conservative. However, because of the very low safety significance and because the issue was entered into the licensee's corrective action program (Notifications 20122599 and 20123601) it is being treated as a non-cited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy:
NCV 50-354/2003-002-04, Inadequate IST Acceptable Criteria for HPCI Developed Pump Flow.

.2.2 Adequacy of HPCI Suction Sources Relative to Lubricating Oil (LO) Temperature

Introduction: The team identified an unresolved HPCI operability issue concerning the LO temperature requirements and the procedures controlling the HPCI suction sources. The issue is unresolved pending licensee completion of additional analysis and potential procedure changes and subsequent NRC review. The two related issues are described separately below:

Description:

- With respect to the LO temperature requirements the team found that:

    - There was no technically supported basis for the high temperature limit for the LO cooler outlet temperature (LO supply temperature) of 140 degrees Fahrenheit (°F) in the Quarterly IST. This acceptance criteria was used as an IST quarterly operability verification for check valves on the piping that supplied cooling water to the LO cooler.

    - The EPRI guidance for the equipment indicated that during an IST the LO temperature should never exceed 140 °F.

    - No calculation existed to ensure that the HPCI LO would be adequately cooled during design basis events that would cause the suppression pool temperature to increase.

    - During the April 2002 quarterly IST test, the LO supply temperature reached 143 °F while operating with CST suction at approximately 70 °F, and that there was no specific operability evaluation due to the temperature exceeding the setpoint.

    - The LO temperature was near 140 °F during the December 2002 quarterly test with approximately 70 °F CST suction water, based on team observations.

    - The TS operability requirements assume that HPCI takes a suction from the suppression pool; the CST volume was not credited toward operability. No specific analysis related the CST required volume of 135,000 gallons to HPCI functionality, in particular the LO cooling requirement during design basis accidents or an SBO or ATWS condition.

    - The licensee stated that the actual limit for the equipment was 155 °F, based on the equipment technical manual. The licensee was not able to show that the HPCI LO temperature would not exceed the maximum allowed temperature of 155 °F during a design basis event while taking suction from the suppression pool.

- With respect to procedures, the team identified that several EOPs and at least one abnormal operating procedure (AOP) referred to using the CST as the preferred suction for the HPCI pump. For example:

    - HC.OP-EO.ZZ-0101, “Reactor/Pressure Vessel (RPV) Control,” stated that the CST was the preferred injection source, if available, for HPCI; if necessary, bypass the high suppression pool level suction transfer interlock.

- HC.OP-AB.ZZ-0135, “Station Blackout / Loss of Offsite Power / Diesel Generator Malfunction,” stated that HPCI can take suction from the CST or suppression pool. Heated suppression pool water should only be used if CST water was unavailable and sufficient net positive suction head (NPSH) was available.

The Boiling Water Reactor Owners Group basis document for the EOPs, with respect to the operation the HPCI pump, stated that the system’s LO and control oil are cooled by the water being pumped. Operation with high suction temperature may result in bearing damage or loss of control capability. The preferred suction source for HPCI is always the CST. While the CST is a smaller volume than the suppression pool, it provides higher quality water, and is not affected by containment heat-up or steam discharges from the RPV. NPSH and component cooling limitations are thus less likely to be challenged. The CST should be refilled as water is depleted to maintain the suction source.

Two operators were interviewed with respect to restoring the suction for the HPCI pump to the CST. Neither operator described the required steps correctly.

The team was concerned about system operability if the HPCI was required to operate with the suppression pool as the suction source during a design basis events. This also applied to system functionality for the beyond design basis events of SBO and ATWS.

- Suppression pool water temperature could itself exceed the 155 °F temperature during a design basis LOP and Medium LOCA and could be near 212 °F during SBO or ATWS conditions. It appeared based on the IST results, with the CST water at about 70 °F and LO temperature at or above 140 °F, that the 155°F could be exceed if suppression pool temperature reached 85°F.

- Neither the AOPs nor the EOPs provided the operators with the details on how to respond to high temperature conditions including how to return the suction to the CST. This operation was considered a “skill of the craft” evolution, meaning that the operators were expected to know the necessary steps and be able to perform the evolution without reference to the procedure. The HPCI system operating procedure did not provide procedural steps to return the suction to the CST, only precautionary information describing the affects of performing the evolution incorrectly.

- The team also noted that the TS for HPCI possibly needed clarification relative to the necessity of the CST as a suction source.

In response to these issues the licensee began a review of HPCI LO temperature requirements and the associated suction source control procedures. Further, the licensee took action to instruct their operators, through night orders, to ensure that the HPCI suction remains on the CST during events that cause suppression pool water temperature to increase.

The team determined that high LO temperatures caused by high suppression pool temperatures or unclear procedural guidance on the use of the CST as a suction source, potentially could affect the operability of HPCI. This issue was considered unresolved pending completion of a review of temperature and procedural issues by the licensee and subsequent review by the NRC: URI 50-354/2003-002-05, Adequacy of HPCI Suction Sources Relative to LO Temperature

.2.3 Inadequate Temperature Monitoring

Introduction: The team identified an NCV of very low safety significance (Green) regarding the licensee's failure to identify and correct problems with the HPCI LO temperature monitoring instrumentation.

Description: A control room HPCI Trouble Alarm annunciated during quarterly HPCI IST observed by the team on December 10, 2002. The alarm response procedure (ARP) stated that the local dial temperature indicating switch FDTIS-5770 should cause this alarm at 155 °F and that with this condition, the system should be shutdown. Local observation of FDTIS-5770 showed 139 °F, so the operators completed the test.

The team identified numerous problems with the HPCI LO temperature monitoring instrumentation including:

- The HPCI wiring diagram indicated that FDTIS-5770 or a temperature switch FDTS-4773 would cause the HPCI Trouble control room alarm, while the ARP only referred to FDTIS-5770.
- FDTIS-5770 alarm appeared to be set at about 140 °F, while the ARP stated that it was set at 155°F.
- FDTS-4773 was not referenced in the ARP or on the LO P&ID. The P&ID only showed FDTIS-5770 with an incorrect alarm setpoint of 155 °F.
- The licensee had previously determined that the LO thermocouple strip chart recorder, which indicated bearing temperatures, had not been calibrated since 1998 but had not placed tags on the instrument to warn operators. The team found that the thermocouples that provide input to the chart recorder had also not been calibrated since 1998 and had not been previously identified.
- Numerous notifications existed for related problems with the LO temperature switches or the alarm setpoints, which should have led to identifying and correcting the issues that the team identified. They included:
    - Notification 20030720 (May 2000) was written to identify that the HPCI oil temperature alarm (155 °F) was received although local indication was 124 °F and based on the local indication the HPCI test was continued.
    - Notification 20068602 (June 2001) requested faulty thermocouple FDTS-4773 or FDTIS-5770 switch be calibrated or replaced because they were actuating the HPCI Trouble overhead alarm in the control room. The

gauge and switch were found to be in calibration and no other action was taken.

- Notification 20089821 (January 2002) reported that FDTS-4773 had been damaged twice during system outage work and references Order 7002204 which requested protective covers be installed on FDTS-4773. Covers were never installed.
- Notification 20095546 (April 2002) stated that digital point D5434 (HPCI turbine oil cooler temperature Hi alarm, FDTIS-5770) was received and did not clear after the test. The notification does not address why the alarm occurred with LO temperature at 143 °F.
- Notification 20097307 (April 2002) reported that the problem with the alarm indication related to D5434 was temperature alarm input FDTS-4773. The switch was replaced during a system outage.
- Notification 20099517 (May 2002) reported that the system engineer observed that the LO temperature gauge (FDTIS-5770) read 120 °F while actual temperature was 110 °F. Additionally, the engineer observed that the alarm set on the gauge was 140 °F in contrast to the required alarm setpoint of 155 °F.

Analysis: This issue affected the objective of the mitigating systems cornerstone and was more than minor because the licensee failed to provide reliable indication in the control room potentially affecting the ability to monitor and assess equipment performance, which could affect the availability and reliability of HPCI. The finding screened to very low safety significance finding (Green) in SDP Phase I, because there was no loss of safety function and the actual oil temperature was below the technical manual temperature limit.

Enforcement: 10 CFR Part 50, Appendix B, Criterion XVI requires, in part, that conditions adverse to quality such as failures are promptly identified and corrected. Contrary to this requirement, the licensee failed to determine the cause of repeat alarms on the control room annunciator panel and failed to find and correct errors in several support documents and equipment. However, because of the very low safety significance and because the issue was entered into the licensee's corrective action program (Notification 20113475, 20124695, 20124310, 20124618, 20124351,20122451, 20124558) it is being treated as a non-cited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 50-354/2003-002-06, Failure to Identify and Correct Issues with the HPCI LO Temperature Monitoring Instrumentation.

.2.4 Incomplete Valve Position Verification

Introduction: The team identified an NCV of very low safety significance (Green) regarding the licensee's failure to properly conduct a TS Surveillance required monthly valve lineup in the HPCI flowpath, because one valve was not included.

Description: The team reviewed the valve lineup that was performed to satisfy the requirement of TS 4.5.1.a) 1.b, which requires that ECCS be demonstrated operable at least once per 31 days by verifying that each valve, manual or automatic, in the flow path that is not locked, sealed or otherwise secured in position is in its correct position. The team identified that manual valve BJ-048 was not locked in position nor included in the HPCI System Piping and Flow Path Verification - Monthly (HC.OP-ST.BJ-0001 Revision 9) as required. Valve BJ-048 controls the flow of cooling water through the HPCI LO cooler..

Analysis: This issue affected the objective of the mitigating systems cornerstone and was more than minor because a TS required valve position verification was not performed (Question 1.c. in Appendix E Manual Chapter 0612), which had the potential to impact HPCI availability and reliability in reference to the configuration control attribute for operating equipment. Mis-positioning of this valve could result in damage due to inadequate LO cooling. The finding screened to very low safety significance finding (Green) in SDP Phase I, because there was no loss of safety function, and the valve was found to be in the proper position during a subsequent valve line-up.

Enforcement: Contrary to TS 4.5.1 the licensee did not verify the correct valve alignment for HPCI. However, because of the very low safety significance and because the issue was entered into the licensee's corrective action program (Notifications 20122608 and 20124587) it is being treated as a non-cited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 50-354/2003-002-07, Incomplete HPCI TS Valve Line-up

.2.5 High HPCI Room Temperatures Relative to SBO Coping Analysis

Introduction: The team identified an NCV of very low safety significance (Green) regarding the licensee's failure to evaluate the risk significant impact of the five identified steam leaks in the HPCI room on the station's capability for coping with an SBO as required by 10CFR50.63

Description: The team identified that the operability determination and engineering assessment regarding five steam leaks identified in September 2001 did not determine what the direct effect of HPCI room elevated temperature would be on the station's ability to cope with an SBO. The licensee chose to defer maintenance to correct the steam leaks and operated the plant with the leaks until the summer of 2002. The team found that the room temperature of 108 °F exceeded the initial temperature of 93 °F assumed in the SBO four-hour coping analysis and no technical evaluation was performed with regard to the change in peak environmental temperatures following an SBO.

Operating procedures were found to contain steps to open the HPCI room compartment doors within the first 20 minutes of an SBO. This action had the undesirable aspect of connecting a room that would be expected to reach or exceed 173 °F to the enclosed HPCI DC bus room which was not evaluated and had an operating limit of 108 °F.

During the inspection, design engineering performed draft calculations to determine the impact of the five steam leaks on the station's SBO coping ability with respect to HPCI operation. Draft calculations showed that the steam leaks would not likely cause a short term failure of HPCI to operate; however, the higher initial starting temperature of the room equated to what would be the elimination of most conservatism in the SBO temperature calculation. The licensee also presented existing evaluations and information, indicating that the short term effects on the HPCI DC bus in less than four hours would be unlikely.

Analysis: This issue affected the objective of the mitigating systems cornerstone and was more than minor because, for an SBO event, both the expected HPCI pump room temperatures and HPCI DC bus room temperatures would be above the evaluated temperature limits and had the potential to affect the availability and reliability of HPCI. The finding screened to very low safety significance finding (Green) in SDP Phase I because there was no loss of safety function.

Enforcement: Contrary to 10 10CFR50.63(a)2 the licensee operated the facility outside of the valid coping analysis for an SBO condition. However, because of the very low safety significance and because the issue was entered into the licensee's corrective action program (Notifications 20123367, 20123979, and 20124587) it is being treated as a non-cited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 50-354/2003-002-08, Failure to Evaluate the Potential Effects of Steam Leaks on SBO Coping Assumptions.

.2.6 Incomplete ATWS Evaluation

Introduction: The team identified a finding of very low safety significance (Green) regarding the licensee's inadequate modification and system design controls with respect to the station ATWS evaluation. The team identified two modifications as well as other configuration differences that could change the results for the ATWS evaluation of record.

Description: The team found that design changes and other identified deviations from the generic ATWS evaluation had not been evaluated with regard to the total combined impact on suppression pool water temperatures. Specifically the modification issues not evaluated were:

- The change to core doppler and core average void coefficients which resulted from a mixed vendor fuel load.
- The change to ultimate heat sink temperatures.

The team noted several other configuration differences between the station and the generic evaluation including: HPCI flow capacity, RHR heat exchanger heat transfer coefficients, and boron injection delay, which were never reconciled or evaluated in an integrated fashion.

During the inspection the licensee reconciled the differences from the generic ATWS analysis and identified additional time critical manual actions of establishing two trains of RHR in the suppression pool cooling mode.

Prior to the modifications, the plant staff did perform engineering analysis and 10CFR 50.59 reviews with regard to the relevant design basis events (LOP and LOCA) to verify design inputs and limits through the modification process. Thus, there was no violation of NRC requirements. However this procedural/process error which allowed modification to the plant without expected design controls is more than minor because this condition if left uncorrected had the potential to render several risk significant systems inoperable for the ATWS event scenario. Performing design changes to the plant without reviewing the impact on the ATWS event sequence could eventually render the primary containment or ECCS injection pumps incapable of performing the risk significant functions credited in the licensee's station risk analysis.

Analysis: This issue affected the objective of the mitigating systems cornerstone and was more than minor because the condition if left uncorrected had the potential to affect the availability and reliability of HPCI. The finding screened to very low safety significance finding (Green) in SDP Phase I because there was no loss of safety function and the reconciled and corrected ATWS evaluation demonstrated that suppression pool temperature remains below design limits and would provide low pressure injection pump net positive suction head (NPSH) to assure accident recovery capability. The licensee entered this issue into the corrective action program (Notification 20124546.) No violation of regulatory requirements occurred:
FIN 50-354/2003-002-09, Inadequate Modification and System Design Controls with Respect to the Station ATWS Evaluation

4OA2 Identification and Resolution of Problems

a. Inspection Scope

The team reviewed a sample of notifications associated with the HPCI and emergency electrical power systems, as identified in Attachment 1, to verify the licensee was identifying issues at an appropriate threshold, entering them in the corrective action program, and taking appropriate corrective actions.

b. Findings

The several of the findings described in the above sections of this report indicated a weak identification and corrective action process implementation. Specifically noted were incomplete and untimely corrective actions: to restore TS compliance concerning the EDG lockout relay feature testing, and to address issues with the important temperature monitoring instrumentation on the HPCI LO system.

4OA6 Meetings, Including Exit

.1 Management Meeting

The team presented the inspection results to Mr. D. Garchow and other members of the licensee's staff at an exit meeting on December 16, 2002. The team verified that the inspection report does not contain proprietary information.

Subsequent to the exit meeting on December 18, team members discussed the testing of the EDG lockout relays with Mr. G. Salamon, Manager - Licensing, and other members of the licensee staff. During the discussion the licensee agreed to complete testing to comply with TS 4.8.1.1.2.h.14 (a, b, and c). This discussion is incorporated in Section 1R21.1 of this report.

ATTACHMENT 1
SUPPLEMENTAL INFORMATION

a. Key Points of Contact

B. Berg, Operations
M. Pfizenmeier, Reliability Engineering
G. Salamon, Nuclear Safety and Licensing
G. Daves, Production Engineer
P. Walsh, Reliability Engineering
J. O'Connor, Design Engineering
D. Notigan, Engineering Supervisor, Reactor and Safety Analysis Group
K. Buckwheat, Senior Engineer, Reactor and Safety Analysis Group
S. Afarian, SACS System Engineer
S. Peng, Reactor and Safety Analysis Group
G. Morrison, Design Engineering
K. Buddenbohn, Nuclear Safety and Licensing
H. Berrick, Nuclear Safety and Licensing
R. Burk, Work Week Manager
D. Lyons, Inservice Testing Engineer

b. List of Items Opened, Closed, and Discussed

Opened

AV 50-354/2003-002-01	Inadequate EDG Lockout Relay Testing Frequency
URI 50-354/2003-002-02	Degraded Grid Time Delay Relay Setting Relative to LOCA Analysis Assumptions
URI 50-354/2003-002-03	Offsite Power Grid Separation Vulnerability - Adequacy of Calculation Assumptions
URI 50-354/2003-002-05	Adequacy of HPCI Suction Sources Relative to LO Temperature

Opened and Closed

NCV 50-354/2003-002-04	Inadequate IST Acceptable Criteria for HPCI Developed Pump Flow
NCV 50-354/2003-002-06	Failure to Identify and Correct Issues with the HPCI LO Temperature Monitoring Instrumentation.
NCV 50-354/2003-002-07	Incomplete HPCI TS Valve Line-up
NCV 50-354/2003-002-08	Failure to Evaluate the Potential Effects of Steam Leaks on SBO Coping Assumptions.
FIN 50-354/2003-002-09	Inadequate Modification and System Design Controls with Respect to the Station ATWS Evaluation

Closed

None

c. List of Acronyms

AC	Alternating Current
AOP	Abnormal Operation Procedure
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
CFR	Code of Federal Regulations
CSR	Condensate Storage Tank
DC	Direct Current
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
HCGS	Hope Creek Generating Station
HPCI	High Pressure Coolant Injection
IST	Inservice Test
LO	Lubricating Oil
LOCA	Loss of Coolant Accident
LOP	Loss of Offsite Power
NCV	Non Cited Violation
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
P&ID	Piping and Instrumentation Drawing
PSEG	Public Service Electric Gas
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
SACS	Safety Auxiliaries Cooling System
SBO	Station Blackout
SST	Station Service Transformer
ST	Surveillance Test
TS	Technical Specification
UFSAR	Updated Final Safety Evaluation Report
VAC	Volts Alternating Current
VDC	Volts Direct Current
WO	Work Order

d. List of Documents Reviewed

Procedures

HC.EP-AM.ZZ-0001, (SAG-2) Containment & Radioactivity Release Control (SAG), Revision 0
HC.EP-AM.ZZ-0001, (SAG-1) Primary Containment Flooding, Revision 0
HC.EP-AM.ZZ-0001, (EOP/SAG) RPV & Containment Information, Revision 0
HC.EP-AM.ZZ-0001, Technical Support Time Charts (SAG), Revision 0
HC.EP-AM.ZZ-0001, (TSG-1) Technical Support Guidelines (SAG), Revision 0
HC.IC-FT.PE-0005(Q), Functional Test /Time Interval Test Emergency Load Sequencer System Diesel Generator A, 1AC428, Revision 4
HC.IC-PM.KJ-0005(Q), Preventive Maintenance of Emergency Diesel Generators Neutral Grounding Transformers, Revision 1
HC.MD-CM.KJ-0001(Q), Diesel Engine Overhaul, Revision 12
HC.MD-ST.KJ-0001(Q), Diesel Generator Technical Specification Surveillance and PM Rev. 28

HC.MD-ST.PB-0014(Q), Class 1E 4.16 KV Degraded Voltage 18 Month Instrument Channel Calibration & Functional Test 10-A-40101, Revision 1
HC.MD-ST.PB-0021(Q), Class 1E 4.16 KV Degraded Voltage 18 Month Instrument Channel Calibration & Functional Test 10-A-40408, Revision 1
HC.OP-AB.COOL-0002(Q), Safety/Turbine Auxiliaries Cooling System, Revision 0
HC.OP-AB.COOL-0004, Fuel Pool Cooling, Revision 0
HC.OP-AB.ZZ-0000, Reactor Scram, Revision 1
HC.OP-AB.ZZ-0001, Transient Plant Conditions, Revision 0
HC.OP-AB.ZZ-0135, Station Blackout // Loss of Offsite Power // Diesel Generator Malfunction, Revision 21
HC.OP-AB.ZZ-0136, Loss of 120 VAC Inverter, Revision 5
HC.OP-AB.ZZ-0149, 250 VDC System Malfunction, Revision 3
HC.OP-AB-ZZ-0322, Core Spray Injection Valve Override, Revision 1
HC.OP-AR.KJ-0001(Q), Diesel Generator Remote Generator Control Panel 1AC422, Revision 5
HC.OP-AR.KJ-0001(Q), Diesel Generator Remote Engine Control Panel 1AC423, Revision 14
HC.OP-AR.KJ-0002, Diesel Generator Remote Generator Control Panel 1AC422; alarm window D-2, Generator Neutral Overvoltage, Revision 5
HC.OP-AR.KJ-0002(Q), Diesel Generator Remote Generator Control Panel 1AC422, Revision 5
HC.OP-AR.ZZ-0006, Overhead Annunciator Window Box B-1, Revision 17
HC.OP-AR.ZZ-0017(Q), Overhead Annunciator Window Box E4, Revision 3
HC.OP-DL.ZZ-0006(Q), Log 6 Auxiliary Building Log, Revision 34
HC.OP-EO.ZZ-0101, EOP: RPV Control, Revision 8
HC.OP-EO.ZZ-0101-CONV EOP-0101, Conversion Document, Revision 4
HC.OP-EO.ZZ-0101A-CONV EOP-0101A Conversion Document, Revision 1
HC.OP-EO.ZZ-0101A, EOP: ATWS - RPV Control, Revision 1
HC.OP-EO.ZZ-0102, EOP: Primary Containment Control, Revision 10
HC.OP-EO.ZZ-0102-CONV EOP-0102 Conversion Document, Revision 5
HC.OP-EO.ZZ-0103/4-CONVEOP-0103/4 Conversion Document, Revision 4
HC.OP-EO.ZZ-0103/4, EOP: Reactor Building & Radiation Release Control, Revision 6
HC.OP-EO.ZZ-0202, EOP: Emergency RPV Depressurization, Revision 6
HC.OP-EO.ZZ-0202-CONV EOP-0202 Conversion Document, Revision 4
HC.OP-EO.ZZ-0206, EOP: Reactor Pressure Vessel Flooding, Revision 6
HC.OP-EO.ZZ-0206-CONV EOP-0206 Conversion Document, Revision 6
HC.OP-EO.ZZ-0312, Suppression Chamber Make-Up Using HPCI, Revision 3
HC.OP-EO.ZZ-0316, Suppression Chamber Reduction Using HPCI, Revision 5
HC.OP-EO.ZZ-LIMITS-CONVEOP Limit Curves & Cautions Conversion Document, Revision 2
HC.OP-EO.ZZ-PSTG, EOP: Plant Specific Technical Guidance, Revision 6
HC.OP-IO.ZZ-0008, Shutdown from Outside the Control Room, Revision 16
HC.OP-IS.BJ-0001(Q), HPCI Main and Booster Pump Set –0P204 and 0P217 – Inservice Test, Revision 39
HC.OP-IS.EG-0001(Q), A SACS Pump - AP210 – Inservice Test, Revision 24
HC.OP-IS.EG-0101(Q), SACS – Subsystem A valves Inservice Test, Revision 45
HC.OP-IS.JE-0001(Q), Diesel Fuel Oil Transfer Pump - AP401 - Inservice Test, Revision 18
HC.OP-SO.BC-0001, Residual Heat Removal System Operation, Revision 34
HC.OP-SO.BJ-0001, High Pressure Coolant Injection System Operation, Revision 20
HC.OP-SO.EA-0001(Q), Service Water System Operation, Revision 21

HC.OP-SO.EG-0001, Safety and Turbine Auxiliaries Cooling Water System Operation, Revision 29
HC.OP-SO.JE-0001(Q), Diesel Fuel Oil Storage & Transfer System Operation, Revision 18
HC.OP-SO.KJ-0001, Emergency Diesel Generators Operation, Revision 38
HC.OP-SO.KJ-0001(Q), Emergency Diesel Generators Operation, Revision 38
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NC.CC-AP.ZZ-0001(Q), Design Basis/Input for Engineering Changes, Revision 1
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HC.OP-ST.KJ-0004(Q)	Emergency Diesel Generator DG-400 Operability Test - Monthly (multiple recent performances)
HC.OP-ST.KJ-0005(Q)	Integrated Emergency Diesel Generator 1AG400 Test - 18 Months, Revision 18 (performed 10/23/01)
HC.OP-ST.KJ-0006(Q)	Integrated Emergency Diesel Generator 1BG400 Test - 18 Months, Revision 21 (performed 10/19/01)
HC.OP-ST.KJ-0007(Q)	Integrated Emergency Diesel Generator 1CG400 Test - 18 Months, Revision 20 (performed 10/25/01)
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20006549
20015246
20018126
20030024
20045257
20046560
20047762
20053150
20056987
20060204
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20063601
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20124695*

Drawings

01761776, Electrical Schematic - Remote Generator Control Panel (Sheets 10, 14)
E-0001-0(Q), HPGS Single Line Diagram Meter & Relay Diagrams,
E-0002-1(Q), HPGS Single Line Meter & Relay Diagram, Power System
E-0003-1(Q), HPGS Single Line Meter & Relay Diagram, Generator – Main Transformer
E-0004-1(Q), HPGS Single Line Meter & Relay Diagram, 7.2KV Station Power System
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E-0005-1(Q), Sh 1 of 2, HPGS Single Line Meter & Relay Diagram, 4.16KV Station Power System

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E-0006-1(Q), HPGS Single Line Meter & Relay Diagram, 4.16KV Station Power System
E-0007-1(Q), HPGS Single Line Diagram, Synchronizing
E-0008-1(Q), HPGS Single Line Meter & Relay Diagram, Diesel Generators
E-0009-1(Q), HPGS Single Line Meter & Relay Diagram, 125V DC System
E-0011-1(Q), HPGS Single Line Meter & Relay Diagram, 250V DC – Unit 1
E-0018-1, Single Line Meter & Relay Diagram, 480 Volt Class 1E Unit Substation. 10B410, 10B420, 10B430, 10B440, 10B450, 10B460, 10B470, 10B480 Unit 1
E-0084-0, Electrical Schematic Diagram Class 1E 4.16KV Sta Pwr Sys Swgr Diesel Gen Circuit Brkr (1)52-40107, Revision 12
E-0086-0, Electrical Schematic Diagram Class 1E 4.16KV Sta Pwr Sys Swgr Diesel Gen Circuit Brkr (1)52-40307, Revision 12
E-0106-0, Electrical Schematic Diagram Class 1E 4.16KV Station Power System Revision 14
E-0107-0, Electrical Schematic Diagram Diesel Generator Regular and Backup Lockout Relaying, Revision 10
E-0217-0, Electrical Schematic Diagram 4.16KV Circuit Breaker Control Safety Auxiliaries Cooling Pump, Revision 4
E-3060-0, Logic Diagram, Class 1E Station PWR SWGR-4.16KV System Main Circuit Breaker, Revision 16
E-3062-0, Logic Diagram, 4.16KV Class 1E Busses –Unit 1 Bus Diff, Overcurrent & Undervoltage Protection Revision 4,
E-6074-0, Electrical Schematic Diagram High Pressure Coolant Injection Turbine Aux Oil Pump Revision 7,
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E-6431-0, Electrical Schematic Diagram HPCI Pump Turbine ECCS Jockey Pump 1AP228 Revision 3,
E-6442-0, Electrical Schematic Diagram 4.16KV Circuit Breaker Control Core Spray Pumps Revision 6,
J-105-0, Logic Diagram Sequencer Fan-Out Index Sheet
J-107-0, Logic Diagram Emergency Load Sequencer Index Sheet
M-11-1(Q), Safety Auxiliaries Cooling Reactor Building P&ID
M-12-1(Q), Safety Auxiliaries Cooling Auxiliary Building P&ID
M-30-1(Q), Diesel Engine Auxiliary Systems intercooler and Jacket Water P&ID
761E270AC, Process Diagram, High Pressure Coolant Injection System, Revision 4
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E-0001-0, Single Line Electrical Diagram, Hope Creek Station, Revision 9
E-0006-1, Single Line Meter & Relay Diagram, 4.16KV Class 1E Power System, Revision 6
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M-30-1, Diesel Engine Auxiliary Systems, Starting Air & Lube Oil, Revision 12
M-41-1, Nuclear Boiler, Revision 32
M-51-1, Residual Heat Removal, Revision 28
M-52-1, Core Spray, Revision 20
M-55-1, High Pressure Coolant Injection, Revision 36
M-56-1, HPCI Pump Turbine, Revision 16
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PSEG Letter NLR-N93103, Unresolved Item 50-354/92-80-10
PSEG Letter NLR-N92094, Response to Notice of Deviations and Unresolved Items
PSEG Letter NLR-N92071, 10CFR21 Notification Degraded Grid Trip and Bus Transfer Scheme
PSEG Letter NLR-N92022, Response to Issues Identified During the EDSFI
PSEG Letter NLR-N97082, Response to TAC No. 64740
PSEG Letter NLR-N87019, Request for Amendment Facility Operating License
10855-D3.10, Design, Installation and Test Specification for Safety and Turbine Auxiliaries Cooling System
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Beloit Power Systems Drawing 01761770 Sh. 6, “Electrical Schematic Engine Control”, Revision 7
Beloit Power Systems Drawing 01761770 Sh. 7, “Electrical Schematic Engine Control”, Revision 7
EPRI Terry Turbine Maintenance & Troubleshooting Guide
BWOG RCIC/HPCI Committee Meeting Notes, February 7-9, 2001, Atlanta GA.

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50000069	50020375	50056434
50000074	50020461	60025265
50000083	50021347	60026509
50001837	50055205	80054655
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DEH-93-00158	Safety Evaluation