UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PA 19406

November 9, 2007

Mr. William Levis
President and Chief Nuclear Officer
PSEG Nuclear LLC
80 Park Plaza, T4B
Newark, NJ 07102

SUBJECT: HOPE CREEK GENERATING STATION - NRC PROBLEM IDENTIFICATION AND RESOLUTION INSPECTION REPORT 05000354/2007006

Dear Mr. Levis:

On September 28, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed a team inspection at your Hope Creek Generating Station. The enclosed inspection report documents the inspection results, which were discussed on September 28, 2007, with Mr. George Barnes and other members of your staff.

This inspection was an examination of activities conducted under your license as they relate to the identification and resolution of problems, and compliance with the Commission's rules and regulations and the conditions of your operating license. Within these areas, the inspection involved examination of selected procedures and representative records, observations of activities, and interviews with personnel.

Based on the samples selected for review, the inspectors concluded that overall, problems were properly identified, evaluated, and corrected. There were two Green findings identified during this inspection involving repetitive problems with a safety-related breaker and control room emergency filtration damper controller power supply. The two findings were determined to involve violations of NRC requirements. However, because each violation was of very low safety significance (Green) and because they were entered into your corrective action program, the NRC is treating these as Non-Cited Violations (NCVs), in accordance with Section VI.A of the NRC's Enforcement Policy. If you deny any of these NCVs, 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, 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 Inspector at the Hope Creek Nuclear Generating Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the 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 by Arthur L. Burritt For/

Mel Gray, Chief
Technical Support and Assessment Branch
Division of Reactor Projects

Docket Nos. 50-354
License Nos. NPF-57

Enclosure: Inspection Report 05000354/2007006
w/Attachment: Supplemental Information

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T. Joyce, Senior Vice President, Operations
R. Braun Site Vice President - Salem
G. Barnes, Site Vice President - Hope Creek
K. Chambliss, Director - Nuclear Oversight
J. Fricker, Vice President – Operations Support
G. Gellrich, Salem Plant Manager
J. Perry, Hope Creek Plant Manager
J. J. Keenan, General Solicitor, PSEG
M. Wetterhahn, Esquire, Winston and Strawn, LLP
L. A. Peterson, Chief of Police and Emergency Management Coordinator
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R. Pinney, Bureau of Nuclear Engineering, NJ Dept. of Environmental Protection
H. Otto, Ph.D., Administrator, Interagency Programs, DNREC Division of Water Resources
Consumer Advocate, Office of Consumer Advocate, Commonwealth of Pennsylvania
N. Cohen, Coordinator - Unplug Salem Campaign
E. Zobian, Coordinator - Jersey Shore Anti Nuclear Alliance

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DOCUMENT NAME: G:\DRP\BRANCH3\HOPE CREEK\Next IR findings & feeders\HC 2007 PI&R Inspection\HC PI&R Insp Report draft.wpd

SUNSI Review Complete: KMT1 (Reviewer's Initials)
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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	050000354
License No:	NPF-57
Report No:	05000354/2007006
Licensee:	PSEG Nuclear, LLC
Facility:	Hope Creek Generating Station
Location:	P.O. Box 236 Hancocks Bridge, NJ 08038
Dates:	September 17, 2007 - September 28, 2007
Lead Inspector:	S. Barber, Senior Project Engineer
Inspectors:	T. Wingfield, Resident Inspector C. Zoia, Senior Reactor Engineer D. Tifft, Reactor Engineer
State Observers:	E. Rosenfeld, NJ DEP J. Humphreys, NJ DEP
Approved By:	Mel Gray, Chief Technical Support and Assessment Branch Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000354/2007006; 09/17/2007 – 09/28/2007; Hope Creek Generating Station; Biennial Baseline Inspection of the Identification and Resolution of Problems (PI&R).

This inspection was performed by three regional inspectors and one resident inspector. Two finding of very low safety significance (Green) were identified during this inspection. The findings were classified as a Non-Cited Violation (NCV). The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter (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 4, dated December 2006.

Identification and Resolution of Problems

The inspectors concluded that the implementation of the corrective action program (CAP) at Hope Creek was effective. Hope Creek had a low threshold for identifying problems and entering them in the CAP. Once entered into the system, items were screened and prioritized in a timely manner using established criteria. Items entered into the CAP were properly evaluated commensurate with their safety significance. In general, corrective actions were implemented in an effective manner. PSEG's audits and assessments were generally thorough and probing. The inspectors concluded that PSEG adequately identified, reviewed, and applied relevant industry operating experience. Based on interviews conducted during the inspection, workers at the site were willing to enter safety concerns into the CAP.

A. NRC Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. A self revealing non-cited violation of 10 CFR 50, Appendix B, criterion XVI, 'Corrective Action', occurred when a safety-related 4160 volt breaker did not operate as expected on July 24, 2007, due to hardened grease in the breaker mechanism. This was the third similar breaker failure in which PSEG did not identify or correct deficiencies that led to this nonconforming condition. PSEG subsequently replaced the breaker with a fully refurbished spare breaker, tested the breaker successfully, and revised the preventive maintenance tasks to address this issue in other similar breakers.

  This issue was greater than minor because it affected the equipment performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, not identifying and correcting this condition adverse to quality resulted in unplanned unavailability of various safety-related equipment such as support equipment for the D emergency diesel generator, the B control room emergency filtration supply fan, and the D filtration, recirculation, and ventilation system recirculation fan. The finding was determined to be of very low safety significance (Green) based on a Phase 1 screening evaluation. The finding has a cross-cutting aspect in the area of operating experience review because PSEG did not take appropriate corrective action to address the breaker grease hardening condition in a timely manner. P. 2. (b). (Section 4OA2.3.a)

Green. The inspectors identified, non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” occurred when on June 11, 2006, the “B” Control Room Emergency Filtration (CREF) damper flow controller did not meet its Technical Specification 3.7.2 required flow rate due to failure to implement corrective actions identified on October 1, 2004. The CREF failure resulted in high flow rate to the CREF Charcoal Filters and inoperability of the “B” CREF System. At the time of the event, PSEG repaired the affected power supply. During this inspection, replacement of 34 Westinghouse Model 75IC controller power supplies was incorporated into the Preventive Maintenance (PM) program.

The finding was greater than minor because it affected the barrier performance attribute of the Barrier Integrity cornerstone and adversely affected the objective to maintain the radiological barrier functionality of the control room. Specifically, the failure to implement corrective actions and correct a condition adverse to quality resulted in reduced effectiveness of the CREF Charcoal Filters to limit control room dose and over 19 hours unplanned unavailability of the “B” CREF System. The inspectors determined that the finding was of very low safety significance (Green). The finding was determined to have a cross-cutting aspect in the area of problem identification and resolution because the licensee did not take appropriate corrective actions to address safety issues in a timely manner. P. 1. (d). (Section 4OA2.3.b)

B. Licensee-Identified Violations

None.

REPORT DETAILS

4. OTHER ACTIVITIES (OA)

4OA2 Problem Identification and Resolution (PI&R) (Biennial - IP 71152B)

.1 Assessment of the Corrective Action Program

a. Inspection Scope

The inspectors reviewed procedures describing the corrective action program (CAP) at the Hope Creek (HC) Generating Station. The CAP process at HC requires that station personnel identify issues/problems and enter them into the CAP by writing notifications (NOTFs). PSEG supervisors and managers review these NOTFs to determine if conditions adverse to quality, human performance problems, equipment functionality, industrial or radiological safety concerns, or other significant concerns exist. Subsequently, operations personnel screen the NOTFs for operability and reportability, initially categorizes them by priority and significance levels, and forwards the NOTFs to the station ownership committee (SOC) and the management review committee (MRC) for further review. SOC and MRC review the initial decisions regarding significance, prioritization, and corrective actions and make adjustments, where necessary.

During this inspection, the inspectors reviewed notifications across the seven cornerstones of safety in the NRC's reactor oversight program (ROP) to determine if problems were being properly identified, characterized, and entered into the CAP for evaluation and resolution. The inspectors sampled items from the maintenance, operations, engineering, emergency preparedness, physical security, chemistry, radiation safety, licensed operator training, and nuclear oversight departments to assess performance in these areas. The inspectors also reviewed equipment operability determinations, reportability assessments, and extent-of-condition reviews for selected problems. Additionally, the inspectors reviewed equipment performance results and assessments documented in completed surveillance procedures, operator log entries, and trend data to determine whether the equipment performance evaluations were technically adequate to identify degrading or non-conforming equipment. The inspectors also performed plant walk downs to assess the material condition of the plant to determine whether observed equipment deficiencies were entered into the CAP.

The inspectors considered risk insights from the NRC's and PSEG's risk analyses to focus the sample selection and reviews on risk-significant components. The inspectors focused on high pressure coolant injection (HPCI), 4kV vital alternating current (AC) power, station auxiliaries cooling (SACS), emergency diesel generator (EDG), 120 VAC inverters, residual heat removal (RHR), reactor core isolation cooling (RCIC), service water (SSW), and service water ventilation as the most risk-significant systems. The inspectors also sampled other safety-related systems. For the selected risk significant systems, the inspectors reviewed the applicable system health reports, maintenance rule documents, a sample of engineering documents, and results from surveillance tests and maintenance work orders. In addition, the inspectors expanded the scope of the review to five years for 4kV vital AC power and safety-related 120 VAC inverters. For both of these systems, the inspectors reviewed issues related to the aging of electronic components.

The inspectors selected items from other station processes to verify that PSEG appropriately considered these items for entry into the CAP. Specifically, the inspectors sampled operator log entries, control room deficiency and operator work-around lists, operability determinations, unplanned LCO entries, system engineering walk downs, and completed surveillance tests. In addition, the inspectors interviewed plant staff and management to assess their understanding and involvement with the CAP, as well as the work environment at the station. The NOTFs and other documents reviewed, and a list of key personnel contacted, are listed in the attachment to this report. Selected NOTFs were assessed to determine whether PSEG adequately evaluated and prioritized the identified problems. The inspectors observed SOC and MRC meetings to assess the appropriateness of the assigned priority and significance, the scope and depth of the causal analysis, and the timeliness of the resolutions. For significant conditions adverse to quality, the inspectors reviewed the effectiveness of PSEG's corrective actions to preclude recurrence. The inspectors also reviewed equipment performance results and assessments documented in completed surveillance procedures, operator log entries, and trend data to determine whether the equipment performance evaluations were technically adequate to identify degrading or non-conforming equipment. The inspectors further reviewed selected evaluation methods used to evaluate open notifications such as, root cause analyses (RCA), apparent cause evaluations (ACE), common cause evaluations (CCE), and work group evaluations (WGE).

The inspectors reviewed the corrective actions associated with selected notifications to determine whether the actions addressed the identified problem causes. Notifications for repetitive problems were also selected for review to determine whether previous corrective actions were effective. Furthermore, the inspectors reviewed PSEG's timeliness in implementing corrective actions. The inspectors reviewed the notifications associated with selected non-cited violations (NCVs) and findings (FINs) to determine whether PSEG properly evaluated and resolved these issues.

The inspectors reviewed PSEG's use of NRC generic issues correspondence and industry operating experience (OE) by reviewing the station's OE procedures and verifying plant problems that were documented in notifications were not similar to previously reviewed OE provided to the station. The inspectors also reviewed self-assessment reports and audits to assess PSEG's ability to identify negative trends and enter them into the CAP. The NRC inspection results were compared and contrasted with PSEG audits and self-assessments to identify any significant deviations.

Lastly, during the course of interviews, the inspectors questioned plant management and staff on their willingness to identify plant problems and issues without the fear of retaliation or retribution. The inspectors also reviewed the employee concerns program (ECP) including the number of concerns received, the scope of the concerns, and the action taken in response to the identified concern including the communication between the ECP organization and the concerned individual. The inspectors considered the results of these interviews and document reviews to determine whether issues existed that may represent challenges to the free flow of information regarding safety concerns.

b. Assessment

Identification of Issues

The inspectors determined that PSEG adequately identified problems at an appropriately low threshold. Station personnel actively sought out deficient conditions and wrote NOTFs for these issues. Approximately 19,000 NOTFs were initiated at HC between December 2005 and August 2007. Of these, none were categorized as significance level (SL) 1, approximately 30 were categorized as SL2, approximately 300 were categorized as SL3, and the remainder were categorized as SL4 & SL5.

The inspectors observed high standards for housekeeping and cleanliness with the exception of a few areas. During a tour on September 18, 2007, the inspectors observed an active oil leak on the “D” EDG during a planned 24 hour run and local fire panel alarms without the required equipment deficiency tag. Following questions from NRC inspectors, the oil leak was subsequently entered into the CAP. The inspectors verified that NOTFs existed for the fire panel alarms. These issues were adequately addressed in the CAP.

Trending of identified deficiencies was generally good. The inspectors noted that trending of some administrative control issues in the plant security area was inconsistent in that there were a number of examples of deficient control of security-related information that occurred over approximately a two year period. While performance had improved in the previous six months, the inspectors noted that this trend had not been highlighted as an area needing improvement even though there had been both internal and external audits that had the opportunity to identify this trend.

Prioritization and Evaluation of Issues

The inspectors determined that PSEG appropriately screened notifications and properly classified them for safety significance and evaluation priority. The SOC and MRC meetings were observed to be effective at providing a detailed review and prioritization of issues. The quality of the causal analyses reviewed were generally detailed with adequate technical justification. The inspectors did note a range of quality among the reviewed evaluations. In general, the quality of the evaluations had improved, particularly in the last several months.

The inspectors also reviewed PSEG’s initial operability review and actions to address a high pressure coolant injection (HPCI) injection valve problem that was discovered on July 31, 2007 during routine quarterly valve stroke surveillance testing. Upon discovery that the valve remained shut after receiving and open signal, operations personnel took immediate action to open the valve manually. Personnel successfully freed the valve disc from the valve body using eight actuations of the manual “hammer” apparatus that is integral with the valve hand wheel. Operators fully opened the valve manually to verify free motion, stroked the valve electrically with no problems, and subsequently declared the valve and system operable but degraded. PSEG’s initial review concluded that the problem likely involved a thermal binding condition that occurred during a HPCI system automatic initiation that occurred on May 29, 2007, following a reactor scram. During this event, the HPCI system operated for 17 seconds before operators secured the system. PSEG implemented a procedure change that required cycling the HPCI injection valve for any short duration injections to avoid the potential for future thermal binding events. The inspectors considered this appropriate to limit the likelihood of any future thermal binding events.

Because of the potential safety significance of this event, PSEG initiated a root cause team to review the circumstances that surrounded the stuck HPCI injection valve. The PSEG team considered many potential causes including foreign material between the disk and seat and various mechanisms of thermal binding. The root cause team's final determination was that the valve became stuck due to a new type of thermal binding that was different than that described in previous NRC generic correspondence, such as Generic Letter 89-10 Supplement 6 regarding safety-related motor operated valve testing and GL 95-07 regarding pressure locking and thermal binding of safety-related power-operated gate valves. These references describe thermal binding as a phenomenon that occurs when a plant MOV is shut at high temperature, then the plant is subsequently cooled down, and the valve becomes stuck shut at cold conditions because of different rates of thermal expansion between the valve disk and seat. Immediately after the May scram, the HPCI system remained hot and the root cause team postulated that localized cooling of the disk and seat caused a contraction of both which allowed the disk to insert further (than normal) into the seat. PSEG contracted with a third party vendor who concluded that the type of thermal binding being postulated by PSEG was possible. The root cause team also noted that the successful freeing of the valve by eight actuations of the manual hand wheel hammer function further supported their assertion. The inspectors reviewed generic communication on this issue, interviewed root cause team members, conducted a conference call with the vendor, and consulted with NRC regional and headquarters valve experts. The inspectors did not identify any safety concerns related to PSEG's operability decision.

The inspectors did note that the root cause report and the applicable Licensee Event Report (LER) on this issue had not been approved at the conclusion of this inspection. The inspectors also noted that PSEG planned to perform additional testing of this valve in an upcoming refueling outage. The final disposition of this issue will be documented in a future inspection. This issue is unresolved pending review of the approved root cause report, LER, and the testing results of this valve. **(URI 05000354/2007006-01, Root Cause of HPCI Injection Valve Inoperability)**

Effectiveness of Corrective Actions

The inspectors concluded that corrective actions for identified deficiencies were typically timely and adequately implemented. Administrative controls were in place to ensure that corrective actions were completed as scheduled and reviews were performed to ensure the actions were implemented as intended. The inspectors also concluded that PSEG conducted in-depth effectiveness reviews for significant issues to determine if the corrective actions were effective in resolving the issue. In some cases, the licensee appropriately self-identified ineffective or improper closeout of corrective actions and reentered the issue into the CAP for further action. The inspectors did identify a few minor cases where corrective actions were not fully effective in addressing underlying deficiencies. For significant conditions adverse to quality, the inspectors noted that PSEG's actions were comprehensive and thorough, and generally successful at preventing recurrence.

Two findings of very low safety significance (Green) concerning effectiveness of corrective actions were identified during the inspection. The first self revealing finding involved the failure to identify and correct repetitive failures of a Class 1E 4KV circuit breaker and the second NRC identified finding involved the failure to identify and correct degraded power supplies for the damper controller of a control room emergency filtration (CREF) system.

c. Findings

1. ABB 4kV HK Circuit Breaker For D Vital Bus Failed Due To Hardened Grease

Introduction. A self-revealing non-cited violation of 10 CFR 50, Appendix B, criterion XVI, "Corrective Action," occurred when the 52-40401 D 4kV Class 1E vital bus feeder breaker (01 breaker) did not operate properly for the third time on July 24, 2007. The finding was determined to be of very low safety significance (Green).

Description. In 1991, Asea Brown Boveri (ABB) issued a revision to the HK series 4kV breaker technical manual that called for periodic (10-year) cleaning and lubrication of the breaker operating mechanism with Anderol 757 grease. These breakers are installed in various safety-related applications at HC. On April 21, 1995, the NRC issued Information Notice (IN) 95-22 informing reactor licensees of problems that could result due to grease hardening in ABB HK series 4kV breakers. By 1996, PSEG had developed 10-year preventive maintenance (PM) activities to overhaul the affected breakers in response to this industry issue. However, PSEG extended the PM frequency to 12 years on November 7, 2003 and the 01 breaker was last overhauled by ABB and received by PSEG on May 29, 1996. Subsequently, there were three mis-operations of these breakers that PSEG eventually attributed to grease hardening.

The first mis-operation of the 01 breaker occurred on October 5, 2004. The D vital bus offsite feeder breakers (01 and 08 breakers) failed to manually transfer which resulted in the unplanned tripping of several safety-related loads including the D service water pump, the D safety auxiliaries cooling pump (SACS) pump, and the B control rod drive pump. Plant personnel added a comment to the notification that documented the troubleshooting activities stating that the 01 breaker had not yet been refurbished as part of PSEG's action to address grease hardening issues. This comment was not addressed in the apparent cause evaluation (ACE) and PSEG's troubleshooting was limited to multiple cycles of the 01 breaker in the test stand and bench testing of the bailey logic modules. The resultant apparent cause evaluation did not include grease hardening as a potential cause and concluded that the most likely cause was an intermittent failure in a Bailey logic module that introduced a lag in the 01 breaker closing circuitry and caused a slight delay in the 01 breaker closure.

The second mis-operation of the 01 breaker occurred on April 20, 2007. The D vital bus offsite feeder breakers (01 and 08 breakers) failed to manually transfer which resulted in the unplanned trips of multiple D channel safety-related electrical loads. PSEG's troubleshooting included testing of the 08 breaker, inspection of the auxiliary contacts of the 08 breaker, and bench testing of bailey logic modules. The ACE for this event included a review of the October 2004 event, but did not identify grease hardening as a potential cause. This ACE identified two potential causes; a random failure of a Bailey solid state logic module or slow/sluggish operation of the 01 breaker mechanically operated breaker status cell switch utilized in the 08 breaker trip logic scheme. The inspectors noted that the ACE for this problem did not reasonably consider all likely causes because operating experience indicated that grease hardening could cause sluggish breaker operation.

The third mis-operation of the 01 breaker occurred on July 24, 2007. PSEG completed testing the degraded voltage relays associated with breaker 08 breaker, the normally open breaker that supplies alternate offsite power to the D vital bus. Operations personnel attempted to manually transfer (fast transfer) the D vital bus from the normal offsite supply (01 breaker) to the alternate offsite supply (08 breaker). The operators depressed the close push button on the 08 breaker, the 08 breaker indicated closed, and the 01 breaker indicated opened. When the operator released the 08 breaker close push button, the 08 tripped open and the 01 breaker re-closed to supply power to the D vital bus. The momentary loss of voltage resulted in initiation of the loss of power (LOP) emergency load sequencer and subsequent shedding of the D SACS pump, the B control room chiller and associated fans, and the D switchgear room fan.

PSEG performed a root cause analysis for this repetitive problem and determined that the root cause of the manual transfer failure was grease hardening. PSEG also determined that the 01 circuit breaker that had not been refurbished/overhauled for the last 11 years. In addition, PSEG noted that a 2001 EPRI study showed that ABB circuit breakers with Mobil 28 grease outperformed ABB breakers with Anderol 757 grease that is currently in use at Hope Creek. In response to these conclusions, PSEG initiated corrective actions that included replacement of the 01 breaker and revising the breaker overhaul frequency from 12 years to 6 years. PSEG also plans to replace the existing grease with Mobil 28 when the breakers are refurbished.

The inspectors concluded that the root cause for the July 2007 problem was at an appropriate technical depth to identify grease hardening as the likely cause of these three breaker mis-operations. The inspectors concluded that it was reasonable for PSEG personnel to have identified this cause in April 2007 after the second mis-operation. This conclusion was based on the highlighted operating experience, the breaker overhaul history, and repetitive sluggish breaker operation. The repetitive mis-operation of the ABB HK circuit breaker used for the D 4kV Class 1E vital bus feeder breaker constituted a performance deficiency.

Analysis. This issue was greater than minor because it affected the equipment performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, because PSEG did not identify and correct a condition adverse to quality which resulted in the unplanned unavailability of various safety-related support equipment for the D emergency diesel generator, the B control room emergency filtration supply fan, and the D filtration, recirculation, and ventilation system recirculation fan. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," the inspectors conducted a Phase 1 screening and determined the finding to be of very low safety significance (Green). The finding was not a design or qualification deficiency, did not represent a loss of system safety function, did not represent an actual loss of safety function of a single train for greater than its Technical Specification Allowed Outage

Time, did not represent an actual loss of safety function of one or more non-Technical Specification trains of equipment designated as risk significant per 10 CFR 50.65, for greater than 24 hours, and did not screen as potentially risk significant due to external events. The finding has a cross-cutting aspect in the area of operating experience review because PSEG did not adequately implement operating experience regarding grease hardening in breakers while evaluating a breaker mis-operation that occurred in April 2007. P.2 (b).

Enforcement. 10 CFR 50, Appendix B, criterion XVI, “Corrective Action,” states, in part, that “measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected.” Contrary to the above, PSEG did not correct the grease hardening condition for a safety-related ABB 4kV HK circuit breaker used in the D vital bus that resulted in sluggish operation and a failed manual transfer on July 24, 2007. The resultant momentary loss of power caused unplanned unavailability of multiple safety-related electrical loads. Because this violation was of very low safety significance and it was entered into PSEG’s corrective action program (20330712), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. **NCV 05000354/2007006-02, ABB 4kV HK Circuit Breaker For D Vital Bus Failed Due To Hardened Grease.**

2. Control Room Emergency Filtration Flow high due to failed Damper Controller Power Supply

Introduction The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, criterion XVI, “Corrective Action,” for a control room emergency filtration (CREF) damper controller that failed on June 11, 2006 to maintain its technical specification required flow rate. This failure occurred because PSEG did not implement corrective actions related to replacement of safety-related damper controller power supplies that was originally scheduled on October 1, 2004. The finding was determined to be of very low safety significance (Green).

Description On October 1, 2004, the CREF System Manager scheduled a number of report corrective action (CRCA) tasks to replace 42 power supplies associated with Westinghouse Model 75IC damper controllers that were in excess of 21 years old which was beyond the vendor expected lifetime of 12 to 15 years old. These power supply replacements were subsequently deferred twice, first on February 28, 2005, and then again on October 28, 2005. Subsequently, PSEG planners initiated a Preventive Maintenance (PM) order to replace the power supplies on November 17, 2005. This order closed the original CRCA tasks based on the expected completion of the PM. However, none of the power supplies have been replaced.

Subsequently, on June 11, 2006, a CREF damper controller, (H1GK-1GKFIC-9595B) did not maintain its technical specification required flow rate and was declared inoperable. During surveillance test, a nuclear plant operator found that CREF flow rate was 4600 cubic feet per minute (CFM), as compared to a setpoint of 4000 CFM with an acceptance band of +/- 400 CFM. After troubleshooting, PSEG personnel identified a faulty power supply as the cause of the excessively high flow and replaced a degraded filter capacitor in the power supply. The inspectors concluded that the high out of specification flow condition was due to faulty power supply that had not been replaced or refurbished as originally scheduled in October 2004. Similarly, on August 23, 2007, a non-safety related power supply for Westinghouse Model 75IC Auxiliary Building ventilation temperature controller, GJ-TV-9768B, was replaced after it failed due to use beyond its expected life.

The inspectors determined that PSEG personnel had recently incorporated 34 of the highest priority power supplies previously identified in 2004 into their current PM program. The first four power supplies are scheduled to be replaced on December 24, 2007, when they will been in service for approximately 24 years. The remaining power supply replacements have been scheduled over the next 16 months with the last final two power supplies scheduled to be replaced on April 15, 2009. The inspectors determined that only corrective maintenance had been performed on these power supplies after their failure.

The inspectors concluded the CREF damper failure that occurred on June 11, 2006 was foreseeable because the power supplies for the Westinghouse Model 75IC controllers installed in various safety-related applications were in service beyond their expected lifetime. PSEG had scheduled damper power supply replacements in October 2004, but the replacements were not accomplished. As a result, a safety-related flow controller for the CREF system did not operate as required by Technical Specifications. The inspectors concluded that PSEG personnel did not implement timely corrective actions for the replacement of power supplies for various safety-related dampers and this constituted a performance deficiency.

<u>Analysis</u> The finding was greater than minor because it affected the structures, systems, or components (SSCs) performance attribute of the Barrier Integrity cornerstone and adversely affected the objective to maintain radiological barrier functionality of the Control Room. Specifically, the failure to implement corrective actions and correct a condition adverse to quality resulted in reduced effectiveness of the CREF Charcoal Filters to limit Control Room dose and led to unplanned unavailability of the CREF System for approximately 19 hours. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," the inspectors conducted a Phase 1 screening and determined the finding to be of very low safety significance (Green). The finding was screened as Green because the condition only represented a degradation of the radiological barrier function of the control room environment. The finding was determined to have a cross-cutting aspect in the area of problem identification and resolution because PSEG did not implement appropriate corrective actions to address safety issues in a timely manner. P.1 (d)

<u>Enforcement</u> 10 CFR 50, Appendix B, criterion XVI, “Corrective Action,” states, in part, that “measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected.” Contrary to the above, PSEG failed to implement the corrective actions identified by the October 1, 2004 CRCA tasks. As a result, the “B” CREF Damper controller did not maintain its Technical Specification required flow rate on June 11, 2006. The resultant equipment problem reduced the effectiveness of the CREF Charcoal Filters to limit post accident control room dose and added approximately 19 hours of unplanned unavailability to the “B” CREF System. Because this finding was of very low safety significance and it was entered into PSEG’s corrective action program (20287588), this violation is being treated as an NCV, consistent with section VI.A.1 of the NRC Enforcement Policy. **NCV 05000354/2007006-03, B CREF Failure Due To Damper Controller Power Supply Failure**

.2 Assessment of the Use of Operating Experience

a. Inspection Scope

The inspectors reviewed a sample of operating experience (OE) issues for applicability to Hope Creek and the associated actions PSEG implemented to address the potential issues. The inspectors selected the samples from NRC Generic Communications, industry OE sources, and noteworthy issues from other reactor sites. The inspectors reviewed the method in which OE was communicated through out the station, and where appropriate, verified that applicable issues were entered into the CAP. The inspectors also reviewed open notifications to determine if adverse trends in equipment performance were reflective of inadequate implementation of the lessons learned form operating experience.

b. Assessment

No findings of significance were identified in the area of operating experience.

The inspectors determined that OE was transmitted to individual departments and was reviewed by either the supervisor or the CAP coordinator. Additionally, some departments, such as engineering, had department staff perform additional reviews. High priority OE was coded by its significance by the corporate staff and forwarded to plant personnel with a list of expected actions. The inspectors reviewed open notifications and assessed equipment performance and concluded that, in general, operating experience was being adequately reviewed and assessed at the station with one exception related to the finding on 4kV breakers. In that case, the inspectors found that some OE was not sufficiently analyzed for applicability. NRC Information Notice (IN) 93-26, Grease Solidification Causes Molded Case Circuit Breaker Failure To Close, was evaluated as not applicable to Hope Creek because the exact breaker type associated with the OE was not used at Hope Creek. Another example screened out as not applicable to Hope Creek was NRC IN 96-43, Failures of General Electric Magne-Blast Circuit Breakers, which also contained OE describing grease hardening in breakers. In addition, PSEG could not retrieve CAP documentation associated with NRC IN 95-22, Hardened Or Contaminated Lubricants Cause Metal-Clad Circuit Breaker Failures, that informed reactor licensees of problems that could result due to grease hardening in ABB HK series 4kV breakers which Hope Creek uses in various safety related applications. PSEG's CAP included a 1999 self-assessment of their ABB 4kV breaker maintenance program in response to a 1998 industry operating experience notice on circuit breaker reliability. That self-assessment noted that the breaker maintenance training documentation did not reference IN 95-22 and considered that fact noteworthy because of the information notice title and because the breakers at the Salem station were experiencing breaker grease hardening issues at the time.

.3 Assessments and Audits

a. Inspection Scope

The inspectors reviewed a sample of nuclear oversight (NOS) audits and other assessments. The inspectors verified that problems identified through the audits and assessments were entered into the CAP. The effectiveness of the audits and self-assessments was evaluated by comparing audit and self-assessment results against NRC findings and NRC observations during the inspection.

b. Assessment

No findings of significance were identified in the area of assessments and audits.

The inspectors found that PSEG's audits and assessments were generally thorough and probing. The NOS audits evaluated the performance of each department quarterly and color coded the results (green, white, yellow, red). Managers developed action plans to address weaknesses identified by NOS. Department self assessments were also sampled and found to be of generally good quality.

.4 Safety Conscious Work Environment

a. Inspection Scope

The inspectors assessed the willingness of plant staff to raise concerns and use the CAP without fear of retaliation during interviews with plant employees and management. The interviews spanned several different departments and levels in the organization. The inspectors also reviewed the Employee Concerns Program (ECP) to determine if employees were aware of the program and used it to raise concerns. Several ECP cases were reviewed to assess the safety conscious work environment (SCWE) at the station.

b. Assessment

No findings of significance were identified related to SCWE. During interviews, plant staff expressed a willingness to use the CAP to identify plant issues and deficiencies and stated that they were willing to raise safety issues. The inspectors noted that no one interviewed stated that they personally experienced or were aware of a situation in which an individual had been retaliated against for raising a safety issue.

The inspectors reviewed approximately 20 of the 73 ECP files that had been generated since January 2007 to determine if there were potentially adverse trends that could be reflective of work environment issues. The inspectors did note that some of these concerns related to personnel administrative practices within the Hope Creek operations department. Inspector follow up of these issues noted that many of these issues were no longer concerns because they are being adequately addressed to minimize their effect on the work environment at the station.

4OA6 Meetings, including Exit:

On September 28, 2007, the inspectors presented the inspection results to Mr. George Barnes and other members of the PSEG staff. The inspectors confirmed that no proprietary information reviewed during inspection was retained.

ATTACHMENT: Supplemental Information

In addition to the documentation that the inspectors reviewed (listed in the attachment), copies of information requests given to the licensee are located in the Agencywide Document Access and Management System (ADAMS), under accession number ML071150197.

ATTACHMENT - SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel:
J. Molner – HC Emergency Preparedness Manager
M. Gaffney - Regulatory Assurance Manager
P. Duka - Regulatory Assurance Engineer
F. Possessky - Regulatory Assurance Engineer
G. Neron - Regulatory Assurance Engineer
P. Kordziel - System Engineer, High Pressure Coolant Injection
R. LaSala, - System Engineer, Service Water
D. Schiller, - System Engineer, 481/482 Inverters
J. Schaeffer – System Engineer, Electrical Systems
J. Cichello - System Engineer, Control Room Emergency Filtration
R. Schmidt – Principal Nuclear Engineer
R. Binz – IST Program Administrator
A. Tramontana – NSSS Systems Engineering Manager
T. Baban – BOP Systems Engineering Manager
G. Daves – Electrical Systems Engineering Manager
M. Pfizenmaier – HC Program Manager
K. Knaide – Senior Manager, Plant Engineering
D. Boyle, - Operations Support Manager
W. Kopchick - Operations Shift Superintendent
B. Booth - Operations Director
J. Pike – Maintenance Superintendent, I&C
W. Schmick – Maintenance Superintendent, Electrical
M. Crisafulli – Maintenance Superintendent, Mechanical
M. Headrick - Employee Concerns Manager
M. Patti – Manager, Security Programs
W. Guthrie – Manager, Security Operations
K. Hoffman – Security Program Analyst

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Items Opened and Closed:

05000354/2007006-01	URI	Root Cause of HPCI Injection Valve Inoperability

Items Opened and Closed:

05000354/2007006-02	NCV	ABB 4kV HK Circuit Breaker For D Vital Bus Failed Due To Hardened Grease
05000354/2007006-03	NCV	B CREF Failure Due To Damper Controller Power Supply Failure

LIST OF DOCUMENTS REVIEWED

Procedures
ER-AA-1200, Critical Component Failure (CCF) Clock, Revision 3
ER-AA-2020, EPIX & MSPI Failure Determination Evaluation, Revision 3
ER-AA-310, Implementation of the Maintenance Rule, Revision 6
ER-AA-310-1001, Maintenance Rule - Scoping, Revision 3
ER-AA-310-1002, Maintenance Rule - SSC Risk Significance Determination, Revision 2
ER-AA-310-1003, Maintenance Rule - Performance Criteria Selection, Revision 3
ER-AA-310-1004, Maintenance Rule - Performance Monitoring, Revision 5
ER-AA-310-1005, Maintenance Rule - Dispositioning Between (a)(1) and (a)(2), Revision 5
ER-AA-310-1006, Maintenance Rule - Expert Panel Roles and Responsibilities, Revision 3
ER-AA-310-1007, Maintenance Rule - Periodic (a)(3) Assessment, Revision 4
ER-AA-310-1008, Exelon Maintenance Rule Process Map, Revision 0
ER-AA-600-1044, Maintenance Rule Support, Revision 3
HC.ER-DG.ZZ-0002(Z), System Function Level Maintenance Rule Scoping Vs Risk Reference, Revision 3
HC.OP-AP-ZZ-0108(Q), Operability Assessment and Equipment Control Program, Revision 31
HU-AA-101, Human Performance Tools and Verification Practices, Revision 3
HU-AA-104-101, Procedure Use and Adherence, Revision 2
HU-AA-1081, Fundamentals Tool Kit, Revision 2
HU-AA-1081-F-01, Functional Area and Cross-Functional Fundamentals, Engineering Fundamentals, Revision 1
HU-AA-1081-F-02, Functional Area and Cross-Functional Fundamentals, Chemistry, Radwaste and Environmental Fundamentals Trifold, Revision 1
HU-AA-1081-F-03, Functional Area and Cross-Functional Fundamentals, Maintenance Fundamentals Trifold, Revision 1
HU-AA-1081-F-04, Functional Area and Cross-Functional Fundamentals, Nuclear Oversight Fundamentals, Revision 3
HU-AA-1081-F-05, Functional Area and Cross-Functional Fundamentals, Operations Fundamentals, Revision 4
HU-AA-1081-F-06, Functional Area and Cross-Functional Fundamentals, Radiation Protection Fundamentals, Revision 1
HU-AA-1081-F-07, Functional Area and Cross-Functional Fundamentals, Nuclear Security Fundamentals, Revision 2
HU-AA-1081-F-08, Functional Area and Cross-Functional Fundamentals, Training Fundamentals, Revision 3
HU-AA-1081-F-09, Functional Area and Cross-Functional Fundamentals, Daily Work Control Fundamentals, Revision 2
HU-AA-1081-F-10, Functional Area and Cross-Functional Fundamentals, Leadership Fundamentals, Revision 1
HU-AA-1081-F-11, Functional Area and Cross-Functional Fundamentals, Outage Management Fundamentals, Revision 1
HU-AA-1081-F-12, Functional Area and Cross-Functional Fundamentals, Supply/Procurement Engineering Fundamentals, Revision 1
HU-AA-1081-F-13, Functional Area and Cross-Functional Fundamentals, Project Management Fundamental, Revision 1
HU-AA-1081-F-14, Functional Area and Cross-Functional Fundamentals, Outage Services Fundamentals Trifold, Revision 1

HU-AA-1081-F-15, Functional Area and Cross-Functional Fundamentals, Emergency Response Organization Fundamentals, Revision 2
LS-AA-115, Operating Experience Procedure, Revision 10
LS-AA-120, Issue Identification and Screening Process, Revision 6
LS-AA-125, Corrective Action Program (CAP) Procedure, Revision 11
LS-AA-125-1001, Root Cause Analysis Manual, Revision 6
LS-AA-125-1002, Common Cause Analysis Manual, Revision 5
LS-AA-125-1003, Apparent Cause Evaluation Manual, Revision 7
LS-AA-125-1004, Effectiveness Review Manual, Revision 2
LS-AA-125-1005, Coding and Analysis Manual, Revision 5
LS-AA-126, Self-Assessment Program, Revision 4
LS-AA-126-1001, Focused Area Self-assessments, Revision 4
LS-AA-126-1002, Management Observation of Activities, Revision 1
LS-AA-126-1005, Check-in Self-assessments, Revision 3
LS-AA-126-1006, Benchmarking Program, Revision 1
MA-AA-716-003, Tool Pouch-Minor Maintenance, Revision 2
NC-WM-AP-ZZ-0003(Q), Regular Maintenance Process, Revision 5
NC-WM-AP-ZZ-0004(Z), Technical Support Order Process, Revision 0
OP-AA-108-112, Definition and Measurement of Mispositioned Plant Components, Revision 2
WC-AA-101, On-Line Work Control Process, Revision 13
WC-AA-106, Work Screening and Processing, Revision 5
HC.MD-CM.EA-0002(Q), Service Water Pump Overhaul Repair, Revision 17
HC.MD-PM.PB-0001, 4.16 kV Breaker Cleaning and PM, Revision 22
HC.MD-ST.PB-0003, Class 1E 4.16 kV Feeder Degraded Voltage Monthly Instrumentations Channel Functional Test, Revision 23 & 24
HC.OP-IS.BJ-0101, High Pressure Coolant Injection System Valves – Inservice Test, Revision 52
HC.OP-IS.BD-0101, Reactor Core Isolation Cooling System Valves – Inservice Test, Revision 48
HC.ER-DG.ZZ-0002, System Function Level Maintenance Rule Scoping Vs. Risk Reference, Revision 3
HC PRA-005.06, PRA System Notebook High Pressure Coolant Injection System, Revision 2
HC.OP-ST.BJ-0002, HPCI System Function Test (Low Pressure) – 18 Months and HPCI System Response Time Test (High Pressure), Revision 31
HC.OP-EO.ZZ-0322, Core Spray Injection Valve Override, Revision 1
HC.OP-EO.ZZ-0101A, ATWS – RPV Control, Revision 2
HC.OP-AR.ZZ-0016, Overhead Annunciator Window Box E3, Att. E2, Revision 12
HC.OP-AB.ZZ-0173, Loss of 4.16 kV Bus 10A404 D Channel, Revision 3
WC-MW-112, Predefine Deferrals, Revision 3
SY-AA-102, Exelon's Nuclear Fitness For Duty Program, Revision 10 & 11

QA Audits/Reports
NOH07-013 Elevation Notice - Learning Programs final
NOSA-HPC-07-01 Corrective Action Program Audit
NOSA-HPC-07-03 Security Plan FFD & Access Authorization Audit Report
NOSA-HPC-07-05 Engineering Design Contrl Audit Plan Final
NOSA-HPC-07-12 ISFSI AUDIT REPORT FINAL
NOSPA-HC-07-1Q Final Report
NOSPA-HC-07-2Q NOS 2Q07 Assessment Plan

SEC-07-004 Response to NOH07-003 Elevation Notice
2006 NIEP Audit of PSEG - Final
HC-2007-C-04 MTE ELEVATION RESPONSE
noh06-002 NOSPA-HC-05-4Q Report
NOH06-005elevation letter - maintenance 2-3-06
NOH06-006 elevation letter - engineering 2-3-06
NOH06-007 elevation letter - work management 2-3-06
NOH06-010 Security Escalation Notice 3-06
NOH06-011 Hope Creek Second Quarter 2006 Assessment Plan
NOH06-017 NOSPA-HC-06-1Q Report
NOH06-018 Hope Creek Third Quarter 2006 Assessment Plan
NOH06-020 NOSPA-HC-06-2Q Report Revision 1
noh06-020 NOSPA-HC-06-4Q HC Assessment cycle assessment plan rev1
NOH06-021 NOSPA-HC-06-3Q Final Report
NOH06-021 NOSPA-HC-06-4Q R1 Final
NOH06-023 HCNOS Assessment Cycle 1Q07 Assessment Plan
NOH07-001 NOSPA-HC-06-4Q Final Report
NOH07-003 Security Elevation Notice Final
NOH07-004 HC MTE Elevation Notice Final
NOH07-014 NOSPA-HC-07-2Q Report FINAL
NOSA-HPC-06-01 Maintenance Audit Report Final
NOSA-HPC-06-02 report
NOSA-HPC-06-03 EP report
NOSA-HPC-06-04 HC Audit Report final
NOSA-HPC-06-05 Eng Prog report r1
NOSA-HPC-06-06 Audit Rpt Fina
NOSA-HPC-06-07 Surveilance and Test Audit Report final
NOSA-HPC-06-08 Audit Report Final
NOSA-HPC-06-9 FP Audit Report
NOSA-HPC-06-10 ISFSI Audit Report
NOSA-HPC-06-15 report
NOSA-HPC-07-02 Audit Report FINAL
NOSA-HPC-07-02 Matls Mgt and Procurement Eng Audit Report
NOSA-HPC-07-04 EP report
NOSA-HPC-07-06 RP Audit Report (Final)
NOSA-NCS-07-08 Nuclear Fuels Audit Report
NOSA-SLMHPC-07-11 M&TE Audit Report
NOSPA-HC-07-2Q Report FINAL

Self Assessments
Focused Area Self-assessment Report, NIEP FASA - PSEG Nuclear, dated July 21, 2006
1999 Self-Assessment of ABB 4kV Breaker Maintenance Program

Notifications * Denotes Notifications written as a result of this inspection

20041987	20143746	20175867	20196819	20206876	20229788
20048405	20169848	20178847	20198128	20209097	20229807
20115504	20172171	20178849	20201232	20214464	20231195
20137236	20172173	20190392	20205725	20217401	20234319
20142659	20175866	20193401	20206078	20217402	20243009

20243041	20271284	20280148	20291444	20307164	20318499
20243514	20271345	20280230	20291738	20307384	20319420
20244964	20271356	20280542	20292498	20307432	20319430
20244976	20271564	20280547	20293086	20307547	20319594
20244977	20271573	20280567	20293503	20307780	20319872
20248264	20271650	20280686	20294178	20307945	20320049
20251203	20271691	20280698	20294192	20308040	20321400
20255050	20271709	20280820	20294990	20309043	20321570
20255078	20271737	20281564	20295252	20309117	20321643
20255566	20271798	20281774	20296072	20309123	20321659
20262671	20271832	20281790	20296604	20309233	20321805
20262825	20271894	20281853	20296634	20309749	20321964
20263903	20272116	20281876	20297489	20310317	20322186
20265316	20272559	20281920	20297976	20310863	20322712
20265325	20273035	20282023	20298025	20311345	20322814
20265382	20273069	20282629	20298054	20311542	20323208
20265393	20274055	20283086	20298057	20311744	20323238
20265676	20274095	20283421	20298063	20311948	20323371
20265716	20274220	20283666	20298092	20312135	20324033
20266189	20274277	20283808	20298616	20312147	20324374
20266247	20274522	20284290	20298617	20312189	20324801
20266584	20274560	20284486	20298753	20312201	20324854
20266835	20275262	20284489	20299595	20312821	20324899
20266868	20275663	20284490	20299710	20313117	20325041
20267024	20276098	20284501	20299713	20313218	20325131
20267234	20277376	20284502	20299866	20313443	20325469
20267265	20277661	20284503	20299963	20313568	20325620
20267346	20277705	20284718	20301531	20313677	20325677
20267348	20277825	20284753	20301667	20313793	20325769
20267355	20277898	20284755	20302555	20313795	20326142
20267563	20278036	20285436	20302956	20314136	20326404
20267642	20278135	20285645	20303125	20314399	20326419
20267643	20278147	20285718	20303148	20314451	20326697
20267666	20278563	20286373	20303359	20314917	20326941
20268019	20278937	20287588	20303494	20315097	20326984
20268147	20278956	20287606	20303535	20315484	20327597
20268148	20279121	20287766	20303704	20315485	20327673
20268175	20279141	20287801	20303705	20316895	20328443
20268827	20279379	20287803	20303878	20316916	20328462
20268972	20279402	20287897	20303983	20316927	20328657
20269217	20279403	20288268	20304407	20316934	20328839
20269255	20279461	20288775	20305789	20317039	20328891
20269308	20279513	20288875	20305949	20317089	20329229
20269755	20279654	20289498	20306197	20317090	20329566
20270115	20279813	20290600	20306267	20317145	20329666
20270198	20279873	20291029	20306473	20317397	20329727
20270536	20279909	20291203	20306773	20317417	20330226
20271104	20280007	20291204	20306791	20317940	20330350
20271175	20280055	20291318	20306870	20318268	20330358

20330373	20330952	20331508	20333243	20333931	20336658
20330593	20331042	20332175	20333260	20333932	20336684*
20330611	20331273	20332565	20333431	20334099	20336872
20330677	20331274	20332582	20333517	20335238	
20330712	20331299	20332936	20333652	20336239	
20330846	20331410	20333154	20333852	20336507	

Orders and Evaluations

70031552	70053478	70057642	70061960	70067742	70071577
70034128	70053592	70057665	70062078	70068032	70071700
70041611	70053737	70057668	70063062	70068108	70071759
70041887	70053797	70057669	70063813	70068113	70071798
70041887	70053837	70057672	70064279	70068320	70071847
70046039	70053956	70057673	70064438	70068418	70071848
70049601	70054002	70057674	70064509	70068576	70071884
70050760	70054090	70057968	70064520	70068662	70071901
70051096	70054180	70058226	70064556	70068680	70071901
70052037	70054207	70058227	70064952	70068691	70072044
70052492	70054235	70058352	70065091	70068739	70072045
70052568	70054261	70058550	70065093	70068964	70072347
70052632	70054399	70059468	70065209	70069296	70072388
70052652	70054516	70059573	70065209	70069348	70072808
70052666	70054833	70059803	70065279	70069721	70072858
70052848	70054968	70059940	70065286	70069857	70072903
70052848	70055323	70060148	70065621	70069878	70073007
70052848	70055764	70060655	70065696	70069878	70073586
70052891	70055864	70060799	70065880	70069960	80020613
70052891	70055873	70060968	70066410	70070081	80065896
70053150	70056098	70060995	70066535	70070090	80082924
70053221	70056140	70061214	70066563	70070364	80087548
70053240	70056200	70061423	70066564	70070579	80087880
70053311	70056421	70061946	70066650	70071038	80092597
70053344	70056789	70061956	70066785	70071289	80093351
70053378	70057128	70061959	70067589	70071400	

Work Orders

30086675	30127369	50093848	50106145	60061563	60071042
30096485	30135397	50096190	60012463	60061841	60071086
30110360	30135436	50096345	60046069	60062625	60071086
30115856	50078422	50100581	60047238	60063300	60071157
30118573	50091817	50102695	60048374	60067490	60071600
30123522	50092292	50103575	60048500	60068876	
30124662	50092293	50104895	60058718	60068976	

Operating Experience Reviews

NRC Bulletin (BL) 83-08, Electrical Circuit Breakers With An Undervoltage Trip Feature In Use In Safety-Related Applications Other Than The Reactor Trip System

NRC Information Notice (IN) 93-26, Grease Solidification Causes Molded Case Circuit Breaker Failure To Close
NRC Information Notice (IN) 95-22, Hardened Or Contaminated Lubricants Cause Metal-Clad Circuit Breaker Failures
NRC Information Notice (IN) 95-47, Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage
NRC Information Notice (IN) 96-08, Thermally Induced Pressure Locking of a High Pressure Coolant Injection Gate Valve
NRC Information Notice (IN) 96-43, Failures of General Electric Magne-Blast Circuit Breakers
NRC Information Notice (IN) 98-02, Nuclear Power Plant Cold Weather Problems and Protective Measures
NRC Information Notice (IN) 98-22, Deficiencies Identified During NRC Design Inspections

Non-Cited Violations and Findings Reviewed

FIN 2005004-02, Automatic Trip Of Service Air Compressor
FIN 2005004-03, Emergency Instrument Air Compressor Capacity
NCV 2005004-04, Untimely Licensee Event Report For The 'A' Control Room Emergency Filtration Subsystem
NCV 2005005-01, Vacuum Breaker Mechanical Environmental Qualification Implementation
NCV 2005007-03, High Pressure Coolant Injection Minimum Flow Valve Degraded Condition
NCV 2006002-01, Failure To Implement Corrective Actions For Service Water Pump Packing
NCV 2006002-03, Failure To Identify Conditions Adverse To Quality On 'D' Service Water Strainer
NCV 2006002-04, Inadequate Corrective Action Results In Unavailability Of The 1AK400 Control Room Chiller
NCV 2006003-01, Corrective Actions To Prevent Repeat Failures Of Service Water Strainer Overloads Not Implemented
NCV 2006003-02, Loss Of Shutdown Reactor Pressure Vessel Level Indication
NCV 2006003-03, Deficiency In Access Control To Radiological Areas
NCV 2006004-01, 'A' Core Spray Minimum Flow Check Valve Failure
FIN 2006004-02, Inadvertent Instrument Air Compressor Trip
NCV 2006005-01, Unplanned Trip Of The 'B' Control Room Chiller And Ventilation Train
NCV 2006005-02, Mispositioned Valve Results In Reduced Cooling Flow To The 'C' Emergency Diesel Generator
NCV 2006005-03, High Temperature Condition On 'B' Emergency Diesel Generator Not Fully Identified In A Timely Manner
NCV 2006015-01, Inadequate Strainer Differential Pressure Design Control To Ensure Adequate
FIN 2006015-02, Inadequate Containment Vent Valve Backup Pneumatic Supply
Service Water Flow
NCV 2007003-01, Failure To Perform A Risk Assessment When Required By 10 CFR 50.65(A)(4)
NCV 2007003-02, Failure To Follow Test Procedure Results In Reactor Scram
NCV 2007007-01, Fitness-For-Duty (FFD) Collection Personnel Collecting FFD Samples From Co-Workers
NCV 2007007-02, Fitness-For-Duty (FFD) Collectors Leaving FFD Specimens Unattended

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J-105-0, Sh. 9, Logic Diagram Sequencer Fan-Out, Revision 5
J-105-0, Sh. 10, Logic Diagram Sequencer Fan-Out, Revision 6
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J-107-0, Sh. 2, Emergency Load Sequencer, Revision 5
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LIST OF ACRONYMS

ACE	Apparent Cause Evaluations
CAP	Corrective Action Program
CCE	Common Cause Evaluation
CFCU	Containment Fan Coil Unit
DCP	Design Change Package
DRP	Division of Reactor Projects
ECP	Employee Concerns Program
EMIS	Equipment Malfunction Identification System
FASA	Functional Area Self-Assessments
FFD	Fitness-For-Duty
FIN	Finding
HELB	High Energy Line Break
IMC	Inspection Manual Chapter
MPFF	Maintenance Preventable Functional Failure
MR	Maintenance Rule
MRC	Management Review Committee
NCVs	Non-Cited Violations
NOS	Nuclear Oversight
NOTF	Notification
NRC	Nuclear Regulatory Commission
OE	Operating Experience
PI&R	Problem Identification and Resolution
RCA	Root Cause Analyses
ROP	Reactor Oversight Process
RP	Radiation Protection
SCWE	Safety Conscious Work Environment
SDP	Significant Determination Process
SFF	System Functional Failure
SOC	Station Ownership Committee
SPAR	Standardized Plant Analysis Risk
SRA	Senior Risk Analyst
SRST	Spent Resin Storage Tank

SSC	System, Structure, or Component
SW	Service Water
SWIS	Service Water Intake Structure
TDAFW	Turbine Driven Auxiliary Feedwater
TS	Technical Specification
URI	Unresolved Item
WGE	Work Group Evaluation