October 10, 2003

Mr. John L. Skolds, President
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3
NRC COMBINED BIENNIAL ENGINEERING INSPECTION
REPORT 050000237/2003008(DRS); 05000249/2003008(DRS)

Dear Mr. Skolds:

On August 29, 2003, the U.S. Nuclear Regulatory Commission (NRC) completed combined biennial baseline engineering inspections at the Dresden Nuclear Power Station, Units 2 and 3. The enclosed report documents the inspection findings which were discussed on August 29, 2003, with Mr. R. Hovey and other members of your staff.

The inspection examined activities conducted under your license as they related to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

The combined inspections focused on: (1) the evaluations of changes, tests and experiments; (2) permanent plant modifications; and (3) safety system design and performance capability (SSDI) of selected important systems. Specifically, the SSDI portion of the inspections focused on the design and performance capability of the High Pressure Coolant Injection (HPCI) and the 4 kV and the 480 volt electrical systems to assure that the selected systems were capable of performing required safety related functions.

Based on the results of this inspection, there were four NRC-identified findings of very low safety significance which involved violations of NRC requirements. However, because these violations were of very low safety significance and because the issues were entered into the corrective action program, 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 contest the subject or severity of a Non-Cited Violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 801 Warrenville Road, Lisle, IL 60532-4351; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Dresden Nuclear Power Station.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be made available electronically for public inspection in the NRC Public Document Room or from the Publically Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Julio F. Lara, Chief
Electrical Engineering Branch
Division of Reactor Safety

Docket Nos. 50-237; 50-249
License Nos. DPR-19; DPR-25

Enclosure: Inspection Report 05000237/2003008(DRS); 05000249/2003008(DRS) w/Attachment: Supplemental Information

cc w/encl: Site Vice President - Dresden Nuclear Power Station
Dresden Nuclear Power Station Plant Manager
Regulatory Assurance Manager - Dresden
Chief Operating Officer
Senior Vice President - Nuclear Services
Senior Vice President - Mid-West Regional Operating Group
Vice President - Mid-West Operations Support
Vice President - Licensing and Regulatory Affairs
Director Licensing - Mid-West Regional Operating Group
Manager Licensing - Dresden and Quad Cities
Senior Counsel, Nuclear, Mid-West Regional Operating Group
Document Control Desk - Licensing
M. Aguilar, Assistant Attorney General
Illinois Department of Nuclear Safety
State Liaison Officer
Chairman, Illinois Commerce Commission

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Vice President - Mid-West Operations Support
Vice President - Licensing and Regulatory Affairs
Director Licensing - Mid-West Regional Operating Group
Manager Licensing - Dresden and Quad Cities
Senior Counsel, Nuclear, Mid-West Regional Operating Group
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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-237; 50-249
License Nos:	DPR-19; DPR-25
Report No:	05000237/2003008(DRS); 05000249/2003008(DRS)
Licensee:	Exelon Generation Company, LLC
Facility:	Dresden Power Station, Units 2 and 3
Location:	6500 North Dresden Road Morris, IL 60450
Dates:	August 11 through August 29, 2003
Inspectors:	H. Walker, Lead Inspector S. Sheldon, Engineering Inspector C. Baron, Contract Inspector, Mechanical S. Spiegelman, Contract Inspector, Mechanical G. Skinner, Contract Inspector, Electrical
Approved by:	Julio F. Lara, Chief Electrical Engineering Branch Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000237/2003008(DRS); 05000249/2003008(DRS); 08/11/03 - 08/29/03; Dresden Nuclear Power Station, Units 2 & 3; Safety Systems Design and Performance Capability.

This report covered a 3-week period of combined engineering inspections by regional engineering specialists with both electrical and mechanical consultant assistance. The inspection focused on: (1) the evaluations of changes, tests and experiments; (2) permanent plant modifications; and (3) safety system design and performance capability of a selected important system or systems. These inspections were performed in accordance with NRC baseline inspection procedure 71111.DS, “Plant Design - Pilot,” which was written to combine three baseline inspection procedures.

Four Green findings associated with four non-cited violations were identified. Violations of very low safety significance were identified during the inspection and are issued as Non-Cited Violations. The significance of most findings is indicated by their color (Green, White, Yellow, 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 3, dated July 2000.

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green. A finding of very low safety significance was identified by the inspectors for a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." The licensee failed to translate Motor Operated Valve (MOV) duty cycle limitations into specifications, drawings, procedures, or instructions. The High Pressure Coolant Injection (HPCI) turbine trip set point was set such that the turbine would experience repetitive starts and stops in certain types of small or medium loss of coolant accidents. This cycling could potentially challenge the reliability of the 2301-8 HPCI injection motor operated valves, which have a design limit of five strokes followed by 30 minutes of cooldown time.

  The issue was more than minor because this vulnerability affected the mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of the HPCI system. The issue was of very low safety significance because it did not represent an actual loss of a safety function. The issue was a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, which required translation of applicable design basis into specifications, drawings, procedures, and instructions. (Section 1R21.1)

- Green. A finding of very low safety significance was identified by the inspectors for a violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control.” Although previously identified by the licensee, the licensee failed to protect equipment required to shut down the reactor and maintain it in a safe shutdown condition from the environmental effects of a postulated high energy steam line break. A High Energy Line

Break (HELB) in the HPCI system could make the swing diesel, required by both Units 2 and 3, inoperable.

This issue was more than minor because the Unit 2/3 swing diesel generator and associated engineered safety features systems could be degraded by the HELB conditions. The issue was of very low safety significance because it did not represent an actual loss-of-safety function. This issue was a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, which required translation of regulatory requirement and the design basis into drawings, specifications, procedures, and instructions. (Section 1R21.2).

- Green. A finding of very low safety significance was identified by the inspectors for a violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions." The licensee failed to promptly identify and correct a malfunction within the High Pressure Coolant Injection (HPCI) system Motor Gear Unit (MGU). Operators identified that the MGU did not operate as designed on May 25, 2001. After two unsuccessful attempts to correct the problem, troubleshooting was accomplished on November 6, 2002, which identified degradation within the MGU motor. The motor was replaced, returning the system to full functionality, on March 12, 2003.

  This issue was more than minor because the lack of timeliness associated with resolution of this issue impacted the mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of the HPCI system. This finding is of very low safety significance because there was not a complete loss of function as demonstrated by surveillance testing. This issue was a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XVI, which required prompt identification and correction of conditions adverse to quality. (Section 4OA2.b.1)

- Green. A finding of very low safety significance was identified by the inspectors for a violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions." The licensee failed to take appropriate corrective action for multiple failures of safety related 4160V circuit breakers.

  This issue is more than minor because it affected the mitigating system cornerstone objective of equipment reliability, in that failure of circuit breakers to operate on demand could cause loss of function of safety related loads needed to mitigate an accident. The issue is of very low safety significance because the failure of two breakers serving redundant loads would be needed to cause loss of safety system function, and there was no evidence that two such breakers were inoperable at the same time. This issue was a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XVI which required prompt identification and correction of conditions adverse to quality. (Section 4OA2.b.2)

B. <u>Licensee-Identified Violations</u>

- No findings of significance were identified.

REPORT DETAILS

Summary of Plant Status

Dresden Units 2 and 3 operated at or near full power throughout the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R02 Evaluations of Changes, Tests, or Experiments (71111.DS)

.1 Review of 50.59 Evaluations and Screenings

a. Inspection Scope

The inspectors reviewed six evaluations performed pursuant to 10 CFR 50.59. The evaluations related to permanent plant modifications, setpoint changes, procedure changes, conditions adverse to quality, and changes to the updated final safety analysis report. The inspectors confirmed that the evaluations were thorough and that prior NRC approval was obtained as appropriate. The inspectors also reviewed fifteen screenings where the licensee personnel had determined that a 10 CFR 50.59 evaluation was not necessary. In regard to the changes reviewed where no 10 CFR 50.59 evaluation was performed, the inspectors verified that the changes did not meet the threshold to require a 10 CFR 50.59 evaluation. These evaluations and screenings were chosen based on risk significance of samples from the different cornerstones.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.DS)

a. Inspection Scope

The inspectors reviewed six permanent plant modifications that had been installed in the last two years. The modifications were chosen based upon the effect on systems that had high probabilistic risk analysis (PRA) significance in the plant Individual Plant Evaluation (IPE) or high maintenance rule safety significance. The inspectors reviewed the modifications to verify that the completed design changes were in accordance with the specified design requirements and the licensing bases and to confirm that the changes did not affect any systems' safety function. Design and post-modification testing aspects were verified to ensure the functionality of the modification, its associated system, and any support systems. The inspectors also verified that the modifications performed did not place the plant in an increased risk configuration.

b. Findings

No findings of significance were identified.

1R21 Safety System Design and Performance Capability (71111.DS)

Introduction: Inspection of safety system design and performance capability verifies the initial design and subsequent modifications and provides monitoring of the ability of the selected systems to perform design bases functions. As plants age, the design bases may be lost and important design features may be altered or disabled. The plant's risk assessment model was based on the capability of the as-built safety system to perform the intended safety functions successfully. This inspectable area verifies aspects of the mitigating systems cornerstone for which there are no indicators to measure performance.

The objective of the safety system design and performance capability inspection (SSDI) is to assess the adequacy of calculations, analyses, other engineering documents, and operational and testing practices that were used to support the performance of the selected systems during normal, abnormal, and accident conditions.

The systems and components selected for the inspection were the High Pressure Coolant Injection (HPCI) and the 4 kV and 480 volt electrical systems. These systems were selected for review based upon:

- having a high probabilistic risk analysis ranking;
- having had recent significant issues;
- not having received recent NRC review; and
- being interacting systems.

The criteria used to determine the acceptability of the system's performance was found in documents such as:

- applicable technical specifications;
- applicable updated safety analysis report (USAR) sections; and
- the systems' design documents.

The following system and component attributes were reviewed in detail:

System Requirements

Process Medium - water, air, electrical signal;
Energy Source - electrical power, steam, air;
Control Systems - initiation, control, and shutdown actions;
Operator Actions - initiation, monitoring, control, and shutdown; and
Heat Removal - cooling water and ventilation.

System Condition and Capability

Installed Configuration - elevation and flow path operation;
Operation - system alignments and operator actions;
Design - calculations and procedures; and
Testing - level, flow rate, pressure, temperature, voltage, and current

Component Level

Equipment/Environmental Qualification - temperature and radiation;
Equipment Protection - fire, flood, missile, high energy line breaks (HELBs), freezing, heating, ventilation and air conditioning.

.1 System Requirements

a. Inspection Scope

The inspectors reviewed the USAR, technical specifications, system descriptions, drawings and available design basis information to determine the performance requirements of the HPCI and the 4 kV and the 480 volt electrical systems. The reviewed system attributes included process medium, energy sources, control systems, operator actions and heat removal. The rationale for reviewing each of the attributes was:

Process Medium: This attribute required review to ensure that the selected systems' flow paths would be available and unimpeded during/following design basis events. To achieve this function, the inspectors verified that the systems would be aligned and maintained in an operable condition as described in the plant's USAR, technical specifications and design bases.

Energy Sources: This attribute required review to ensure that the selected systems motive/electrical source would be available/adequate and unimpeded during/following design basis events, that appropriate valves and system control functions would have sufficient power to change state when required. To achieve this function, the inspectors verified that the interactions between the systems and their support systems were appropriate such that all components would operate properly when required.

Controls: This attribute required review to ensure that the automatic controls for operating the systems and associated systems were properly established and maintained. Additionally, review of alarms and indicators was necessary to ensure that operator actions would be accomplished in accordance with design requirements.

Operations: This attribute was reviewed because the operators perform a number of actions during normal, abnormal and emergency operating conditions that have the potential to affect the selected systems operation. In addition, the emergency operating procedures (EOPs) require the operators to manually realign the systems flow paths during and following design basis events. Therefore, operator actions play an important role in the ability of the selected systems to achieve their safety-related functions.

Heat Removal: This attribute was reviewed to ensure that there was adequate and sufficient heat removal capability for the selected systems.

b. Findings

HPCI Turbine Trip Setpoint

Introduction: The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green) for failure to translate Motor Operated Valve (MOV) duty cycle limitations into specifications, drawings, procedures, or instructions. This issue was considered to be NRC-identified because the licensee had failed to implement the appropriate modification since identification in July 2000.

Description: The inspectors reviewed the adequacy of control for the HPCI system. The HPCI system was initiated by low reactor water level or high drywell pressure. The HPCI system would trip, per design, on high reactor water level at a setpoint of 42.975 inches of water. The water level trip would reset as the water level decreased below 42.6 inches and a HPCI start would then reinitiate if the high drywell signal was still present. The inspectors noted that for some small or medium break LOCA scenarios, the potential existed for the HPCI system to cycle on and off several times within the first few minutes of the event.

In order to evaluate the response of the system, a LOCA was simulated in the Dresden training simulator corresponding to .1 percent of a recirculating pump discharge line. Without any operator action, the HPCI system initiated, then tripped and reset twice within the first few minutes of the event. The simulator response was used for demonstration purposes as it related to the inspectors' concerns. The inspectors recognized that the simulator time response might be somewhat different from the actual plant response.

This potential for cycling challenges the reliability of the 2301-8 HPCI injection MOVs. Dresden Operating Procedure (DOP) 0040-01, "Station Motor Operated Valve Operations," limits the operation of MOVs to five starts (or strokes) within 1 minute followed by a minimum 30 minutes of cooldown. The inspectors concluded that operator action would be required to prevent the repetitive cycling from exceeding the duty cycle limits.

Licensee personnel had a previous opportunity to address this issue, since this vulnerability was identified during a self-assessment at Quad Cities in July 2000. A modification was subsequently developed and installed to address this issue at Quad Cities. A problem identification form documenting Quad Cities concerns in this area and modifications to address the concerns was provided to Dresden as required by the operating experience program. Dresden engineering evaluated the Quad Cities issue and the modification and determined that the change was not necessary at Dresden. The inspectors agreed with the licensee that operators were trained to keep the water level below this band of vulnerability and that procedures existed to secure HPCI so that cycling would be stopped prior to damaging the equipment, to ensure that HPCI remained operable. However, reliance on operator action to prevent potential damage to an MOV is contrary to the USAR description regarding automatic operation of HPCI. Furthermore, while operators had guidance and training to control water level in a certain band, in certain accident scenarios (e.g., small break LOCA), operators would be challenged to take control prior to exceeding the design limit of five strokes.

Analysis: The inspectors determined that the failure to translate MOV duty cycle limitations into appropriate specifications, drawings, procedures, and instructions was a performance deficiency warranting a significance determination. The inspectors concluded that the finding was greater than minor in accordance with Inspection Manual Chapter (IMC) 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Screening,” issued on June 20, 2003. The finding was determined to be greater than minor because it affected mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The potential existed to cycle the HPCI system repeatedly during a small break LOCA, which would challenge the reliability of the 2301-8 MOV.

The inspectors completed a significance determination of this issue using IMC 0609, “Significance Determination Process,” dated April 21, 2003, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” dated March 18, 2002. This finding is considered a design deficiency which did not result in loss of function per Generic Letter 91-18, “Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions,” Revision 1. Therefore, this finding was considered to be of very low safety significance (Green). This finding was assigned to the mitigating systems cornerstone for both units.

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, "Design Control," required, in part, that measures be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions.

Contrary to the above, as of August 27, 2003, the design basis requirements were not correctly translated into specifications, drawings, procedures, or instructions. Specifically, MOV duty cycle requirements were not adequately translated into a specification for the reset value on the HPCI high reactor water level turbine trip set-point. The licensee entered the issue into its corrective action program as CR 173179 on August 26, 2003. During the inspection, the licensee implemented corrective actions to revise the reset setpoint. Because this violation was of very low safety significance and it was entered into the licensee’s corrective action program, the violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000237/2003008-01; 05000249/2003008-01).

.2 System Condition and Capability

a. Inspection Scope

The inspectors reviewed design basis documents and plant drawings, abnormal and emergency operating procedures (EOPs) , requirements, and commitments identified in the USAR and technical specifications. Information in these documents was compared to applicable electrical, instrumentation and control, and mechanical calculations, setpoint changes and plant modifications. The inspectors also reviewed operational procedures to verify that instructions to operators were consistent with design assumptions.

Information was reviewed to verify that the actual system condition and tested capability was consistent with the identified design bases. Specifically, the inspectors reviewed the installed configuration, the system operation, the detailed design, and the system testing, as described below.

Installed Configuration: The inspectors confirmed that the installed configuration of the High Pressure Coolant Injection (HPCI) and the 4 kV and the 480 volt electrical systems met the design basis by performing detailed system walkdowns. The walkdowns focused on the installation and configuration of piping, components, and instruments; the placement of protective barriers and systems; the susceptibility to flooding, fire, or other environmental concerns; physical separation; provisions for seismic and other pressure transient concerns; and the conformance of the currently installed configuration of the systems with the design and licensing bases.

Operation: The inspectors performed procedure walk through of selected manual operator actions to confirm that the operators had the knowledge and tools necessary to accomplish actions credited in the design basis.

Design: The inspectors reviewed the mechanical, electrical and instrumentation design of the HPCI and the 4 kV and the 480 volt electrical systems to verify that the systems and subsystems would function as required under accident conditions. The review included a review of the design basis, design changes, design assumptions, calculations, boundary conditions, and models as well as a review of selected modification packages. Instrumentation was reviewed to verify appropriateness of applications and set-points based on the required equipment function. Additionally, the inspectors performed limited analyses in several areas to verify the appropriateness of the design values.

Testing: The inspectors reviewed records of selected periodic testing and calibration procedures and results to verify that the design requirements of calculations, drawings, and procedures were incorporated in the system and were adequately demonstrated by test results. Test results were also reviewed to ensure automatic initiations occurred within required times and that testing was consistent with design basis information.

b. Findings

Failure to Incorporate Design Bases into Design Documents

Introduction: The inspectors identified a Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” having very low safety significance (Green) for failing to correctly translate design basis requirements into drawings, specifications, procedures, and instructions. In some cases equipment required to shut down the reactor and maintain it in a safe shutdown condition was not protected from the environmental effects of a postulated high energy line break (HELB) should the break occur in the high pressure coolant injection (HPCI) turbine/pump area. This issue was considered to be NRC-identified. This condition had been previously identified by licensee personnel in 1995 and a Justification for Continued Operation (JCO) was prepared and later closed. However, as of August 27, 2003, design changes or licensing basis changes to resolve the issue had still not been implemented.

Description: Section 3.6.1.1.1.1 of the Dresden Updated Final Safety Analysis Report (UFSAR), "Postulated Line Breaks," addressed the protection of plant equipment and structures from ruptures in high energy pipes outside of containment. This section stated that equipment and structures required to shut down the reactor and maintain it in a safe shutdown condition should be protected from all effects resulting from ruptures in pipes carrying high energy fluid (assuming a concurrent and unrelated single active failure of protected equipment). The rupture effects to be considered included pipe whip, structural, and environmental.

Calculation DR-055-M-001, "P/T Response Following a HPCI Steam Line Break in the HPCI Room," Revision 0, dated September 2, 1994, specified the potential environmental consequences of the rupture of the 10 inch steam supply piping to the HPCI pump turbine in the Unit 2 or 3 HPCI pump room. The calculation stated that a postulated steam line break could result in elevated pressures and temperatures in other reactor building compartments, including the Unit 2/3 swing diesel generator room. The calculation predicted peak conditions of approximately 16 psia and 290 degrees Fahrenheit in the swing diesel generator room.

In response to the inspector's concerns regarding possible inappropriate closing of the JCO, licensee personnel identified a 10 CFR 50.59 safety evaluation, dated July 3, 1996. The evaluation was issued to support UFSAR changes reflecting electrical equipment environmentally qualified for conditions associated with a HPCI steam line break. The evaluation did not address other effected electrical equipment that was not qualified (i.e., the Unit 2/3 swing diesel generator and related equipment). Licensee personnel did not protect or environmentally qualify the Unit 2/3 swing diesel generator and related equipment. As a result, the postulated HPCI steam line break (assuming a loss of offsite power and a concurrent and unrelated single active failure of the other safety related diesel generator) could result in the interruption of all AC power to the affected unit. This scenario would still require the use of other non-safety related equipment, as described in the JCO.

Licensee personnel stated that CR 173612 had been initiated, and that a new 10 CFR 50.59 evaluation would be performed to determine if NRC approval had been required prior to implementing a licensing basis change to allow the loss of the Unit 2/3 swing diesel generator as a consequence of this steam line break event.

Analysis: The inspectors determined that the failure to provide EQ protection for plant equipment and structures required to shut down the reactor and maintain it in a safe shutdown condition was a performance deficiency warranting significance determination. The inspectors concluded that the finding was greater than minor in accordance with Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening" issued on June 20, 2003. The finding was determined to be greater than minor because it affected mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The potential existed for the Unit 2/3 swing diesel generator and associated engineered safety features systems to be degraded by these conditions. Loss of this electrical power supply could affect equipment required to shut down the reactor and maintain it in a safe shutdown condition after the HELB event.

The inspectors completed a significance determination of this issue using IMC 0609, “Significance Determination Process,” dated April 21, 2003, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” dated March 18, 2002. This finding is considered a design deficiency which did not result in loss of function per Generic Letter 91-18, “Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions,” Revision 1. Therefore, this finding was considered to be of very low safety significance (Green). This finding was assigned to the mitigating systems cornerstone for both units.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," required, in part, that measures be established to assure that applicable regulatory requirements and the design basis are correctly translated into drawings, specifications, procedures, and instructions. Dresden UFSAR Section 3.6.1.1.1.1 stated, in part, that equipment and structures required to shutdown the reactor and maintain it in a safe condition should be protected from all effects resulting from ruptures in pipes carrying high energy fluid.

Contrary to the above, as of August 27, 2003, equipment required to shut down the reactor and maintain it in a safe shutdown condition was not protected from the environmental effects of a postulated high energy line break. Specifically, the Unit 2/3 swing diesel generator and associated engineered safety features systems could have been degraded by a HPCI steam line break. The licensee entered the issue into its corrective action program as CRs 172179 and 173092 on August 19 and August 26, 2003, respectively. Because this violation was of very low safety significance and it was entered into the licensee’s corrective action program, the violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy, (NCV 05000237/2003008-02; 05000249/2003008-02).

4. OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems (71152)

Routine Review of Identification and Resolution of Problems

a. Inspection Scope

During baseline inspection activities the inspectors routinely review issues to verify that they were being entered into the licensee’s corrective action program at an appropriate threshold, that adequate attention was given to timely corrective actions, and that adverse trends were identified and addressed. Minor issues entered into the corrective action program as a result of inspectors’ observations are included in the list of documents reviewed, which are attached to this report.

b. Findings

.1 HPCI Turbine Motor Gear Unit Corrective Action

Introduction: The inspectors identified a NCV of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," having very low safety significance (Green) for failing to promptly identify and correct a malfunction within the HPCI system Motor Gear Unit (MGU).

Description: The inspectors reviewed Work Orders and Condition Reports associated with a failure in the HPCI MGU. On May 25, 2001, operators noted that the HPCI MGU did not operate as expected with the control switch in the “slow raise” position. A condition report was initiated to document the condition and a work order was initiated to replace the switch. The determination that the system was operable was recorded in the operations logs.

On August 24, 2001, the MGU control switch was replaced. However, this did not correct the problem and CR 00073238 was initiated to document the issue. During the refueling outage on October 25, 2001, the MGU control switch was replaced again, and again this did not correct the problem.

Troubleshooting was delayed until November 6, 2002, at which time licensee personnel determined that the motor itself was degraded and needed to be replaced. The licensee determined that the degradation was due to grease from the gear case entering the motor. Had this degradation been allowed to progress, it would ultimately prevent HPCI from performing its safety function. In summary, licensee personnel allowed this degraded condition to exist for 18 months, entering and exiting a refueling outage, before determining that the MGU motor needed to be replaced. On March 12, 2003, the motor was replaced, returning the system to full functionality. During this time period, the HPCI was considered operable as demonstrated by satisfactory completion of technical specification required surveillance testing of the system.

Analysis: The inspectors concluded that the failure to promptly identify and correct a degraded condition with the HPCI MGU was a performance deficiency warranting significance determination. The inspectors concluded that the finding was greater than minor in accordance with Inspection Manual Chapter (IMC) 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Screening,” issued on June 20, 2003. The finding was determined to be greater than minor because it affected the mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the MGU motor was degraded and further degradation would ultimately result in affecting HPCI mitigation system function.

The inspectors completed a significance determination of this issue using IMC 0609, “Significance Determination Process,” dated April 21, 2003, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” dated March 18, 2002. The inspectors answered “no” to all five screening questions in the Phase 1 Screening Worksheet under the Mitigating System column. The inspectors concluded the issue was of very low safety significance (Green). This finding was assigned to the mitigating systems cornerstone for both units.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures 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, a degraded condition in the safety-related HPCI MGU, which was first reported on May 25, 2001, was not identified until November 6, 2002 and was not corrected until March 12, 2003. The licensee entered this issue into its corrective action program as CR 173178 on August 26, 2003. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program, the violation is being treated as a NCV consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000237/200308-03; 05000249/200308-03).

.2 Inadequate Corrective Actions on Breaker Failures

Introduction: The inspectors identified a NCV of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," having very low safety significance (Green) for failing to promptly identify and correct multiple failures of 4 kV circuit breakers.

Description: The inspectors identified five similar instances of horizontal draw-out breakers failing to operate on demand, either open or close, since March 12, 2002. Two of these failures occurred on the first operation attempt after being racked in following maintenance. Three instances occurred on breakers that were in service and had been previously operated successfully. The re-racking practice was performed in accordance with Procedure DOP 6500-07 which allowed re-racking a breaker if did not close on the first attempt and an electrical control failure was suspected. The re-racking practice was intended to re-align the breaker in the cubicle so that the position switches would operate properly. The "52h" contacts in the cubicle position switch are used in the closing and tripping circuits of the breaker and are used to detect whether the breaker is in the "test" or "operate" position. If the 52h contacts do not close, the breaker may fail to close or open on demand.

The practice of re-racking a breaker after failing to operate on its first attempt after racking in was recently questioned by station personnel (CR 13154, dated October 16, 2002), but was re-affirmed as an acceptable method of ensuring proper breaker operation. However, the station missed at least three opportunities to recognize that breakers that had been sufficiently aligned to operate at their initial operation, subsequently failed in service, apparently due to position switch malfunction (CRs 126262, 130697, and 160504). These cases demonstrated that successful initial alignment did not ensure future operability. In each of these cases the breakers were re-racked and returned to service without investigating whether the breaker could fail again due to breaker movement in its cubicle, or some other mechanism.

The five failed breakers in this discussion included modified hybrid breakers consisting of Merlin-Gerin operating mechanisms mounted on the original GE truck, as well as original GE Magne-Blast breakers. Both the modified breakers and the unmodified breakers were mounted in unmodified GE type MC-4.76 switchgear cubicles. Therefore, the interface between the breakers and the cubicles suspected of causing the problem was the same in all five cases.

Licensee personnel failed to correct the misalignment of the breakers with the switchgear cubicles and instead implemented bumping or re-racking the breaker to get the breaker to operate. This did not address the root cause of the failures and failed to prevent subsequent failures of breakers in service due to apparent movement of the breakers and/or position switches.

Analysis: The inspectors determined that the failure to adequately evaluate and correct the breaker failures was a performance deficiency warranting a significance determination. The inspectors concluded that the finding was greater than minor in accordance with Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening" issued on June 20, 2003. The finding was determined to be greater than minor because it affected mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences.

The inspectors completed a significance determination of this issue using IMC 0609, "Significance Determination Process," dated April 21, 2003, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," dated March 18, 2002. The inspectors answered "no" to all five screening questions in the Phase 1 Screening Worksheet under the Mitigating System column. The inspectors concluded the issue was of very low safety significance (Green). This finding was assigned to the mitigating systems cornerstone for both units.

Enforcement: Title 10 CFR Part 50 Appendix B, Criterion XVI, requires that conditions adverse to quality be promptly identified and corrected. Contrary to this requirement, the licensee failed to promptly correct five breaker failures since March 12, 2002. The licensee entered the issue into its corrective action program as CRs 173092 and 173179 on August 26, 2003. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program, this violation is being treated as a NCV consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000237/2003008-04; 05000249/2003008-04).

4OA3 Event Follow-Up (71153)

(Closed) LER 50-237/03-002-00: High Pressure Coolant Injection Room Cooler Bearing Degradation

On July 10, 2003, licensee personnel observed excessive noise and vibration from the Unit 2 High Pressure Coolant Injection (HPCI) room cooler. The room cooler was secured and was later restarted for trouble shooting. It was again secured and was taken out of service. On July 11, 2003, maintenance personnel replaced the inboard and outboard fan shaft bearings for Unit 2, performed post maintenance testing and the Unit 2 system was declared operable. The Unit 3 High Pressure Coolant Injection Room Coolers were inspected and no problems were identified.

Subsequent investigation by licensee personnel indicated that in 1996 the preventive maintenance lubrication frequency for the HPCI room coolers was incorrectly changed from quarterly to every two years. As part of the corrective actions, the lubrication frequency for both Unit 2 and Unit 3 HPCI room coolers was changed from the two year frequency back to quarterly. Licensee personnel indicated that the present preventive maintenance program now has adequate controls in place to prevent this type of event from occurring.

The LER was reviewed by the inspectors and the necessity for operability of the room cooler was discussed with licensee personnel. No findings of significance were identified. This item is closed.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. R. Hovey and other members of licensee management at the conclusion of the inspection on August 29, 2003. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
R. Hovey, Site Vice President
D. Bost, Plant Manager
C. Byers, Mechanical Engineer
P. DiSalvo, HPCI Systems Engineer
J. Fox, Mechanical Design Engineer
D. Galanis, Design Engineering Manager
J. Hansen, Regulatory Assurance Manager
J. Henry, Operations Manager
T. Leffler, Systems Engineer
R. Rybak, Regulatory Assurance
A. Shahkarami, Engineering Director
J. Siepek, Nuclear Oversite Manager
J. Strasser, Electrical Design Engineer

Nuclear Regulatory Commission
D. Smith, Senior Resident Inspector
P. Pelke, Resident Inspector
J. Lara, Chief, Electrical Engineering Branch

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000237/2003008-01 05000249/2003008-01	NCV	Failure to Translate HPCI Motor Operated Duty Cycle Limitations into Specifications, Drawings, Procedures, or Instructions (Section 1R21.1)
05000237/2003008-02 05000249/2003008-02	NCV	Failure to Correctly Translated Design Basis Environmental Requirements into Equipment Required to Shut down the Reactor and Maintain it in a Safe Condition (Section 1R21.2)
05000237/2003008-03 05000249/2003008-03	NCV	Failure to Promptly Identify and Correct a Malfunction Within the HPCI System Motor Gear Unit (Section 4OA2.b.1)
05000237/2003008-04 05000249/2003008-04	NCV	Inadequate Corrective Action to Determine the Cause and Correct Similar Failures of Safety Related 4160v Circuit Breakers to Operate on Demand (Section 4OA2.b.2)

Closed

05000237/2003008-01 05000249/2003008-01	NCV	Failure to Translate HPCI Motor Operated Duty Cycle Limitations into Specifications, Drawings, Procedures, or Instructions (Section 1R21.1)
05000237/2003008-02 05000249/2003008-02	NCV	Failure to Correctly Translated Design Basis Environmental Requirements into Equipment Required to Shut down the Reactor and Maintain it in a Safe Condition (Section 1R21.2)
05000237/2003008-03 05000249/2003008-03	NCV	Failure to Promptly Identify and Correct a Malfunction Within the HPCI System Motor Gear Unit (Section 4OA2.b.1)
05000237/2003008-04 05000249/2003008-04	NCV	Inadequate Corrective Action to Determine the Cause and Correct Similar Failures of Safety Related 4160v Circuit Breakers to Operate on Demand (Section 4OA2.b.2)
05000237/2003002	LER	High Pressure Coolant Injection Room Cooler Bearing Degradation (Section 4OA3)

Discussed

None.

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed for the Dresden Nuclear Power Station during the combined engineering inspections of: (1) safety system design and performance capability; (2) evaluation of changes, tests or experiments; and (3) permanent plant modifications. This list includes documents prepared by licensee personnel as well as documents prepared by others for the licensee. Inclusion on the list does not imply that NRC inspectors reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document, unless specifically stated in the inspection report.

Calculations

057604 (CEMD); Evaluation of Components in ECCS Rooms for Elevated Temperature; Revision 1

0591-576-001; HPCI Flooding Analysis; Revision 1

065895 (EMD); Blow-down Force at HPCI Piping Terminal End; Revision 0

7317-43-19-1; Dresden Unit 2 Electrical Load Monitoring System (ELMS)-AC; Revision 22A

7927-25-19-1; Protective Device Settings Verification; Revision 0

87-981/982; HPCI Static Pressure and Total Developed Pressure; Revision 0

8982-13-19-6; Calc. for Second-level Undervoltage Relay Setpoint; Revision 4

9198-19-19-3; Dresden 3/I Safety Related Continuous Load; Revision 3A

BSA-D-00-01; Dresden 2/3 ECCS Room Temperature Response with Loss of Room Cooler; Revision 0

D2-HPCI-FTG; HPCI System Fatigue Assessment

DR-019-E002; 4kV Bus 23-1/33-1 & 24-1/34-1 Coordination Study; Revision 3

DR-27D-E014; SBO Bus-ties Relay Settings and Coordination Study; Revision 1

DR-030-M-001; Pressure/Transient Analysis for HPCI Steamline Break in Torus; Revision 3

DR-055-M-001; P/T Response Following a HPCI Steamline Break in the HPCI Room; Revision 0

DR-245-M-003; ECCS Vortex Flow Limits for CCST, Tanks 2/3-A, B-3303; Revision 0

DR-721-M-003; The Emergency Air Cooler Coil Performance Evaluation for the HPCI Turbine/Pump Rooms; Revision 0

DRE96-0124; Dresden HPCI NPSH Temperature Limits; Revision 1

DRE96-126; Motor Terminal Voltage Calculation for Dresden 250VDC Motor Operated Valves; Revision 001B

DRE96-0152; Condensation Inside Rosemount Transmitters Following a HPCI Steam Line Break; Revision 0

DRE96-0206; HPCI Pump Discharge Pressure for 5000 gpm Flow to Reactor Vessel; Revision 1

DRE96-0215; Pressure Drop Analysis for HPCI Exhaust Steam Piping; Revision 0

DRE97-0068; Minimum Flow through the LPCI and HPCI Pump Minimum Flow Lines; Revision 2

DRE97-0145; Maximum Flow Through the LPCI and HPCI Minimum Flow Lines; Revision 2

DRE97-154; Dresden unit 3 Estimates of Insulation Debris for ECCS Strainer Head Loss Calculations; Revision 3

DRE98-007; Dresden HPCI Room Thermal Response with Reduced Room Cooler Capability; Revision 0

DRE98-0018; ECCS Strainer Head Loss Estimates for Dresden Station Units 2/3; Revision 2

DRE98-0030; Determination of Setpoint of CST Low-Low Level Switches to Prevent Potential Air Entrainment from Vortexing during HPCI Operation; Revision 0

DRE98-056; Sources if Fibrous Debris in Unit 2 Drywell Considered for Clogging of ECCS Strainers; Revision 2

DRE98-0183; Sizing of Restriction Orifice in HPCI GSLO Drain Pump Vent Line; Revision 1

DRE99-0012; Frictional Pressure Loss in HPCI Turbine Steam Supply Piping; Revision 0

DRE99-0013; Hydraulic Performance of the HPCI System; Revision 2

DRE00-0040; HPCI Steam Line Flow Timer Error Analysis; Revision 0

DRE00-0054; Condensate Storage Tank Level Error Analysis; Revision 0

DRE02-0049; High Pressure Core Injection (HPCI) Valve Open Permissive Pressure Switches; Revision 001

DRE03-0025; Baseline Calculation for 125VDC ELMS-DC Conversion to DCSDM; Revision 000

E098-0077; Dresden HPCI Room Thermal Response with Reduced Room Cooler Capacity; Revision 0

NED-I-EIC-0096; High Pressure Coolant Injection (HPCI) Turbine Pressures; Revision 2

NED-I-EIC-108; High Pressure Coolant Injection Turbine and Pump Area Temperature Switch Setpoint Error Analysis; Revision 0

NED-I-EIC-0109; High Pressure Coolant Injection (HPCI) Pump Discharge Flow Loop Accuracy and Minimum Flow Setpoints; Revision 5

NEP-12-02.04; Loss of Voltage Relay Setpoint for Busses 23-1, 24-1, 33-1 and 34-1; Revision 001

PMED-898230-01; Development of a Duty Cycle Based on a More Conservative Application of Coincident Starting Currents for the 250-VDC Battery System; Revision 013C

Condition Reports Written as a Result of the Inspection

170031; HPCI Test Return Valve Stroke Time Greater than Assumed; dated August 1, 2003

171132; USAR Section 7.3 Discrepancy; dated August 11, 2003

171368; Calculation does not Represent Current Plant Conditions; dated August 13, 2003

171423; Assumption in Calc DRE97-0068 Needs Clarification; dated August 13, 2003

171491; Scaffold Erected to Replace Hoist in D2 HPCI Room; dated August 14, 2003

171534; Editorial Errors Found in UFSAR Section 6.2.1.3.2.1 Text; dated August 14, 2003

171657; EQ Binder EQ 44D Tab E Does Not Contain Tab C Requirements; dated August 14, 2003

171675; Non-Conservative Assumption in HPCI Line Break Calculation; dated August 14, 2003

171677; Stroke Time Concern with 2301-10 MOVs; dated August 14, 2003

171679; Inappropriate Access Level provided for NRC Inspectors; dated August 14, 2003

171742; Cannot Retrieve SILs for HPCI system; dated August 14, 2003

171764; Formal Documentation of HPCI Lube Oil Cooler; dated August 15, 2003

171788; Potential Enhancement to the 50.59 Evaluation Sign Off Sheet; dated August 15, 2003

171809; Foam Spacers not Adequately in Place; dated August 15, 2003

172047; MOV 2301-8 Full Load Current Used in 250 ELMS-DC Database; dated August 18, 2003

172179; Procedure Revision Required to Aid in Performance Monitoring; dated August 19, 2003

172215; Scram Derate and PHC Commitments in Passport Deleted; dated August 19, 2003

172217; Commitment deleted from W.O. 98107178; dated August 19, 2003

172537; Inadequate Documentation of Response to SIL 555; dated August 21, 2003

172551; No CR for Out of Tolerance Temperature Instrument Surveill; dated August 21, 2003

172737; SSDI Identifies Labeling Error on Graph in EQ Calculation; dated August 22, 2003

172856; Dresden SSDI Inspection; dated August 25, 2003

173092; 4kV Horizontal Breaker Failures; dated August 26, 2003

173114; Potential Hot Spots for HPCI room; dated August 28, 2003

173178; HPCI Unit 2 MGU - Timeliness of Corrective Action; dated August 26, 2003

173179; SSDI Identifies Potential HPCI Design Inadequacy; dated August 26, 2003

173612; HPCI/HELP 10 CFR 50.59 Documentation; dated August 28, 2003

173626; Investigate how Dresden Dispositioned Quad Cities CAs; dated August 28, 2003

173674; HELB Temperature Effects on Degraded Voltage Relays; dated August 29, 2003

174387; Impact of Loss of HPCI Room cooler on Appendix K Analysis; undated

Condition Reports Reviewed During the Inspection

D2001-03053; U2 HPCI MGU Manual Control Switch Doesn’t Work in SLOW RAISE Mode; dated June 7, 2001

34594-02; Vertical Lift Cubicle overhauls (DSDC-HIT); dated December 15, 2002

50987-01; CR Document D2001-02340; dated May 1, 2001

50987-02; Perform Root Cause Analysis to Determine Cause of 3B Recirc Pump Transient; dated July 02, 2001

50987-12; Revise PMTs for all Vertical Lift 4KV Breakers to include verification of 52H contact continuity (trip capability); dated September 10, 2001

50987-14; Revise MA-AB-EM-5-00113 to check trip paddle bolts for tightness; dated June 27, 20003

50987.02; Root Cause Evaluation Report – Reactor Scram Due to Low Reactor Level Caused by 3B Reactor Recirculation Run-up & Trip; dated April 27, 2001

73238; U2 HPCI MGU C/S Replaced & Problem Still Exists; dated August 25, 2001

77171; Pipe Support Was Not Taking Dead Weight; dated September 28, 2001

77181; Walkdown of HPCI Support M-1187D-80 with NRC Inspectors; dated September 28, 2001

77498; Bus 12 to TR 15 high Side Bkr Overcurrent Trip; dated October 2, 2001

77966; 2/3 E Lift Pump Malfunction; October 6, 2001

78351; Calibration Data Incorrect for EC 7957; dated September 26, 2001

79603; Torus Level Switches 2351A & B out of Tech Spec.; dated October 21, 2001

81358; 2-2301-8 Valve Rotated in Wrong Direction; dated November 1, 2001

83169; Lift Pp Breakers failed to operate as required; dated November 21, 2001

86333; 3-2301-45 Valve Failed LLRT; dated December 12, 2001

91144; Incorrect Motor Margin Calculated by MOV Margin Review Software; dated January 15, 2002

91206; Inability to Repair U2 HPCI Control Problem; dated January 17, 2002

94868; Failure of Breaker to Trip due to Inadequate PMT; dated February 12, 2002

94868; Reactor Recirculation Pump Motor Generator failed to trip; dated April 27, 2002

95072; 2B RBCCW pump breaker racking shutter cannot completely close; dated January 13, 2002

96713; 2/3E Lift Pimp Failed to Start; dated February 26, 2002

96739; Fire Protection SA Identifies Issues with Unit Dependence; dated February 26, 2002

97594; 4KV Breaker Resistance out of spec; dated March 3, 2002

98830; 4KV Breaker 152-3401 failed to close in test position; dated March 12, 2002

98830-01; ACE for 4KV Breaker 152-3401 failed to close in test position; dated April 22, 2002

99431-71; Present to Dresden's Maintenance Rule Expert Panel Performance Monitoring Criteria; dated January 8, 2003

101056; HPCI not Properly Aligned During Reactor Startup; dated March 26, 2002

101087; Unit 2 HPCI Room Cooler Coil Tubes Have Extensive Wall Thinning; dated March 26, 2002

117222; 3A Circulating Water Pump Trip; dated July 26, 2002

126262; TR 32 to Bus 33 feed breaker would not close; dated October 8, 2002

127924; 3BRBCCW trip on start; dated October 18, 2002

130544; [U2 HPCI MGU Motor Degraded]; dated November 6, 2002

131524; 3B RBCCW Pump Breaker failed to close on demand; dated October 16, 2002

134479; 3A SDC pump trip; dated December 7, 2002

134670; 2/3 ‘E’ Lift Pump failed to start; dated December 9, 2002

136271; HPCI Inlet Drain Pot Trap Leak; dated December 17, 2002

138662; Electrolyte Residue under Battery Cell #10 on Unit 2 250VDC Battery; dated January 8, 2003

141063; HPCI Drain Pot High Level Alarm Received; dated January 25, 2003

142975; Unit 2 HPCI Turbine Inlet Drain Pot Hi Alarm; dated February 4, 2003

150798; Pin Hole Leak in D3 HPCI Suction Line from CST; dated March 26, 2003

151092-11; Perform an EACE for Unit 2/3 4KV breaker “Trip Free”; dated May 23, 2003

151969; Foreign Material found between Turbine Stop Valve and Restricting Orifice; dated February 2, 2003

157770; Failure Analysis Report for D2 HPCI MGU Removed Motor; dated May 8, 2003

160504; 2C Condensate Pump Breaker Trip Upon Start of 2C Pump; dated May 26, 2003

161223; HPCI Deficiencies Found During FASA for NRC SSDI; dated May 30, 2003

163620; HPCI Turbine Gland Seal Pressure High Alarm; dated June 17, 2003

164330; 2C Hot Canal Cooling Tower Pump Trip; dated June 22, 2003

167124; Excessive HPCI Room Cooler Vibrations; dated July 10, 2003

169994; Potential not to meet Technical Specification Surveillance; dated July 31, 2003

170031; Core Spray Potentially Inop with Test Return Valve Open; dated August 1, 2003

Modifications

008206; RPV Steam Dryer Modification; Revision 0

008238; Add C Clamp Supports to the sensing lines of the Reactor Pressure vessel jet pumps; Revision 0

330713; Install Hot Tap Valves on line 2-3769; Revision 1

333974; Lower the Unit 3 Torus Hi Level Switches 3-2351A/B and add test taps; Revision 0

333548; Modify HPCI Torus High Level Switch Piping to Lower Switches 2-2351-A(B); dated October 23, 2001

335428; Install Marathon Control Blades in the RPV and Control Rod Drive Systems; Revision 1

Design Changes

D.P. 990792; Change ESS Bus Degraded Voltage and Time Delay Settings; Revision 1

E12-2-97-201; HPCI Gland Seal Leak Off Subsystem Upgrade; dated June 12, 1998

EC 4810; Install a Varistor on the 2-2301-5 Valve; dated December 16, 1991

EC 6436; Design Change M12-2-96-006 - ECCS Suction Strainer Replacement; dated April 13, 1998

M12-3-96-006 - ECCS Suction Strainer Replacement; dated June 4, 1997

M12-2-96-006 - ECCS Suction Strainer Replacement; dated October 2, 1997

EC 6437; Design Change M12-3-96-006 - ECCS Suction Strainer Replacement; dated March 1, 1997

EC 7266; Replace Condensate Storage Tank Low Low Level Switches and Raise Setpoint; dated February 2, 2001

EC 7812; Design Change 9900471 - Install Vendor Recommended Turbine Stop Valve Stem Bushing Retaining Plate to Provide Alternate Means of Retaining the Bushing in Place and thus Prevent Steam Leaks; dated April 24, 2000

EC 7966; Design Change 9900625 - Change HPCI Steam Timer Setting Tolerance from +/- 1.0 Seconds to +/- 0.5 Seconds (As-Left Value); dated January 3, 2001

EC 330637; 4KV Bus Bracing; Revision 1

EC 333407; Change vent valve to Torus Catwalk and install temperature monitoring system; Revision 1

EC 333548; Modify HPCI Torus High Level Switch Piping to Lower Switches 2-2351-A(B); undated

EC 333933; Replace 250VDC MCC 3 Cubicles; Revision 001

EC 337829; Installation of Strainer Upstream of HPCI Turbine Drain Pot Steam Trap; Revision 0

EC 338242; Modify Opening Logic for HPCI 2-2301-8 Valve by Adding Pressure Switch; Revision N/A

EC 338243; Modify Opening Logic for HPCI 3-2301-8 valve by Adding Pressure Switch; Revision 000

Drawings

12E-2302A; Station Key Diagram 4160V and 480V SWGRS Part 1; Revision U

12E-2302B; Station Key Diagram 480V Motor Control Centers Part 2; Revision AL

12E-2303, Sheet 1; Key Diagram 4160V Switchgear 21 and 22; Revision T

12E-2303, Sheet 2; Key Diagram 4160V Switchgear 23 and 24; Revision U

12E-2303, Sheet 3; Key Diagram 4160V Switchgear 23 and 24; Revision B

12E-2304; Key Diagram 4160V Switchgears 23-1 and 24-1; Revision T

12E-2320; Key Diagram - Reactor Building 480V Motor Control Centers 29-4, 28-7 & 29-7; Revision AM

12E-2328; Single Line Diagram - Emergency Power Systems; Revision M

12E-2332; Relaying & Metering Diagram Unit Auxiliary Transformer 21 & 4160V Switchgears 21 & 22; Revision L

12E-2333; Relaying & Metering Diagram Unit Auxiliary Transformer 21 & 4160V Switchgears 23 & 24; Revision P

12E-2334; Relaying & Metering Diagram 4160V Switchgears 24-1 &23-1; Revision AA

12E-2338; Schematic Diagram Generator & Transformer Tripping Relays Primary System; Revision AK

12E-2338A; Schematic Diagram Generator & Transformer Tripping Relays Backup System; Revision AB

12E-2339; Schematic Diagram Reserve Auxliary Transformer 22 Tripping Relays; Revision T

12E-2340; Schematic Diagram 4160V Bus 21 Main & Reserve Feed A.C.B.s; Revision R

12E-2342; Schematic Diagram 4160V Bus 23 Main & Reserve Feed G.C.B.s; Revision AD

12E-2343; Schematic Diagram 4160V Bus 24 Main & Reserve Feed G.C.B.s; Revision AE

12E-2345; Schematic Diagram 4160V Bus 23-1 Undervoltage Relays; Revision AL

12E-2345; Schematic Diagram 4160V Bus 23-1 4 KV Switchgear Bus 40 Feed Breaker; Revision AV

12E-2346; Schematic Diagram 4160V Bus 24-1 Undervoltage Relays Standby Diesel Generator 2 Overvoltage Relays; Revision AK

12E-2351B, Sheet 2; Schematic Diagram Diesel Generator 2/3 Auxiliaries and Start Relays; Revision AT

12E-2499; Schematic Diagram High Pressure Coolant Injection System Process Instrumentation Part 10; Revision AA

12E-2526; Schematic Diagram High Pressure Coolant Injection System Block Diagram & Control Switch Development; Revision Y

12E-2527, Sh. 1; Schematic Diagram High Pressure Coolant Injection System Sensors and Auxiliary Relays; Revision B

12E-2527, Sh. 2; Schematic Diagram High Pressure Coolant Injection System Sensors and Auxiliary Relays; Revision C

12E-2527A; Schematic Diagram High Pressure Coolant Injection System Valve and Turbine Aux. Relays; Revision H

12E-2528; Schematic Diagram High Pressure Coolant Injection System Valves and Turbine Auxiliaries; Revision BC

12E-2529, Sh. 1; Schematic Diagram High Pressure Coolant Injection System Steam, Main Pumps & Cond. Valves; Revision AN

12E-2529, Sh. 2; Schematic Diagram High Pressure Coolant Injection System Steam, Main Pumps & Cond. Valves; Revision AN

12E-2529, Sh. 3; Schematic Diagram High Pressure Coolant Injection System Steam, Main Pumps & Condensate Valves; Revision AM

12E-2529, Sh. 4; Schematic Diagram High Pressure Coolant Injection System Steam, Main Pumps & Condensate Valves; Revision AN

12E-2530; Schematic Diagram High Press. Coolant Injection Sys Auxiliary Valves; Revision AE

12E-2532; Schematic Diagram High Pressure Coolant Injection System Turbine Auxiliary; Revision AJ

12E-2533; Schematic Diagram High Pressure Coolant Injection System Turbine Motor Gear Unit Speed Changer & Auxiliary Valves; Revision AA

12E-2655C; 4160V SWGR. Bus 23-1 Cub. 1,2,3,6 & 12 Internal; Revision J

12E-2958; Schematic Diagram Auxiliary Transformer 32 Trip Relays; Revision S

12E-3302A; Station Key Diagram 4160V and 480V SWGRS Part 1; Revision T

12E-3302B; Station Key Diagram 480V Motor Control Centers Part 2; Revision AF

12E-3303, Sheet 1; Key Diagram 4160V Switchgear 33 and 34; Revision D

12E-3303; Key Diagram 4160V Switchgears 31, 32, 33 and 34; Revision P

12E-3304; Key Diagram 4160V Switchgears 33-1 and 34-1; Revision 5

12E-3319; Key Diagram - Reactor Building 480V MCC 39-1; Revision X

12E-3345; Schematic Diagram 4160V Bus 33-1 Undervoltage Relays Control Switch Development; Revision AM

12E-3346, Sheet 2; Schematic Diagram 4160V Bus 34-1 Standby Diesel 3 FEED & 24-1 Tie Breaker; Revision AN

12E-6400C; MOV Limit Switch Development; Revision F

12E-6400E; MOV Limit Switch Development; Revision C

12E-7582E; Schematic Diagram ATWS Recirc. Pump Trip System ECCS Initiation & HPCI Turbine Trip Div II - Part 5; Revision M

165A202EF, Sheet 2; Cooler-Oil; dated March 14, 1967

M-3; General Arrangement Mezzanine Floor Plan; Revision P

M-4; General Arrangement Ground Floor Plan; Revision AB

M-5; General Arrangement Basement Floor Plan; Revision M

M-7; General Arrangement Sections “A-A” & “B-B”; Revision D

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M-1151A-4, Sheet 1; Computer Piping Isometric - High Pressure Coolant Injection; Revision 0

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M-1151B-1, Sheet 1; Blume Curve Piping Isometric - High Pressure Coolant Injection; Revision 0

M-1151B-1, Sheet 2; Blume Curve Piping Isometric - High Pressure Coolant Injection; Revision 0

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M-1151B-9, Sheet 1; Blume Curve Piping Isometric - High Pressure Coolant Injection; Revision 0

M-1187A, Sheet 1; High Pressure Coolant Injection System; Revision E

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M-1187A, Sheet 3; High Pressure Coolant Injection System Line No. 3-2318-2”-LX; Revision A

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M-1187A-4, Sheet 3; Computer Piping Isometric - High Pressure Coolant Injection; Revision 0

M-1187B-1, Sheet 1; Blume Curve Piping Isometric - High Pressure Coolant Injection; Revision 0

M-1187B-1, Sheet 2; Blume Curve Piping Isometric - High Pressure Coolant Injection; Revision 0

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M-1187B-1, Sheet 4; Blume Curve Piping Isometric - High Pressure Coolant Injection; Revision 0

M-1187B-1, Sheet 5; Blume Curve Piping Isometric - High Pressure Coolant Injection; Revision 0

M-1187B-1, Sheet 6; Blume Curve Piping Isometric - High Pressure Coolant Injection; Revision 0

M-1187B-6, Sheet 1; Blume Curve Piping Isometric - High Pressure Coolant Injection; Revision 0

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M-1187B-9, Sheet 1; Blume Curve Piping Isometric - High Pressure Coolant Injection; Revision 0

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SIL 093*; Topaz inverter cover clearance in HPCI and RCIC Systems; dated July 31, 1974

SIL 094*; HPCI & RCIC Turbine Controls; dated July 31, 1974

SIL 120*; Prevention of Gasket Failures in HPCI System Gram Gland Condensers; dated March 31, 1975

SIL 223*; HPCI Control Valve Lift Rod Bending**; dated July 8, 1977

SIL 233*; Warm-up of HPCI and/or RCIC steam Supply Lines; dated September 30, 1976

SIL 274*; HPCI Solid Wheel Turbine – Reversing Chamber Bolting**; dated May 1978

SIL 319 HPCI Generic Lubrication Problem**

SIL 336; Surveillance Test Recommendations for HPCI and RCIC; Revision 1

SIL 351; HPCI and RCIC Turbine Control System Calibration**; Revision 1

SIL 352*; HPCI Turbine Stop Valve Steam Balance Chamber Pressure Adjustment **; dated February 18, 1981

SIL 353*; Part 1,2, and 3 HPCI Turbine Mechanical Overspeed Trip**; dated February 18, 1981

SIL 360* Rev 1; Part 1 and 2, HPCI Turbine Stop Valve Hydraulic Cylinder Failure**; Rev 1; dated November 1979

SIL 392;Improved Turbine Mechanical Trip Design**; Revision 1

SIL 461 MOV Actuator Thrust

SIL 475 RCSI and HPCI High Steam Flow Analytic Limit

SIL 480 HPCI Startup Transient Improvement**

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SIL 557; Agastat relay replacement; dated April 22, 1992

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DAN 023-5-H-4; U3 HPCI PP Area Temperature HI; Revision 09

DEOP 0010-00; Guidelines for use of Dresden Emergency Operating Procedures and Severe Accident Management Guidelines; Revision 10

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DES 6700-09; Inspection and Maintenance of General Electric MC-4.76 Horizontal Drawout Metal Clad Switchgear; Revision 06

DGA-12; Partial or Complete Loss of AC Power; Revision 47

DIS 2300-09; HPCI Turbine Pressure Switch Calibration; Revision 12

DOA 2300-01; HPCI System Standby Operation; Revision 24

DOA 2300-02; HPCI Fast Start-up; Revision 28

DOA 2300-03; HPCI Local Manual Operation; Revision 11

DOA 6100-03; Aux Power Transformer Trouble; Revision 03

DOP 2300-02; HPCI System Turning Gear Operation; Revision 7

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DOP 2300-07; Filling and Draining between HPCI No 2(3) –2301-4 and MO 2(3) 2301-5; Revision 1

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DOS 2300-02; HPCI Over speed Test; Revision 29

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99212090; 24M TS Test Div I ECCS Undervoltage; dated May 3, 2002

99212772; 18M/RFL TS Test Div II ECCS UV Relays; dated October 14, 2002

99212781; D3 24M TS HPCI Steam Line Area Temp; dated October 15, 2002

99267890; Locknut Threads in Drive Sleeve Damaged, Overhaul Actuator, dated November 30, 2001

00003235; Sluggish Movement of the MGU in Manual Mode of Operation; dated May 25, 2001

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2003-0104; Replace 250 Vdc Bus 2A and 2B MCC Cubicles with Equivalent EQ/SQ Qualified Replacements; Revision 00

2003-0118; Unit 2 Jet Pump Riser Brace Repair and Vibration Mitigation Clamp Installation; Revision 00

2003-0146; DCR doe P & Ids M-51 and M-374; Revision 00

2003-0153; Radwaste Solidification Building Bridge Crane Operation; Revision 10

2003-0160; Fill and Vent of Shutdown Cooling System; Revision 23

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2003-0215; Local Leak Rate Testing of Unit 2(3) High Pressure Coolant Injection (HPCI) System Valves; Revision 03

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UFSAR Section 6.3; Emergency Core Cooling System; Revision 4

UFSAR Section 7.3.1.3; High Pressure Coolant Injection System Instrumentation and Control; Revision 2

UFSAR Section 8.3; Onsite Power Systems; Revision 5

Z6700-01; Maintenance Rule Performance Criteria Emergency 4 kV

Z6700-02; Maintenance Rule Performance Criteria Normal 4 kV

LIST OF ACRONYMS USED

ADAMS	Agency-wide Documents Access and Management System
CFR	Code of Federal Regulations
CR	Condition Report
DOP	Dresden Operating Procedure
DRS	Division of Reactor Safety
EOP	Emergency Operating Procedure
HELB	High Energy Line Break
HPCI	High Pressure Coolant Injection
IMC	Inspection Manual Chapter
IPE	Individual Plant Evaluation
JCO	Justification for Continued Operation
LOCA	Loss of Coolant Accident
MGU	Motor Gear Unit
MOV	Motor Operated Valve
NCV	Non-Cited Violation
NRC	United States Nuclear Regulatory Commission
PRA	Probabilistic Risk Analysis
SDP	Significance Determination Process
SSDI	Safety System Design Inspection
UFSAR	Updated Final Safety Analysis Report