November 16, 2006

Mr. Christopher M. Crane
President and Chief Nuclear Officer
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3, NRC EVALUATION OF CHANGES, TESTS, OR EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORT 05000237/2006012(DRS); 05000249/2006012(DRS)

Dear Mr. Crane:

On October 6, 2006, the U.S. Nuclear Regulatory Commission (NRC) completed a combined baseline inspection of the Evaluation of Changes, Tests, or Experiments and Permanent Plant Modifications at the Dresden Nuclear Power Plant. The enclosed report documents the results of the inspection which was discussed with Mr. T. Hanley and other members of your staff at the completion of the inspection on October 6, 2006.

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

Based on the results of the inspection, two NRC-identified findings of very low safety significance were identified. However, because these violations were of very low safety significance and because they were entered into your corrective action program, the NRC is treating the issues as Non-Cited Violations (NCVs) 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, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Dresden Nuclear Power Plant.

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

Sincerely,

/RA/

David E. Hills, Chief
Engineering Branch 1
Division of Reactor Safety

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

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
Assistant Attorney General
Illinois Emergency Management Agency
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
Document Control Desk - Licensing
Assistant Attorney General
Illinois Emergency Management Agency
State Liaison Officer
Chairman, Illinois Commerce Commission

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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:	05000237/2006012(DRS); 05000249/2006012(DRS)
Licensee:	Exelon Generation Company
Facility:	Dresden Nuclear Power Station, Units 2 and 3
Location:	Morris, IL 60450
Dates:	September 18, 2006 through October 6, 2006
Inspectors:	A. Dahbur, Reactor Inspector (Team Lead) C. Acosta, Reactor Inspector J. Jacobson, Senior Reactor Inspector
Approved by:	D. Hills, Chief Engineering Branch 1 Division of Reactor Safety (DRS)

SUMMARY OF FINDINGS

IR 05000237/2006012(DRS); 05000249/2006012(DRS); 09/18/2006 through 10/6/2006; Dresden Nuclear Power Plant, Units 2 and 3; Evaluation of Changes, Tests, or Experiments (10 CFR 50.59) and Permanent Plant Modifications.

The inspection covered a two week announced baseline inspection on evaluations of changes, tests, or experiments and permanent plant modifications. The inspection was conducted by three regional based engineering inspectors. Two Green Non-Cited Violations (NCVs) and one Unresolved Item were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red), using Inspection Manual Chapter 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: Initiating Events

Green. The inspectors identified a Severity Level IV NCV of 10 CFR 50.59 for the licensee's failure to perform an adequate safety evaluation review for changes made to the facility per Temporary Modification EC TCCP 354622. Specifically, the licensee failed to appropriately evaluate the installation of a temporary jumper at Electro-Hydraulic Control (EHC) Card 2-5640-A37 to bypass the function of "A" Main Steam Pressure Regulator (MSPR). The licensee's 10 CFR 50.59 safety evaluation 2005-01-001 failed to provide a basis as to why the activity to bypass one of the two MSPRs did not present more than minimal increase in the likelihood of occurrence of a malfunction of a Structures, Systems, and Components (SSC) important to safety previously evaluated in the Updated Final Safety Analysis Report (UFSAR).

Because the issue potentially impacted the NRC's ability to perform its regulatory function, this finding was evaluated using the traditional enforcement process. The finding was determined to be more than minor because the inspectors could not reasonably determine that the activity implemented per Temporary Modification EC TCCP 354622, which adversely affected systems important to safety, would not have ultimately required NRC approval. The inspectors completed a significance determination of the underlying technical issue using NRC's inspection manual chapter (IMC) 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and answered "no" to the Transient Initiator screening question in the Phase 1 Screening Worksheet. Specifically, the MSPR system does not have a mitigating function. This issue was entered into the licensee's corrective action program. (Section 1R02.1.b.1)

Cornerstone: Mitigating Systems

Green. The inspectors identified a NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control" having very low safety significance (Green) for the licensee's failure to have an adequate analysis for the 5KV cables located in the reactor building corner rooms. Specifically, the licensee failed to evaluate and include the conductor temperature rise for the 5KV cables for the Core Spray (CS) and Low Pressure Coolant Injection (LPCI) pump motors in the Equipment Qualification (EQ) Binder EQ-04D as specified in the methodology described in the EQ Binder. This issue was entered into the licensee's corrective action program to revise the EQ Binder EQ-04D to quantify the temperature rise effect.

This finding was more than minor because it affected the mitigating system cornerstone objective to ensure the availability, reliability, and capability of systems that mitigate transients and accidents. Specifically, the licensee EQ Binder EQ-04D did not consider and evaluate the affect of the conductor temperature rise to ensure that the 5KV cables for the CS and LPCI pump motors will perform their safety function during and post Design Basis Accident where ambient temperature could reach 189.4 degrees F. This finding was determined to be of very low safety significance because after identification by the inspectors, the licensee completed preliminary analysis and was able to demonstrate that the high temperature of 189.4 degrees F in the CS and LPCI corner rooms would not prevent the 5KV motor feeder cables for these pumps from performing their safety function during and post accident and the qualified thermal life for these cables still remained at 60 years. (Section 1R17.1.b.1)

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

No findings of significance were identified.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R02 Evaluations of Changes, Tests, or Experiments (71111.02)

.1 Review of 10 CFR 50.59 Evaluations and Screenings

a. Inspection Scope

From September 18 through October 6, 2006, the inspectors reviewed six evaluations performed pursuant to 10 CFR 50.59. The inspectors confirmed that the evaluations were thorough and that prior NRC approval was obtained as appropriate. The inspectors also reviewed 14 screenings where 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. The evaluations and screenings were chosen based on risk significance, safety significance, and complexity. The list of documents reviewed by the inspectors is included as an attachment to this report.

The inspectors used, in part, Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, to determine acceptability of the completed evaluations and screenings. The NEI document was endorsed by the NRC in Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," dated November 2000. The inspectors also consulted Part 9900 of the NRC Inspection Manual, "10 CFR Guidance for 10 CFR 50.59, Changes, Tests, and Experiments."

b. Findings

b.1 Inadequate Basis in 10 CFR 50.59 Evaluation for Temporary Modification

Introduction: The inspectors identified a Severity Level IV Non-Cited Violation (NCV) of 10 CFR 50.59 "Changes, Tests, and Experiments," having very low safety significance (Green) for the licensee's failure to perform an adequate safety evaluation review for changes made to the facility per Temporary Modification EC TCCP 354622. Specifically, the licensee failed to appropriately evaluate the installation of a temporary jumper at the Electro-Hydraulic Control (EHC) Card 2-5640-A37 to bypass the function of the "A" Main Steam Pressure Regulator (MSPR). The licensee's 10 CFR 50.59 safety evaluation 2005-01-001 failed to provide a basis as to why the activity which bypassed one of the two MSPRs did not present more than minimal increase in the likelihood of occurrence of a malfunction of a Structures, Systems, and Components (SSC) important to safety previously evaluated in the Updated Final Safety Analysis Report (UFSAR).

Description: UFSAR Section 7.7.4 described the Pressure Regulator and Turbine Generator Controls as follows: the pressure regulator controls turbine control valve position to maintain constant reactor pressure. A single pressure regulator, with a backup regulator, is used to position both turbine control valves and the turbine bypass valves. The backup pressure regulator was provided to control pressure in the event that the operating regulator should fail. The setpoint of the backup pressure regulator is normally biased above the setpoint of the operating pressure regulator. The pressure regulator can be assumed to fail in either of two ways, opening the turbine control valves or the bypass valves or closing them. Fuel damage does not occur in either case. The backup pressure regulator reduces the probability that pressure regulator malfunction will cause operational problems.

If the controlling pressure regulator fails as is, the effect upon the plant is dependent upon how the plant pressure behaves. The controlling regulator would be the regulator with the highest value steam flow demand. If system pressure rises, the backup pressure regulator output would rise until it takes over control of the turbine control valves. If reactor pressure drops, the reactor power will decrease, further dropping reactor pressure until the reactor vessel is isolated upon a low steam line pressure signal.

If the controlling pressure regulator fails in the direction to open the control valves, reactor pressure and power will be decreased and the transient will finally be terminated by the closing of the main steam isolation valves. This event was analyzed in UFSAR Section 15.1.3.1 (failure of either turbine pressure regulator in the valve-open direction) and was classified as a moderate frequency event.

If the controlling pressure regulator fails in the direction to close the control valves the effect would be quite similar to the positive pressure setpoint change as the backup pressure regulator would then take over control of reactor pressure. Similar analysis for this event was also discussed in UFSAR Section 15.2.1 and was also classified as a moderate frequency event.

In March 2005, the licensee installed a switch in panel 902-31 to defeat the “A” Main Steam Pressure Regulator circuit and remove it from the diode action circuit per Temporary Modification EC-TCCP 354622. Closure of this switch will place a ground on the Unit 2 EHC Steam Line Resonance Compensator (SLRC) circuit card A37 causing the “B” Main Steam Pressure Regulator circuit to be in control. When this switch is closed only the “B” MSPR circuit will be able to respond to plant parameter changes effectively removing the “A” MSPR circuit. Typically, the “A” MSPR circuit is in control and the “B” MSPR circuit is biased slightly lower. This temporary modification was installed to allow plant startup while the licensee completed their investigation and troubleshooting of the “A” MSPR circuit which was found to be a suspected cause of previous reactor trip (SCRAM). The licensee completed a 10 CFR 50.59 safety evaluation 2005-01-001 in March 2005. The evaluation concluded that this activity per the temporary modification can be implemented per plant procedures without obtaining a License Amendment.

During the inspectors review of the licensee's safety evaluation 2005-01-001, the inspectors questioned the basis for the answer to Question 2 of the evaluation, which provided the basis as to why the activity to bypass one of the two MSPRs did not present more than minimal increase in the likelihood of an occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR. The evaluation stated, "the EHC system and in particular the MSPR circuits and functions are not safety related and are not redundant but are important to safety. The failure of the pressure regulator function has been analyzed in sections 15.1.3.1 and 15.2.1 of the UFSAR and the Core Operating Limits Report (COLR) analyses are the bounding conditions."

Because UFSAR Section 15.2.1 credited the backup regulator to take control of the turbine valves as soon as the failed controlling regulator attempted to close the valves, the inspectors determined that the licensee's conclusion stated in the answer to Question 2 above was not adequate. Specifically, the inspectors determined that the condition of "B" MSPR failing low while the "A" MSPR bypassed, was not bounded by the condition described in UFSAR Section 15.2.1. The failure of one MSPR, within the pressure regulator system, low would not necessarily result in a reactor trip since the backup pressure regulator would take control. However, by changing the facility to allow one pressure regulator to be permanently bypassed, the failure of one MSPR low would have resulted in a reactor trip.

Although, operating with a pressure regulator out-of-service was analyzed in the Core Operating Limits Report (COLR) with certain limitations and penalties to be applied to ensure the integrity of the fuel was not damaged during the transient, the inspectors determined that the licensee inappropriately credited the COLR analysis as the basis to the answer to question 2 in the evaluation. Neither the COLR nor the safety evaluation had evaluated if the activity to bypass one of the MSPRs did not present more than minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in UFSAR Section 15.2.1.

The licensee entered this issue into the station's corrective action program as Assignment Report (AR) 00540443 to address the inspectors' concern and determine additional actions as necessary. The licensee also submitted additional information after the exit meeting for inspectors to review. The licensee stated, in part, that although the 10 CFR 50.59 safety evaluation was correct in reaching the conclusion that prior NRC approval for the system effect was not required, Dresden has since determined that the appropriate process for evaluating this plant change was the applicability review process. The activity per the temporary modification was governed by the COLR, and thus falls within programs controlled by the Operating License or the Technical Specifications (in this case TS 5.6.5, which controls the COLR). The inspectors reviewed the information submitted by the licensee and disagreed with this position in that while an evaluation was required and the COLR ensured the integrity of the fuel was not damaged during the transient, the COLR did not evaluate the effect of this activity with respect to 10 CFR 50.59 requirements. Specifically, the COLR did not evaluate if the activity to bypass one of the MSPRs did not present more than minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in UFSAR Section 15.2.1.

Analysis: The inspectors determined that this issue was a performance deficiency warranting a significance evaluation, because in March 2005, the licensee failed to provide an adequate basis for changes made to the facility in accordance with 10 CFR 50.59. Specifically, the licensee failed to provide a basis as to why changes made per Temporary Modification EC TCCP 354622 did not present more than minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety (Main Steam Pressure Regulator System). The finding was determined to be more than minor because the inspectors could not reasonably determine that the activity to temporarily bypass the function for one of the two MSPRs would not have ultimately required NRC prior approval.

Because violations of 10 CFR 50.59 are considered to be violations that potentially impede or impact the regulatory process, they are dispositioned using the traditional enforcement process instead of the significance determination process (SDP). However, if possible, the underlying technical issue is evaluated under the SDP to determine the severity of the violation. In this case, the inspectors completed a significance determination of the underlying technical issue using NRC's inspection manual chapter (IMC) 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and answered "no" to the Transient Initiator screening question in the Phase 1 Screening Worksheet which stated, "Does the finding contribute to both the likelihood of reactor trip AND the likelihood that mitigating equipment or function will not be available?" Specifically, the MSPR system did not have a mitigating function. Based upon this Phase 1 screening, the inspectors concluded that the issue was of very low safety significance (Green). In accordance with the Enforcement Policy, the violation was therefore classified as a Severity Level IV violation.

Enforcement: Title 10 CFR 50.59(d)(1) states, in part, that the licensee shall maintain records of changes in the facility, of changes in procedures, and of tests and experiments as described in the UFSAR. These records must include a written evaluation which provides a basis for the determination that the change, test, or experiment does not require a license amendment.

Contrary to the above, the licensee, in March 2005, in their safety evaluation, 2005-01-001, the licensee failed to provide an adequate basis for the determination that the change to the facility per Temporary Modification EC TCCP 354622 was acceptable without a licensee amendment. Specifically, the licensee failed to provide basis to why the installation of a temporary jumper to bypass the function of the "A" MSPR (equipment important to safety) did not present more than a minimal increase in the likelihood of occurrence of a malfunction of SSC important to safety. In accordance with the Enforcement Policy, this violation of the requirements of 10 CFR 50.59 was classified as a Severity Level IV Violation because the underlying technical issue was of very low safety significance. Because this non-willful violation was non-repetitive, and was captured in the licensee's corrective action program (CAP), it is considered a NCV consistent with VI.A.1 of the NRC Enforcement Policy. (NCV 05000237/2006012-01; 05000249/2006012-01 (DRS)) b.2 Change of Systems Credited to Mitigate a High Pressure Coolant Injection (HPCI) Pump Room High Energy Line Break (HELB)

Introduction: The inspectors identified an Unresolved Item (URI) involving the adequacy of a 10 CFR 50.59 safety evaluation for UFSAR changes that the licensee had implemented. Specifically, the inspectors questioned the adequacy of the licensee's basis for determining that changes to UFSAR Section 3.6.1.5, "Use of Isolation Condenser and Control Rod Drive System for Safe Shutdown Following a HELB" did not require a license amendment. This issue is unresolved pending further NRC review of Dresden's licensing basis for mitigating an HELB in the HPCI pump room.

Description: The licensee completed a 10 CFR 50.59 safety evaluation to support a change to the systems credited with safe shutdown following a HPCI pump room HELB from strictly safety related systems to a combination of safety-related and non-safety-related systems. Specifically, the licensee credited the isolation condenser for decay heat removal in lieu of the Automatic Depressurization System (ADS) and Low Pressure Coolant Injection (LPCI)/Containment Cooling Service Water (CCSW); and credited the control rod drive system for control of reactor coolant inventory in lieu of LPCI. The reason for this change as stated by the licensee, was that the original HELB analysis did not evaluate the affects of a HPCI steam line break in the HPCI room on plant equipment in other areas and that the Environmental Qualification (EQ) program only evaluated the equipment in the HPCI room that was needed to isolate the break. The licensee determined that the use of non-safety-related equipment for safe shutdown was made necessary because the postulated break in the HPCI room resulted in a harsh environment in the Unit 2/3 Diesel Generator (D/G) Room. Since the D/G was not environmentally qualified for this environment, it was assumed to fail as a consequence of the HELB. To meet single failure requirements, the Unit 2 D/G was assumed to also fail. This resulted in no AC power available to power the LPCI or CCSW systems. Therefore, the original HELB analysis did not reflect the possibility that the ADS and LPCI would not be available to mitigate the HPCI room HELB.

On August 10, 2005, the licensee completed 10 CFR 50.59 safety evaluation 2005-02-001 and concluded that a License Amendment was not required. Specifically, the licensee determined that the NRC addressed the issue of using the isolation condenser and control rod drive systems in lieu of safety-related systems for safe shutdown following a HELB for other areas during the Systematic Evaluation Program. This implied to the licensee that the use of the isolation condenser for decay heat removal and the control rod drive for reactor coolant inventory control did not result in more than a minimal increase in the likelihood of a malfunction of an SSC important to safety.

The inspectors reviewed the 10 CFR 50.59 evaluation (2005-02-001) and questioned the changes that were made to the UFSAR. The inspectors were concerned about the adequacy of the licensee's basis, as stated in the 10 CFR 50.59 safety evaluation, for determining that changes to UFSAR Sections 3.6.1.5 "Use of Isolation Condenser and Control Rod Drive System for Safe Shutdown Following a HELB" did not require a licensee amendment. More specifically, the inspectors were concerned about the following items.

Updated to reflect the changes, Section 3.6.1.5 of Dresden's UFSAR described the licensees response concerning a potential main feed system line break in the main feed regulatory valve area. It states that a main feed system line break closely parallels a HPCI HELB in the HPCI room but it did not state what were the differences of these two breaks.

The 10 CFR 50.59 safety evaluation did not provide appropriate information as to whether or not the proposed activity resulted in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR. Specifically, the licensee did not evaluate why the activity did not result in more than a minimal increase in the likelihood of occurrence of a malfunction of LPCI and ADS. In addition, the evaluation also failed to provide appropriate information as to whether the activity created a possibility for an accident of a different type than previously evaluated in the UFSAR. The evaluation did not address the loss of coolant inventory resulting from the HPCI break.

Following identification of this issue, the licensee entered the issue into their corrective action program as AR 534275. This issue is unresolved pending further NRC review of Dresden's licensing basis for a high energy line break in the high pressure coolant injection pump room to determine if the licensee had adequately analyzed the consequences of this accident. (URI 05000237/2006012-02; 05000249/2006012-02 (DRS))

1R17 Permanent Plant Modifications (71111.17B)

.1 Review of Permanent Plant Modifications

a. Inspection Scope

From September 18 through October 6, 2006, the inspectors reviewed seven permanent plant modifications that had been installed in the plant during the last two years. The modifications were chosen based upon risk significance, safety significance, and complexity. As per inspection procedure 71111.17B, one modification was chosen that affected the barrier integrity cornerstone. 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 adversely 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.

The inspectors also used applicable industry standards to evaluate acceptability of the modifications. The list of modifications and other documents reviewed by the inspectors is included as an attachment to this report.

b. Findings

b.1 EQ Binder Failed to Include Conductor Temperature Rise

Introduction: The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control” having very low safety significance (Green) for the licensee’s failure to evaluate and include the conductor temperature rise for the 5KV cables for the CS and LPCI pump motors in the Equipment Qualification Binder EQ-04D. The EQ Binder used the cable design limit of 194 degrees F in calculating the qualified life of the 5KV cables instead of the sum of the conductor temperature rise and the ambient temperature, during and post accident, which together exceeded the cable design limit.

Description: During the inspectors’ review of Engineering Change EC-356001 “LPCI Pump Discharge ADS Permissive Pressure Switch Replacement,” the inspectors observed that the calculated design basis temperature in the Reactor Building, LPCI/CS corner rooms during and post accident was changed from 131 degrees F to 189.4 degrees F. The source of the 131 degrees F was based upon calculation DRE97-0214, Revision 0 “Reactor Building Post-LOCA Temperature Analysis,” which assumed that the LPCI/CS pump room emergency coolers are operating. The source of the 189.4 degrees F was based upon calculation DRE01-0041, Revision 1 “Updated EQ Zone Parameter Tables Following Implementation of Extended Power Uprate.” This value was based upon General Electric (GE) Report T0610, Revision 1 “Power-Dependant HVAC,” which assumed that the LPCI/CS pump room emergency coolers were not operating during and post accident conditions. The inspectors were concerned about the qualification of all equipment important to safety including the cables in these areas. The inspectors questioned the licensee whether safety-related equipment located in these areas had been evaluated to ensure increased temperature would not affect equipment qualification. The licensee’s response indicated that all equipment important to safety were qualified for the Design Basis Accident temperature of 189.4 degrees F.

During the inspectors’ review of EQ Binder EQ-04D, Revision 10 for the General Electric Butyl Rubber Insulated 5KV cables, the inspectors noticed the following:

- Table 4-1 “Listing of 5KV Cables” showed that the CS pump motor feeds were 75.16 percent loaded and the LPCI pump motor feeds were 81.9 percent loaded;

- Table 6-2 “Accident/Post-Accident Environmental Parameters” showed that the peak temperature in the Reactor Building CS/LPCI corner rooms could reach 189.4 degrees F for LOCA;

- Section 7.2 “Additional Margin” stated, that a 15 degrees F margin will be considered to account for uncertainties;

- Section 7.5 “Qualification Profiles for 5KV Cables” stated, that the qualification temperatures are the sum of the ambient temperature, conductor temperature rise, and 15 degrees F margin, but shall not exceed the design limit of 194 degrees F;

- Section 7.5 also stated in part, that the pertinent qualification profile for the CS/LPCI motor feeds was 194 degrees F (the design limit of the cables) for 1.05 years, which included the 292-day portion of the post-LOCA period, and 119 degrees F for 59.75 years;

- Section 14.1.1 “Thermal Aging for 5 KV Cables” indicated that cable specimens were aged as follows, 95 degrees C for 100 hours, 110 degrees C for 85 hours and 125 degrees C for 531 hours. Using the Arrhenius Equation, the thermal aging was equivalent to 497.9 years at 119 degrees F; and

- Table 14-2 “Equivalent Duration of Required Qualification Profile (5KV Cables)” and Section 14.1.3 showed that equivalent duration at 119 degrees F for the 5KV CS/LPCI motor feeder cables was 280.1 years. Therefore, qualified life of the 5KV CS/LPCI motor feeder cables was 60 years.

Based on the review of the EQ Binder EQ-04D, the inspectors determined that since the full current loading of the CS/LPCI feeder cables were 75.16 percent and 81.9 percent respectively, which would result in considerable conductor temperature rise, and since the peak rooms temperature could reach up to 189.4 degrees F during and post-LOCA, the temperature value of 194 degrees F used in the calculated qualified life for these cables was inadequate and not conservative. The sum of conductor temperature rise and the rooms ambient temperature during and post accident conditions could exceed 194 degrees F. The licensee also failed to evaluate the potential consequences of adverse effects on these cables due to higher temperatures which could exceed the cables design temperature limits and shorten the qualified life of these cables.

After identification by the inspectors, the licensee completed a preliminary evaluation/calculation which showed that the CS/LPCI cable loading listed in EQ Binder EQ-04D was based on a conservative standard reference for current carrying capacity of 75 degrees C cables. The actual design ampacity of the cables from SLICE program (Dresden’s software program for cable management) was correctly found to be higher due to the fact that the cables are actually rated for 90 degrees C. The CS pumps’ feeder cables were found to be 47.2 percent loaded and the LPCI cables were found to be 51.5 percent loaded instead of 75.16 percent and 81.9 percent, respectively. The licensee’s evaluation showed that the maximum temperature that the CS/LPCI 5KV cables will be exposed to during/post-LOCA in the CS/LPCI corner room was 213.14 degrees F. This temperature value included the conductor temperature rise but did not include the 15 degrees F margin. The licensee’s evaluation concluded that the tested profile enveloped the peak accident temperature and thermally enveloped only a portion of the plant post accident duration. The post accident duration was justified by using a portion of the accelerated aging. The thermal life of the CS/LPCI pump motor feeder cables still remained at 60 years.

The licensee entered this issue into the station’s corrective action program as AR 00539590 to revise the EQ Binder EQ-04D to incorporate the licensee’s preliminary evaluation results.

<u>Analysis</u>: The inspectors determined that this issue was a performance deficiency warranting a significance evaluation, since the licensee failed to evaluate and include the conductor temperature rise in the EQ Binder for the 5KV cables. The inspectors concluded that the finding was greater than minor in accordance with IMC 0612,

Appendix B, “Issue Screening,” because it affected the mitigating system cornerstone objective to ensure the availability, reliability, and capability of systems that mitigate transients and accidents. Specifically, the licensee EQ Binder EQ-04D did not consider and evaluate the affect of the conductor temperature rise to ensure that the 5KV cables for the CS and LPCI pump motors will perform their safety function during and post Design Basis Accident. In addition, the licensee failed to evaluate the potential consequences of adverse effects on these cables due to higher temperatures which could exceed the cables’ design temperature limits. Until the licensee completed a preliminary analysis which demonstrated that the cables will perform their design functions, the inspectors had reasonable operability concerns.

The inspectors determined that the finding was of very low significance (Green) using IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for the At-Power Situations,” because the inspectors answered “no” to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. In particular, the licensee’s preliminary analysis demonstrated that the high temperature of 189.4 degrees F in the CS and LPCI corner rooms would not prevent the 5KV motor feeder cables from performing their safety function during and post accident and the qualified thermal life for these cables still remained at 60 years.

Enforcement: Criterion III of 10 CFR Part 50, Appendix B, requires, in part, that measures shall be established to assure that applicable design bases are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above, on October 3, 2006, the inspectors found that the licensee’s EQ Binder, EQ-04D Revision 10, had not correctly calculated the qualified thermal life for the 5KV feeder cables. Specifically, the licensee used the cable design limit of 194 degrees F in calculating the qualified life of the 5KV cables instead of the sum of the conductor temperature rise and the ambient temperature, during and post accident, which together exceeded the cable design limit. Once identified, the licensee entered the finding into its corrective action program as AR 00538590. 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 an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000237/2006012-03; 05000249/2006012-03 (DRS))

4. OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems

.1 Routine Review of Condition Reports

a. Inspection Scope

From September 18 through October 6, 2006, the inspectors reviewed five Corrective Action Process documents that identified or were related to 10 CFR 50.59 evaluations and permanent plant modifications. The inspectors reviewed these documents to evaluate the effectiveness of corrective actions related to permanent plant modifications and evaluations for changes, tests, or experiments issues. In addition, corrective action documents written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problems into the corrective action system. The specific corrective action documents that were sampled and reviewed by the team are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

4. **OTHER ACTIVITIES**

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. T. Hanley and others of the licensee's staff, on October 6, 2006. Licensee personnel acknowledged the inspection results presented. Licensee personnel were asked to identify any documents, materials, or information provided during the inspection that were considered proprietary. No proprietary information was identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

<u>Licensee</u>

T. Hanley, Director, Engineering
D. Galanis, Manager, Design Engineering
J. Ellis, Manager, Regulatory Assurance
J. Griffin, Regulatory Assurance
D. Knox, Design Engineering
J. Strasser, Design Engineering

<u>Nuclear Regulatory Commission</u>

C. Phillips, Senior Resident Inspector
D. Hills, EB1 Branch Chief

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000237/2006012-02; 05000249/2006012-02	URI	Change of Systems Credited to Mitigate a High Pressure Coolant Injection Pump Room High Energy Line Break

Opened and Closed

05000237/2006012-01; 05000249/2006012-01	NCV	Inappropriate Basis in 10 CFR 50.59 Evaluation for Temporary Modification
05000237/2006012-03; 05000249/2006012-03	NCV	EQ Binder Failed to Include Conductor Temperature Rise

Discussed

None.

LIST OF DOCUMENTS REVIEWED

The following is a list of licensee documents reviewed during the inspection, including documents prepared by others for the licensee. Inclusion on this 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 in this list does not imply NRC acceptance of the document, unless specifically stated in the inspection report.

IR02 Evaluation of Changes, Tests, or Experiments (71111.02)

10 CFR 50.59 Screenings

Screen 2004-0478; EC-331153 Revise Unit 3 Emergency Diesel Generator (EDG) Governor Solenoid to "Energize To Shutdown"; Revision 0

Screen 2005-0049; Temporarily Isolate U2 (U3) CCSW Pump Vault Cooling Coil 2-5700-30B (3-5700-30A); Revision

Screen 2005-0050; Modify Reactor Building Ventilation Exhaust Fan Backdraft Dampers and Actuator - Unit 2; Revision 0

Screen 2005-0051; Install New Pipe Supports and Support Modification on Unit 2 Steam Supply Vent Lines; Revision 0

Screen 2005-0068; Service Water System Operation; Revision 0

Screen 2005-0171; Evaluation of 125V Battery Capacity and System Voltage Without Crediting Manual Load Shed Reactor Building Loads/Baseline Calculation for 125 VDC ELMS-DC Conversion to DCSDM; Revision 0

Screen 2005-0197; Service water Operation; Revision 0

Screen 2005-0227; Revise TRM 3.3.d Based on License Amendments 213 and 205; Revision 0

Screen 2005-0448; Technical Specifications Change Request 05-009; Revision 0

Screen 2006-0146; Upgrade Standby Liquid Heat Trace Controls and Alarms; Revision 0

Screen 2006-0188; UFSAR Section 8.3.2.1; Revision 0

Screen 2006-0197; 4KV Bus DC Power Failure; Revision 0

Screen 2006-0219; Technical Specification 3.8.7 and 3.8.8 Bases Change Request; Revision 0

Screen 2006-0229; Change Setpoint on the SBO DG Starting Air Compressors - Unit 2 and 3; Revision 0

10 CFR 50.59 Evaluations

2005-01-001; Install Temporary Jumper on A37 Steam Line Resonance Compensator Card; Revision 0

2005-02-001; Crediting the Isolation Condenser and Control Rod Drive Systems for Safe Shutdown; Revision 0

2005-03-007; Core Spray Performance Requirement, Design Feature and Procedure Change Due to CS Line Crack; Revision 0

2005-03-006; Unit 2 Condensate Prefilter Bypass Valve Throttling; Revision 0

2005-04-001; Unit 3 Condensate Prefilter Bypass Valve Throttling; Revision 0

2006-01-001; Reactor Building and Replacement Siding and Permanent Enclosure - Unit 3 SDR; Revision 0

IR17 Permanent Plant Modifications (71111.17B)

Modifications

EC 345836; Replacement and Upgrade of 3A Service Water Pump (3-3901-A); Revision 0

EC 347742; Replace Diesel Generator Cooling Water Pump and Motor with New Pump and Motor -Unit 2; Revision 0

EC 350136; Change B-10 Enrichment in Standby Liquid Control Tank - Unit 3; Revision 1

EC 353531; 480 MCC Breaker Bucket Replacement - Unit 3; Revision 0

EC 356001; 2-1554-H and J LPCI Pump Discharge ADS Permissive Pressure Switch Replacement; Revision 0

EC 360423; Upgrade Standby Liquid Heat Trace Controls and Alarm - Unit 3; Revision 1

EC 361010; Change Valves 2(3)-1301-500 From Locked Open to Close; Revision 0

Other Documents Reviewed During Inspection

Corrective Action Program Documents Generated As a Result of Inspection

AR 00533293; Error Made During Incorporation of UFSAR Change 03-013; September 19, 2006

AR 00534275; 1982 NRC Safety Evaluation on HELB Inconsistent with Submittals; September 21, 2006

AR 00537576; Enhancement Opportunity to Demonstrate Margin at High Temperatures; September 28, 2006

AR 00539578; Leak Identified from Cap of 2-1105-B Relief Valve; October 03, 2006

AR 00539590; EQ Binder 5KV Cable Qualified Life; October 03, 2006

AR 00539702; Ladder Found Standing and Unattended; October 10, 2006

AR 00539820; Evaluate Revision to DAN 902(3)-8 F-1 (125 VDC at 4KV SWGR); October 04, 2006

AR 00540443; NRC Questions Conclusions of Safety Evaluation 2005-01-001; October 05, 2006

AR 00540534; NRC Question Raised During Walkdown (Curbing in 2/3 EDG Room); October 05, 2006

AR 00540786; NRC Questions Conclusion of Safety Evaluation 2005-02-001; dated October 06, 2006

Corrective Action Program Documents Reviewed During the Inspection

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

AR 376956; Procedure Compliance Issue During U2 SBLC Pump Testing; date September 23, 2005

AR 394386; 2A SBLC Pump Relief Valve Leak; dated November 3, 2005

AR 429395; Dried Boron Deposits Found Under SBLC Relief Valve; dated December 1, 2005

AR 459900; SBLC Relief Valve Margin Problem in Optima2 Transition; dated February 28, 2006

AR 487350; Standby Liquid Control Valves Do Not Match Calculation; dated May 5, 2006

Calculations

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

DRE98-0128; Cold Shutdown Boron Weight (CSBW) Worksheet, Part 1; Revision 2

DRE98-0129; Hot Shutdown Boron Weight (HSBW) Worksheet, Part 2; Revision 1

DRE98-0197; Standby Liquid Control Tank Boron Injection Volume; Revision 4

DRE01-0066; Dresden Unit 2 and 3 Standby Liquid Control System Discharge Piping Pressure Drop; Revision 2

DRE05-0009; Shutdown Boron Capability for the Standby Liquid Control Systems; Revision 1

NED-M-MSD-2; Revised Sodium Pentaborate Requirements for the SLCS Dresden 2 and 3 Quad Cities 1 and 2; Revision 0

MAD NO. 81-568; CRD Pump Crosstie; dated August 9, 1981

VV-14; Calculation for CCSW Cooler Performance and Effectiveness Curve Essential Calc; Revision 1A

Drawings

12E-2328; Single Line Diagram Emergency Power System; Revision N

12E-3350A Sheet 1; Schematic Diagram Standby Diesel Generator Engine Control and Generator Excitation; Revision AH

12E-2322B; Overall Key Diagram, 125 VDC Distribution Centers; Revision K

Curve AA-73547; Pump Curve for GPS-75L-3S; dated December 21, 2004

M-28; Diagram of Isolation Condenser Piping; dated May 28, 2003

M-29; Sheet 2; Diagram of Low Pressure Coolant Injection Piping

M-33; Diagram of Standby Liquid Control Piping; dated July 17, 2003

Procedures

DAN 902(3)-8 F-1; 4KV Bus DC Power Failure; Revision 07

DCP 2119-02; Standby Liquid Control; Revision 13

DEOP 0500-01; Alternate Standby Liquid Control Injection; Revision 11

DEOP 0500-03; Alternate Water Injection Systems; Revision 16

DOA 2300-03; High Pressure Coolant Injection Systems Local Manual Operation; Revision 11

DOA 6900-02; Failure of Unit 2 125VDC Power Supply; Revision 17

DOP 1100-04; SBLC Storage Tank Chemical Addition; Revision 8

DOP 6900-08; Unit 2 125VDC Battery System Restoration; Revision 07

DR-055-M-001; P/T Response Following a HPCI Steamline Break in the HPCI Room; dated August 2, 1994

Miscellaneous Documents

WO 00712416; D2/3 18M Tstr Fire Door/Oil Spill Barrier Surveillance; dated December 2, 2005

WO 99116251; Replace Pump With Vendor Upgraded Pump and Remove Oiler; dated January 23, 2006

LIST OF ACRONYMS USED

AC	Alternate Current
ADAMS	Agency-Wide Document Access and Management System
ADS	Automatic depressurization System
AR	Assignment Report
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CCSW	Component Cooling Service Water
COLR	Core Operating Limits Report
D/G	Diesel Generator
DAN	Dresden Annunciator
DOA	Dresden Operation Abnormal
DOP	Dresden Operation Procedure
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
EDG	Emergency Diesel Generator
EHC	Electro-Hydraulic Control
ELMS	Electrical Load Monitoring System
EQ	Environmental Qualification
F	Fahrenheit
GE	General Electric
HELB	High Energy Line Break
HPCI	High Pressure Coolant Injection
IMC	Inspection Manual Chapter
IR	Inspection Report
LOCA	Loss of Coolant Accident
LPCI	Low Pressure Coolant Injection
MSPR	Main Steam Pressure Regulator
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records
SDP	Significance Determination Process
SLRC	Steamline Resonance Compensation
SRA	Senior Risk Analyst
SWGR	Switchgear
TRM	Technical Requirements Manual
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item