February 3, 2006

EA-05-232

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 INSPECTION REPORT 05000237/2005014; 05000249/2005014
PRELIMINARY WHITE FINDING

Dear Mr. Crane:

On January 4, 2006, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Dresden Nuclear Power Station, Units 2 and 3. This inspection was to follow up on the resultant inoperability of the high pressure coolant injection system following a reactor scram on Unit 3. The preliminary results of this followup inspection were discussed on January 4, 2006, with the Site Vice President, Mr. Danny Bost, and other members of your staff.

The inspection examined activities conducted under your licenses as they relate to safety and to compliance with the Commission's rules and regulations and with the conditions of your licenses. Specifically, this followup inspection focused on the inoperable condition of the high pressure coolant injection system following a scram event which was identified by your staff on February 1, 2004, and your subsequent corrective actions. The inspectors reviewed selected procedures and records, observed activities, and interviewed station personnel.

This report documents one finding that appears to have low to moderate safety significance. As described in Section 4OA3 of this report, the finding pertains to your staffs' failure to properly verify the adequacy of the extended power uprate design implementation to respond to changes in post-scram reactor vessel water level to prevent water intrusion into the high pressure coolant injection system turbine steam supply line. The finding applies to both units and was assessed based on the best available information, including influential assumptions, using the applicable Significance Determination Process (SDP) and was preliminarily determined to be a White finding (low to moderate safety significance) following the performance of a case-specific Phase 3 SDP evaluation. The final resolution of this finding will convey the increment in importance to safety by assigning the corresponding color, i.e., (White), a finding with some increased importance to safety; which may require additional NRC inspection. This inoperable condition with the high pressure coolant injection system was reported to the NRC on February 1, 2004, via the Emergency Notification System (ENS) and in accordance with 10 CFR 50.73(a)(2)(v)(D) as Licensee Event Report 2004-002-00 on March 30, 2004.

On January 30, 2004, at approximately 11:55 a.m. with Unit 3 at 97 percent power, a reactor scram occurred due to a turbine trip. The licensee determined that the turbine had tripped on low lube oil header pressure while station personnel were swapping the main turbine lube oil coolers. As a result of the scram, reactor vessel water level reached and entered the high pressure coolant injection system turbine steam supply line. Your staff subsequently determined that approximately 60 gallons of water had entered the high pressure coolant injection system turbine steam supply line rendering the system inoperable because the system was not designed to operate with any amount of water.

As immediate corrective action in response to this issue, your staff modified the feedwater level control system post-scram level setpoints to ensure reactor vessel water level would not reach the high pressure coolant injection system turbine steam supply line. Additional corrective actions taken by your staff to ultimately resolve this issue included dynamic modeling of the reactor vessel level response and more post-scram reactor vessel level setpoint changes to the feedwater level control system.

This finding also involved an apparent violation of 10 CFR Part 50, Appendix B, Criterion III, and is being considered for escalated enforcement action in accordance with the Enforcement Policy. The current Enforcement Policy is included on the NRC's Web site at http://www.nrc.gov/reading-rm/adams.html

In accordance with Inspection Manual Chapter (IMC) 0609, we intend to complete our evaluation using the best available information and issue our final determination of safety significance within 90 days of this letter.

The significance determination process encourages an open dialogue between the staff and the licensee, however the dialogue should not impact the timeliness of the staff's final determination. Before we make a final decision on this matter, we are providing you an opportunity (1) to present to the NRC your perspectives on the facts and assumptions, used by the NRC to arrive at the finding and its significance, at a Regulatory Conference; or (2) submit your position on the finding to the NRC in writing. If you request a Regulatory Conference, it should be held within 30 days of the receipt of this letter and we encourage you to submit supporting documentation on the docket at least one week prior to the conference in an effort to make the conference more efficient and effective. If a Regulatory Conference is held, it will be open for public observation. If you decide to submit only a written response, such submittal should be sent to the NRC within 30 days of the receipt of this letter.

Please contact Mark Ring at 630-829-9703 within 10 business days of the date of receipt of this letter to notify the NRC of your intentions. If we have not heard from you within 10 days, we will continue with our significance determination and enforcement decision and you will be advised via separate correspondence of the results of our deliberations on this matter.

Since the NRC has not made a final determination in this matter, no Notice of Violation is being issued for the inspection finding at this time. In addition, please be advised that the characterization of the apparent violation described in this letter may change as a result of further NRC review.

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

Mark A. Satorius, Director
Division of Reactor Projects

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

Enclosure: Inspection Report 05000237/2005014; 05000249/2005014
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
Assistant Attorney General
Illinois Emergency Management Agency
State Liaison Officer
Chairman, Illinois Commerce Commission

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Mark A. Satorius, Director
Division of Reactor Projects

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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
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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:	05000237/2005014; 05000249/2005014
Licensee:	Exelon Generation Company
Facility:	Dresden Nuclear Power Station, Units 2 and 3
Location:	Morris, IL 60450
Dates:	July 1, 2005, through January 4, 2006
Inspectors:	D. Smith, Senior Resident Inspector D. Passehl, Senior Reactor Analyst L. Kozak, Senior Reactor Analyst
Approved by:	M. Ring, Chief Branch 1 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000237/2005014; IR 05000249/2005014; 07/01/2005 - 01/04/2006; Exelon Generation Company, Dresden Nuclear Power Station, Units 2 and 3; Event Followup.

The report covered the followup inspection activities for an unresolved item and a licensee event report regarding the inoperability of the high pressure coolant injection system due to water intrusion into the turbine steam supply line. The inspection was conducted by the resident inspectors and the regional senior risk analysts. The inspection identified one preliminary White finding and associated apparent violation (AV). The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). 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

Preliminary White. An apparent violation (AV) of 10 CFR 50, Appendix B, Criterion III, Design Control, having a preliminary low to moderate safety significance (White) was identified as a result of the inspectors' review of a January 30, 2004, scram event. Water intrusion into the high pressure coolant injection (HPCI) system turbine steam supply line occurred as a result of the scram and rendered the HPCI system inoperable. The inspectors determined that the licensee implemented extended power uprates on Unit 2 in 2001 and Unit 3 in 2002, but failed to verify the adequacy of design for the implementation of extended power uprate to respond to changes in post-scram reactor vessel water level to prevent water intrusion into the HPCI steam supply line.

The finding was determined to be greater than minor because it impacted the mitigating systems cornerstone. The finding was preliminarily determined to be of low to moderate safety significance following the performance of a case-specific Phase 3 SDP evaluation. Corrective actions taken by the licensee included modifying the feedwater level control system post-scram level setpoints and dynamic modeling of the reactor vessel level response. (Section 4OA3)

B. Licensee-Identified Violations

No violations of significance were identified

REPORT DETAILS

4. OTHER ACTIVITIES

Cornerstone: Mitigating Systems

4OA3 Event Followup (71153)

(Closed) Unresolved Item (URI) 0500237/2004002-02; 05000249/2004002-02: Water intrusion in the high pressure coolant injection system steam supply line

(Closed) Licensee Event Report (LER) 05000249/2004-002-00: Unit 3 Automatic Scram Due to Main Turbine Low Oil Pressure Trip and Subsequent Discovery of Inoperability of the Units 2 and 3 High Pressure Coolant Injection Systems

a. Inspection Scope

On January 30, 2004, Unit 3 experienced a scram. As a result of the scram, the feedwater level control system's response was unable to prevent reactor vessel water level from raising to the elevation where water entered the high pressure coolant injection (HPCI) system turbine steam supply line. Following the scram the HPCI system inlet drain pot level high alarm was received and did not clear for approximately 20 minutes. The licensee performed a calculation, based on the time the alarm was locked-in, which determined that approximately 60 gallons of water had entered the HPCI system steam supply line. Therefore, on February 1, 2004, the licensee declared the HPCI system on both units inoperable because the system was not designed to operate with any amount of water in the steam supply line.

The review and closure of this URI and LER constituted a single inspection sample.

b. Findings

Introduction

A finding, associated with an apparent violation (AV), with preliminary low to moderate safety significance (White) was identified following review of an URI and LER. The issue involved a January 30, 2004, scram on Unit 3 which resulted in reactor vessel water level raising to the elevation of the HPCI system turbine steam supply line. The water reached this level in the vessel, due to the feedwater control system's response to the scram, and rendered the HPCI system inoperable because the system's turbine was not designed to operate with any amount of water in the system. The inspectors identified an apparent violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," for the adequacy of design for the implementation of extended power uprate (EPU) to respond to changes in post-scram reactor vessel water level to prevent water intrusion into the HPCI steam supply line.

Description

On January 30, 2004, Dresden Unit 3 experienced an automatic scram due to a turbine trip. The turbine tripped on low turbine bearing lube oil header pressure while swapping from the ‘A’ turbine lube oil cooler to the standby ‘B’ turbine lube oil cooler. During the scram recovery, reactor vessel water level increased and reached the HPCI system steam supply line as indicated by the receipt of the HPCI inlet drain pot level high alarm. The alarm cleared approximately 20 minutes later.

In responding to the scram, the feedwater regulating valves (FRVs) opened from their normal 56 percent open position to 63 percent open position. The feedwater level control (FWLC) system setpoint setdown was activated which locked the FRVs at 63 percent open for 15 seconds.

At the end of the 15 second time period, the FRVs began to reposition closed to 18.9 percent open (i.e., 30 percent of their previous position). Prior to the valves reaching 18.9 percent open, the FWLC system signaled the valves to reopen based on actual water level (-5 inches). The FWLC system used a pre-designated reactor vessel water level controlling setpoint value of +5 inches (setpoint setdown) to direct the FRVs to open to restore level to +5 inches. Subsequently, the FWLC system signaled the FRVs to close. However, with the reactor vessel swell from the FWLC systems’ earlier response, the FWLC system was unable to decrease feedwater flow through the FRVs fast enough to prevent level from reaching the HPCI system steam supply line. Subsequently, the licensee calculated that approximately 60 gallons of water had entered the HPCI system turbine steam supply line. Because the HPCI system turbine was not designed for operations with any amount of water, the licensee declared the HPCI systems inoperable for both Units 2 and 3.

In addressing the event short term, the licensee immediately modified the FWLC system setpoint setdown level value from +5 inches to -10 inches. This new setpoint would allow the FWLC system to start controlling at a lower level while the dynamic response of reactor level reached a stable condition. The licensee conducted a root cause investigation for this event and determined that there was low margin in the FWLC system to accommodate changes in the post-scram vessel level response after implementing EPU in 2001 for Unit 2 and in 2002 for Unit 3. Long term corrective actions by the licensee include dynamic modeling of the reactor vessel level response by General Electric and changes to the FWLC system which were completed in 2004. The licensee entered this issue into the corrective action program as issue reports (IR) 198654 and 204690.

Analysis

In accordance with Inspection Manual Chapter (IMC) 0612, “Power Reactor Inspection Reports, Appendix B, “Issue Screening,” issued on May 19, 2005,” the inspectors determined that the licensee failed to properly verify the adequacy of the extended power uprate design implementation, on Unit 2 in 2001 and Unit 3 in 2002, to respond to changes in post-scram reactor vessel water level to prevent water intrusion into the HPCI system turbine steam supply line.

Phase 1 Screening Logic, Results and Assumptions

The inspectors determined that the issue was more than minor because it was associated with the design control attribute of the mitigating systems cornerstone in the reactor safety strategic performance area. The issue affected the mitigating systems objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences.

In accordance with IMC 0609 Appendix A, the inspectors conducted an SDP Phase 1 screening and determined that the finding degraded the mitigating systems cornerstone function of core decay heat removal. Specifically, the finding represented a loss of system safety function for high pressure coolant injection. Thus, an SDP Phase 2 evaluation was required.

Phase 2 Risk Evaluation

In accordance with IMC 0609 Appendix A, the inspectors conducted an SDP Phase 2 evaluation using the Dresden Nuclear Power Station (Revision 1) Risk-Informed Inspection Notebook.

Unique Design Features - Dresden Units 2 and 3 (2,3) are General Electric Boiling Water Reactor (BWR) Model 3 reactors with a Mark I Containment design and current design electrical ratings of 912 MWe. The units applied for, and were granted an extended power uprate (EPU) of 17 percent on December 26, 2001. In order to accomplish the EPU power upgrade, modifications were made to the high pressure steam turbines, condensate, and feedwater systems. A digital FWLC system was implemented that replaced the original FWLC system. The original operating configuration for full power was two of three main feedwater pumps running. After the increase in licensed power level, three of three main feedwater pumps are normally in operation when at full power.

The Dresden 2,3 design employs a combination of a HPCI and passive isolation condenser for high pressure decay heat removal. Unlike other BWRs, Dresden 2,3 have separate HPCI steam outlet nozzles in the reactor vessel that are located approximately 50-inches below the main steam line outlet nozzles. This factor contributed to the susceptibility for post-scram FWLC system malfunctions leading to water in the HPCI steam line.

Duration - The full period of time which the FWLC system had incorrect settings for post-scram water level settings is not precisely known. The control system settings in place at the time of the January 30, 2004, event had been in place since the implementation of the new digital FWLC system and its commissioning prior to an extended power upgrade in 2001 (Unit 2) and 2002 (Unit 3). For the purposes of this analysis, the condition duration was set to one full year, or 8760 hours.

Recovery Credit - No credit was given for HPCI recovery. Once water has entered the HPCI steam supply line, it would not be practical to perform recovery to restore HPCI, in that, the licensee would have to drain the water from the steam supply line in the midst of an emergency demand. There were no procedures to drain the line during emergency conditions.

Initiating Event Scenarios - The HPCI system was affected by the finding. According to Table 2, "Initiators and System Dependency for Dresden Units 2 and 3," the SDP Worksheets to be evaluated were all but stuck open relief valve (SORV) and large loss of coolant accident (LLOCA). The loss of offsite power (LOOP) worksheet was also not evaluated since the three motor-driven feedwater pumps would lose power upon a LOOP initiator and could not contribute to this overfill event.

SDP Dominant Sequence Worksheet Results

Dominant Core Damage Frequency (CDF) Sequences

Transient Without Power Conversion System Initiator:
TPCS (1) + IC (2) + HPI (0) + LPI (6) = 9

Medium Loss of Coolant Accident Initiator:
MLOCA (4) + HPI (0) + DEP (2) = 6

Results of SDP Phase 2

Based in the SDP counting rule, this issue is considered to be of low to moderate safety significance (White).

Phase 3 Risk Evaluation

Internal Events

The Senior Risk Analysts (SRAs) performed a risk evaluation of the Dresden HPCI flooding issue using Dresden 3 Standardized Plant Analysis Risk (SPAR) Model, Revision 3.12, February, 2005, and generic failure probabilities from NUREG 5750, "Rates of Initiating Events at Nuclear Power Plants." This evaluation determines the change between the conditional probability of core damage for the period in which the condition existed and the nominal core damage probability.

The SRAs analyzed this problem as a condition assessment involving the post-scram loss of 3/3 main feedwater pumps and inoperability of the HPCI system when normal feedwater is not initially lost or is decreasing as a result of the initiating event. In this analysis the HPCI pump is assumed failed and unrecoverable due to water in the steam supply line. The motor-driven feedwater pumps will trip on high reactor water level but are recoverable. The assumed exposure time is 1 year.

The SPAR model described three dominant sequences which together account for more than 90 percent of the change in (Δ) CDF. The three dominant sequences involve operational transients which lead to a post-scram FWLC system malfunction which results in a high reactor vessel water level condition, the tripping of all three running feedwater pumps, loss of high pressure decay heat removal (both the main condenser and the isolation condenser), the failure of HPCI, and the failure to recover reactor water level, by either restarting a main feedwater pump or depressurizing the reactor to allow low pressure makeup systems to inject. Other sequences that involve loss of main feedwater and loss of offsite power would not result in an overfill event because the feedwater system would be unavailable from the start of the event to cause the overfill.

The three dominant sequences and their conditional CDF (CCDF) values are:

> Inadvertent Open Relief Valve (IORV), failure of main feedwater, failure of HPCI, and failure of manual reactor depressurization. (3.9E-6)
>
> General Plant Transient (TRANS), failure of the isolation condenser, failure of main feedwater, failure of HPCI, and failure of manual reactor depressurization (2.9E-6)
>
> Loss of Condenser Heat Sink (LOCHS), failure of the isolation condenser, failure of main feedwater, failure of HPCI, and failure of manual reactor depressurization. (5.3E-7)

Recovery of main feedwater is not modeled in SPAR so the SRAs changed the model to simulate feedwater recovery by adding a conditional probability of high level given a FWLC system malfunction. Given the settings in the FWLC system, any similar trip with 3/3 main feedwater pumps running would likely result in the same outcome. A review of the Dresden-3 operating experience, however, indicates that in 8 such events, only two resulted in reactor pressure vessel overfill to the extent that HPCI was incapacitated. The data is too sparse to perform any significant statistical analysis. The probability of such overfill events given the high power trip with 3/3 main feedwater pumps operating was established based on a simple point estimate as 2.5E-1.

Also, in order to apply the recovery of main feedwater to the three dominant sequences, the SRAs evaluated the dependency between main feedwater recovery and operator actions to depressurize the reactor. The SRAs used the SPAR-H model and concluded that there was moderate dependency between the failure to recover main feedwater and the failure to depressurize the reactor. This was based on the assumptions that failure to recover feedwater and the failure to depressurize would likely be performed by the same crew, at the same location, but not necessarily close in time (seconds or minutes), and the operators would have additional cues upon failing to recover feedwater that would lead them to consider depressurization. Using the Human Reliability Assessment documented via the SPAR-H process, the SRAs determined the probability value for nonrecovery of feedwater as 1E-3. In addition, using the formula from the model, the probability of failure to depressurize given failure to restart feedwater is estimated at 1.4E-1.

A key modeling assumption was made that flooding of the HPCI steam line will not occur if the reactor scrams before the main turbine trips. The reason is that the turbine continues to draw steam from the reactor vessel and this tends to limit the increase in reactor water level. SRA review of the licensee's TRACG simulation runs confirms this assumption.

Using the assumption that flooding of the HPCI steam line will not occur if the reactor scrams before the main turbine trips, the SRAs consulted NUREG 5750, "Rates of Initiating Events at Nuclear Power Plants," Table 4-7. The SRAs determined that 41 percent of general transient events involve turbine trips and excessive feedwater. Therefore, the SRAs used a HPCI failure probability of 4.1E-1 to account for the fraction of TRANS events involving turbine trips and feedwater overfill scenarios.

Similarly, the SRAs used NUREG 5750, Table 4-8, and the licensee's simulation runs, and determined that 30 percent of Loss of Condenser Heat Sink (LOCHS) events will cause overfeed scenarios for this specific case. The table lists the summary of LOCHS contributors as loss of condenser vacuum, unavailability of turbine bypass capability, and closure of all main steam isolation valves. For this analysis, the contribution from the loss of condenser vacuum initiator is zero since at Dresden a reactor scram occurs first on loss of condenser vacuum followed by a turbine trip. Similarly, the contribution from the loss of bypass capability is zero for this analysis since there is no direct scram or turbine trip on loss of bypass. Likely there would be another scram signal first from, for example, electrohydraulic control system malfunction.

Regarding closure of all main steam isolation valves, the licensee determined that half of Group 1 isolation events lead to overfill conditions. The licensee's simulation run shows reactor water level to remain slightly below the bottom of the HPCI steam line. However, the vessel level is calculated to exceed the feedwater pump trip setpoint before completely stabilizing. To account for uncertainty, the licensee assumed that an initiating event that could cause a Group I isolation, concurrent with a reactor scram, could result in HPCI steam line flooding. The licensee concluded that a HPCI steam line probability of 0.5 should be used for Group I isolation events. The SRAs agreed with the licensee's conclusion and used the 0.5 probability value to determine the fraction of LOCHS events involving overfill scenarios. NUREG 5750, Table 4-8, states that 60 percent of Group I isolation events contribute to LOCHS events. Therefore, using half of this value, the SRAs used a HPCI failure probability of 3.0E-1 to account for the fraction of LOCHS events involving feedwater overfill scenarios.

The IORV initiating event scenario has the same effect as a medium steam break loss of coolant accident. The IORV scenario applies to the eight safety valves that discharge directly into the drywell as opposed to the safety relief valves that discharge to the suppression pool. In calculating the initiating event frequency, incidents of interest are those in which the safety (or safeties) valve(s) remain stuck open following the scram, resulting in continuing inventory loss and rapid pressurization of the drywell. Because high drywell pressure initiates HPCI, a running HPCI pump contributes to reactor vessel level rise and thus HPCI steam line flooding. Since there are 62 percent (8 of 13) safety/relief valves that relieve to the drywell, the SRAs used a HPCI failure probability of 6.2E-1 to account for the fraction of IORV events involving feedwater overfill scenarios. The SRAs believe this value is conservative since it is likely that not all eight safety/relief valves will remain stuck open.

<u>NRC Internal Events Analysis Conclusion</u>

The total ΔCDF from internal events is about 7.4E-6, which is in the White range of importance.

Event Tree Name and Sequences	Importance (ΔCDF)
Inadvertent open relief valve, failure of main feedwater, failure of HPCI, and failure of manual reactor depressurization	3.9E-6
General plant transient, failure of the isolation condenser, failure of main feedwater, failure of HPCI, and failure of manual reactor depressurization	2.9E-6
Loss of condenser heat sink, failure of the isolation condenser, failure of main feedwater, failure of HPCI, and failure of manual reactor depressurization	5.3E-7

Fire

The SRAs reviewed the Dresden Fire PRA list of the top ten fire scenarios. These mostly involve loss of feedwater, manual scram, and loss of offsite power initiators which do not lead to flooding of the HPCI steam supply line. In the case of LOOP initiators, the feedwater pumps are motor-driven, and therefore would lose power during a LOOP. In the case of a scram, the licensee's simulation shows that scrams that precede a turbine trip would not have resulted in high reactor water level because the turbine continues to draw steam from the reactor vessel and this tends to limit the increase in reactor water level. Lastly, the SRAs did not identify any safe shutdown equipment impacted by this finding. The SRAs concluded that the impact from fire is not significant in this SDP result.

Seismic

The SRAs reviewed Dresden's seismic CDF risk profile and determined that most seismic-induced accident scenarios (loss of offsite power) are not impacted by this SDP issue. The licensee's TRACG (Transient Reactor Analysis Code - GE) simulation performed by GE demonstrates significant margin to HPCI steam line flooding if the initiating transient is a loss of offsite AC power. Because the dominant seismic risk scenarios are those involving loss of offsite AC power, the seismic CDF risk profile is comprised of seismic-induced accident scenarios which do not lead to flooding of the HPCI steam supply line. The SRAs concluded that the impact from seismic events is not significant in this SDP result.

NRC External Events Analysis Conclusion

The total ΔCDF from external events is negligible compared to the ΔCDF from internal events.

Potential Risk Contribution Due to Large Early Release Frequency (LERF)

Using IMC 0609 Appendix H, the SRA determined that this was a Type A finding (i.e., LERF contributor) for a Mark I Containment. For Mark I containments, CDF sequences important for LERF include transients that can lead to reactor vessel breach at high pressure or at low pressure with a dry drywell floor.

The sequences that have potential for LERF contribution for this event are transients without the power conversion system (TPCS) and medium loss of coolant events (MLOCA). Applying the LERF Factor in IMC 0609 Appendix H, removes the MLOCA event from LERF consideration since its ΔLERF contributor is near zero. Regarding TPCS, an earlier evaluation for a HPCI water hammer event at Dresden (EA-02-269) considered TPCS and concluded the ΔLERF to be on the order of 1.5E-7.

SDP Process Conclusion

For ΔCDF assuming (1) a condition involving improper FWLC system settings for longer than 1 year, (2) a post-scram loss of three out of three feedwater pumps, and (3) HPCI system inoperable, and considering the impact from external events and LERF, the SRAs determined the overall ΔCDF to be about 7.6E-6, an issue of low to moderate safety significance (White).

Licensee Analysis

The licensee used information from their root cause investigation and insights gained from a GE report documenting results obtained from TRACG (Transient Reactor Analysis Code - GE) model runs under various initiating event scenarios. The licensee determined that this condition had existed since implementation of Extended Power Uprate (EPU) operation. The licensee concluded that the ΔCDF due to internal and external events to be on the order of 5E-7, in the Green range of importance. The Regional SRAs, along with the Office of Research, reviewed the licensee's analysis and disagreed with the licensee's assumptions of reducing the initiating event frequency for inadvertent open relief valve and assigning no dependence with actions for re-starting feedwater and depressurization. This accounted for the difference in the risk characterization of ΔCDF values.

Internal Events:

The licensee determined that medium LOCA events contribute 42 percent of ΔCDF. Transients with feedwater and the main condenser available contribute 33 percent to ΔCDF. Loss of cooling water events, such as loss of service water, contribute 14 percent of ΔCDF. As an example, the top 4 cutsets from the licensee's analysis, which together contribute about 62 percent of overall ΔCDF, involve:

> A steam line break medium LOCA, HPCI turbine fails to start, and operator failure to restart feedwater pumps after high level trip and depressurize the reactor. (1.9E-7)
>
> A transient with feedwater and the main condenser available, HPCI turbine fails to start, isolation condenser failure (other than makeup failure), and operator failure to restart feedwater pumps after high level trip, depressurize the reactor, and initiate isolation condenser makeup. (5.3E-8)

A transient with feedwater and the main condenser available, failure to restart feedwater, HPCI turbine fails to start, isolation condenser failure (other than makeup failure), and operator failure to depressurize the reactor, and initiate isolation condenser shell side makeup. (5.1E-8)

Loss of turbine building closed cooling water, HPCI turbine fails to start, isolation condenser failure (other than makeup failure), and operator failure to depressurize the reactor, and initiate isolation condenser makeup. (2.3E-8)

LERF Contribution

The licensee's LERF analysis was 5E-8/yr, which is consistent with a Green risk characterization.

Potential Risk Contribution due to External Events

The licensee evaluated external event contributions and determined that external hazards were not considered credible. The SRAs accepted the licensee's analysis.

Licensee Analysis Conclusion

The licensee's total ΔCDF considering internal events, LERF, and external events was:

ΔCDF - 5E-7/yr (Green)
ΔLERF - 5E-8/yr (Green)

Enforcement

10 CFR 50, Appendix B, Criterion III, Design Control states, in part, that design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.

Contrary to the above, the inspectors determined that the licensee implemented extended power uprates on Unit 2 in 2001 and Unit 3 in 2002, but failed to verify the adequacy of design for the implementation of extended power uprate to respond to changes in post-scram reactor vessel water level to prevent water intrusion into the HPCI steam supply line. This AV of 10 CFR 50, Appendix B, Criterion III, Design Control, which has low to moderate safety significance, was identified as a result of the inspectors' review of a January 30, 2004, scram event. Water intrusion into the high pressure coolant injection system turbine steam supply line occurred as a result of the scram and rendered the high pressure coolant injection system inoperable.

The finding was determined to be greater than minor because it impacted the mitigating systems cornerstone. The finding was preliminarily determined to be of low to moderate safety significance following the performance of a case-specific Phase 3 SDP evaluation. Corrective actions taken by the licensee included modifying the feedwater level control system post-scram level setpoints and dynamic modeling of the reactor vessel. **(AV 05000237/2005014-01; 05000249/2005014-01)**

4OA6 Meetings

Exit Meeting

The inspectors presented the inspection results to the Site Vice President, Mr. Danny Bost, and other members of licensee management on January 4, 2006. The inspectors asked the licensee about proprietary information associated with the inspection. Some proprietary information was identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

D. Bost, Site Vice President
D. Wozniak, Plant Manager
H. Bush, Radiation Protection, Radiological Engineering Manager
R. Conklin, Radiation Protection Supervisor
R. Ford, Emergency Preparedness Manager
J. Fox, Design Engineer
R. Gadbois, Operations Director
D. Galanis, Design Engineering Manager
V. Gengler, Dresden Site Security Director
J. Griffin, Regulatory Assurance - NRC Coordinator
P. Salas, Regulatory Assurance Manager
M. Kanavos, Site Engineering Director
A. Khanifar, Nuclear Oversight Director
J. Kish, ISI Coordinator
S. Kroma, Reactor Services Project Manager
T. Loch, Supervisor, Design Engineering
M. McGivern, System Engineer
M. Mikota, Dry Cask Project Manager, Dresden
M. Overstreet, Lead Radiation Protection Supervisor
J. Strmec, Chemistry Manager
G. Bockholdt, Maintenance Director
S. Taylor, Radiation Protection Manager

NRC

M. Ring, Chief, Division of Reactor Projects, Branch 1

IEMA

R. Schulz, Illinois Emergency Management Agency
R. Zuffa, Resident Inspector Section Head, Illinois Emergency Management Agency

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000237/2005014-01 05000249/2005014-01	AV	Failure to properly evaluate extended power uprate for its impact on post-scram reactor vessel water level to prevent water intrusion into the HPCI steam supply line.

Closed

05000237/2004002-02	URI	Water intrusion in the high pressure coolant injection system steam supply line
05000249/2004-002-00	LER	Unit 3 Automatic Scram Due to Main Turbine Low Oil Pressure Trip and Subsequent Discovery of Inoperability of the Units 2 and 3 High Pressure Coolant Injection Systems

Discussed

None

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the 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 or any part of it, unless this is stated in the body of the inspection report.

4OA3 Event Follow-up

Issue Reports:
204690; HPCI steamline water carryover during design basis event; February 27, 2004
200576; Potential ECCS Design Vulnerability - HPCI water intrusion; February 9, 2004
198654; FWLCS did not control level below 48 inches after U3 scram; January 31, 2004
198543; Unit 3 Reactor scram due to main turbine trip; January 30, 2004
198754; Potential feedwater level control issue impacts U2 HPCI; February 1, 2004
198748; HPCI inlet drain pot level high alarm received after scram; February 1, 2004
198770; Red phone call made re: HPCI inoperable on Unit 2 February 1, 2004
EC347077; Dresden Station U3 Scaling Setpoint Change Package for Dresden Station U3 FWLCS setpoint change post-scram level setpoint to minus 10 inches
EC347075; Evaluation of water in HPCI steam line after Unit 3 scram on 01-30-2004

Drawings and Prints:
—11878-9; Blume Curve Piping Isometric; High Pressure Coolant Injection Dresden Nuclear Station Unit 3; Revision 0
–ISI-122; Inservice Inspection Class I High Pressure Coolant Injection Piping, Sheets 1, Revision G
–ISI-122; Inservice Inspection Class I High Pressure Coolant Injection Piping, Sheets 2, Revision H

Other:
LER 2004-002-00; Unit 3 Automatic Scram Due to Main Turbine Low Oil Pressure Trip and Subsequent Discovery of Inoperability of the Units 2 and 3 High Pressure Coolant Injection Systems

LIST OF ACRONYMS

CDF	Core Damage Frequency
CFR	Code of Federal Regulations
EPU	extended power uprate
FRV	feedwater regulating valves
FWLC	feedwater level control
HPCI	High Pressure Coolant Injection
IMC	Inspection Manual Chapter
IR	Issue Report
LER	Licensee Event Report
LERF	Large Early Release Frequency
LOOP	loss of offsite power
SDP	Significance Determination Process
SPAR	Standardized Plant Analysis Risk
SRA	Senior Risk Analyst
TPCS	transients without the power conversion system
URI	Unresolved Item