April 18, 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: FINAL SIGNIFICANCE DETERMINATION FOR A GREEN FINDING (NRC INSPECTION REPORT NO. 05000237/2006008; 05000249/2006008) DRESDEN NUCLEAR POWER STATION UNITS 2 AND 3

Dear Mr. Crane:

The purpose of this letter is to provide you with the final significance determination for a finding involving the January 30, 2004, Unit 3 automatic scram due to the main turbine low oil pressure trip and subsequent discovery of inoperability of Units 2 and 3 high pressure coolant injection (HPCI) systems.

The finding resulted from an assessment of Unresolved Item (URI) 05000237/2004002-02 of Nuclear Regulatory Commission (NRC) Inspection Report 05000237/2004002; 05000249/2004002 issued on April 30, 2004. The finding was reviewed further in NRC Inspection Report 05000237/2005014; 05000249/2005014 dated February 3, 2006, and was assessed under the significance determination process (SDP) as a preliminary White finding (i.e., a finding with low to moderate increased importance to safety, which may require additional NRC inspection). The cover letter to the inspection report informed Exelon Generation Company, LLC (EGC) of the NRC's preliminary conclusion, provided EGC an opportunity to request a Regulatory Conference or to provide a written response on this matter, and forwarded the details on the NRC's preliminary results for this finding.

In a telephone conversation with Mr. Mark Ring of the NRC, Region III, on February 14, 2006, Mr. Pedro Salas of your staff notified us of your intent to provide a written response on the finding. In a letter, dated March 6, 2006, you agreed that a performance deficiency occurred in that the events of January 2004 led to the water entering the HPCI steam line. You performed your own risk analysis, based on plant specific information, and concluded that the change in Core Damage Frequency (ΔCDF) supported a Green (very low safety significance) finding.

Your March 6, 2006, letter provided comments on four main areas: (1) dependency between operator actions to restart feedwater and depressurize; (2) human error probability associated with post-trip condensate system alignment; (3) human error probability associated with post-trip feedwater pump restart; and (4) initiating event frequency for inadvertent opening of a safety valve.

Regarding dependency between operator actions to restart feedwater and depressurize, you calculated a human error probability of 3.2E-4 for the failure to restart feedwater after the feedwater pumps trip on high reactor water level. The NRC used a human error probability of 1.0E-3. You requested that the NRC consider using your calculated human error probability with the NRC "moderate" dependency factor of 0.14, resulting in an overall conditional failure probability of failing to depressurize as 4.5E-5. The NRC re-evaluated use of the moderate dependency factor and concluded that use of the low dependency factor was more appropriate. Therefore, assuming low dependence, the NRC revised its conditional failure probability of failing to depressurize with a new value of 5.1E-5.

Regarding the human error probability associated with post-trip condensate system alignment, you stated that the cut-sets involved with human failure to start/re-align the condensate system after the feedwater pumps trip should be eliminated. The applicable cut-sets involved operation of the condenser hotwell makeup pumps and valves. The NRC verified that the condenser hotwell makeup pumps and valves automatically function to control hotwell level. Therefore, the NRC eliminated the cut-sets associated with human error in aligning the condenser hotwell makeup pumps and valves.

Regarding the human error probability associated with post-trip feedwater pump restart, you stated that the cut-sets involved with human failure in maintaining feedwater flow should be eliminated since the feedwater system is maintained in automatic once the feedwater pumps are restarted. In our preliminary analysis, the NRC assumed additional human failure probability above that already included in the base model. Based on further review, considering that ample time is available to restart the feedwater system following a transient, and that the feedwater system is placed in automatic after it is restarted, the NRC determined that there should be no increase in risk beyond that already in the base model. Therefore, the NRC eliminated the cut-sets associated with additional human failure probability above that already assumed.

Regarding the initiating event frequency for inadvertent opening of a safety valve, you requested that, allowing for uncertainties, the NRC consider using 1.0E-4/yr. The initiating event frequency the NRC used was 1.5E-2/yr. Your approach was to review available data, declaring actual safety valve lifts to be irrelevant at Dresden for various engineering reasons. However, your approach did not address corresponding changes to the denominator in the fraction of safety valve lifts. The NRC approach was based on actual operating experience for stuck/inadvertent open safety valves at boiling water reactors. The data for such events are very limited, based upon only four actual events in 264.9 reactor critical years. The NRC concluded, considering uncertainties, that the actual spring-loaded safety valve failure rate is likely to be between your assumed value and the NRC value. Due to the large uncertainty, the NRC performed sensitivity analyses to evaluate the impact of changing the initiating event frequency for inadvertent opening of a safety valve.

Incorporating the changes from the first three items above and leaving the safety valve failure frequency unchanged from the original SDP analysis, the overall ΔCDF for this issue was determined to be 9.5E-7, in the Green range of importance. Assuming that the safety valve initiating event frequency is between your assumed value and the NRC value would lower the overall ΔCDF further into the Green range.

In summary, after considering the information developed during the inspection and the additional information you provided in your March 6, 2006, letter, the NRC has concluded that the final significance of the finding is appropriately characterized as Green (i.e., a finding of very low safety significance), in the Mitigating System cornerstone.

Notwithstanding the information provided by EGC in its written response, the NRC determined that one violation occurred, involving the requirements of 10 CFR 50, Appendix B, Criterion III, "Design Controls." The NRC concluded that EGC implemented extended power uprates on Unit 2 in 2001 and Unit 3 in 2002, but failed to verify the adequacy of design of 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 violation was identified as a result of the inspectors' review of the January 30, 2004, scram event. Water intrusion into the HPCI system turbine steam supply line occurred as a result of the scram and rendered the HPCI system inoperable. The NRC determined that EGC was in violation during 2001 through 2004, however, the violation was identified and corrective actions were taken after the January 2004 scram event.

Because of the very low safety significance of this violation and because the issue was entered into your corrective action program, the NRC is treating the violation as a non-cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy. If you contest this NCV, you should provide a response within 30 days of the date of this letter, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC, 20555-0001; with copies to the Regional Administrator Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC, 20555-0001; and the NRC Resident Inspector at the Dresden Nuclear Power Station.

For administrative purposes, this letter is issued as a separate NRC Inspection Report, No. 05000237/2006008; 05000249/2006008 and the above NCV is identified as **NCV 05000237/2006008-01; 05000249/2006008-01:** "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." Accordingly, apparent violation (AV) 05000237/2005014-01; 05000249/2005014-01 is closed.

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

Should you have any questions regarding this letter, please contact Mark Ring, Chief, Reactor Projects Branch 1 at 630-829-9703.

Sincerely,

/RA/

Mark A. Satorius, Director
Division of Reactor Projects
Region III

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

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Dresden Nuclear Power Station Plant Manager
Regulatory Assurance Manager - Dresden
Chief Operating Officer
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Regulatory Assurance Manager - Dresden
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Senior Vice President - Nuclear Services
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