October 10, 2007

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

SUBJECT: QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2, NRC EVALUATION OF CHANGES, TESTS, OR EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORT 05000254/2007008(DRS); 05000265/2007008(DRS)

Dear Mr. Crane:

On September 7, 2007, 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 Quad Cities Nuclear Power Station. The enclosed report documents the results of the inspection, which were discussed with Mr. T. Tulon, and others of your staff at the completion of the inspection on September 7, 2007.

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, which involved violations of NRC requirements. 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 (NCV) in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

If you contest the subject or severity of a NCV, 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 Quad Cities Nuclear Power Station.

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. 50-254; 50-265
License Nos. DPR-29; DPR-30

Enclosure: Inspection Report 05000254/2007008(DRS); 05000265/2007008(DRS)
w/Attachment: Supplemental Information

cc w/encl:
Site Vice President - Quad Cities Nuclear Power Station
Plant Manager - Quad Cities Nuclear Power Station
Regulatory Assurance Manager - Quad Cities Nuclear Power Station
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
Vice President - Law and Regulatory Affairs Mid American Energy Company
Assistant Attorney General
Illinois Emergency Management Agency
State Liaison Officer, State of Illinois
State Liaison Officer, State of Iowa
Chairman, Illinois Commerce Commission
Chief Radiological Emergency Preparedness Section, Dept. Of Homeland Security
D. Tubbs, Manager of Nuclear MidAmerican Energy Company

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Mid American Energy Company
Assistant Attorney General
Illinois Emergency Management Agency
State Liaison Officer, State of Illinois
State Liaison Officer, State of Iowa
Chairman, Illinois Commerce Commission
Chief Radiological Emergency Preparedness Section,
Dept. Of Homeland Security
D. Tubbs, Manager of Nuclear
MidAmerican Energy Company

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Inspection Report to Mr. C. Crane from Mr. D. E. Hills dated September , 2007

SUBJECT: QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2, NRC EVALUATION OF CHANGES, TESTS, OR EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORT 05000254/2007008(DRS); 05000265/2007008(DRS)

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

| | |
|---|---|
| Docket No: | 50-254; 50-265 |
| License Nos. | DPR-29; DPR-30 |
| Report No: | 05000254/2007008(DRS); 05000265/2007008(DRS) |
| Licensee: | Exelon Nuclear |
| Facility: | Quad Cities Nuclear Power Station, Units 1 and 2 |
| Location: | Cordova, IL 61242-9740 |
| Dates: | August 20 through September 7, 2007 |
| Inspectors: | R. Daley, Senior Reactor Inspector<br>Z. Falevits, Senior Reactor Inspector<br>J. Bozga, Reactor Inspector (In Training) |
| Approved by: | D. Hills, Chief<br>Engineering Branch 1<br>Division of Reactor Safety (DRS) |

Enclosure

## SUMMARY OF FINDINGS

IR 05000254/2007008(DRS); 05000265/2007008(DRS); 08/20/2007 through 09/07/2007; Quad Cities Nuclear Power Station, Units 1 and 2; 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 two regional based engineering inspectors. Two Green Non-Cited Violations (NCVs) 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: Mitigating Systems

Green. The inspectors identified a Severity Level IV NCV of 10 CFR 50.59(d)(1) for the licensee's failure to perform an adequate 10 CFR 50.59 evaluation for bypassing a channel of the Main Steam Line (MSL) tunnel high temperature instrumentation and for the failure to perform an adequate 10 CFR 50.59 evaluation for changing the license basis to allow operating the Electrohydraulic Control (EHC) System pressure regulator with only one channel in service. Even though the licensee did not intend to operate the plant permanently with a channel of the MSL tunnel high temperature bypassed or with only one EHC pressure regulator channel, the 10 CFR 50.59 evaluations that were performed allowed it. Because of this, the inspection team could not reasonably determine that these changes would not have required a license amendment, because the bypassing of the MSL tunnel high temperature channel could have resulted in more than a minimal increase in the likelihood of a malfunction of a structure, system, or component important to safety. Additionally, the change to allow operating the EHC System pressure regulator with only one channel in service could have created a possibility of a malfunction of an SSC important to safety with a different result. This issue was entered into the licensee's corrective action program.

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 these 10 CFR 50.59 evaluations would not have ultimately required NRC prior approval. The inspectors evaluated the finding using Inspection Manual Chapter (IMC) 0609, Appendix A, Phase 1 screening for the mitigating systems cornerstone and determined that the finding was of very low safety significance because they were able to answer "no" to the Mitigating Systems screening questions in the Phase 1 Screening Worksheet. Specifically, while the licensee failed to perform an adequate 10 CFR 50.59 evaluation for bypassing a channel of the MSL tunnel high temperature instrumentation and for allowing operation of the EHC System pressure regulator with only one channel in service, the licensee would have been able to perform these same actions under the NRC Part 9900 Technical Guidance for Degraded or Nonconforming Conditions. (Section 1R17.1.b.1)

Green. The inspections identified an NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” that was of very low safety significance. Specifically, Motor Operated Valve (MOV) delays caused by voltage dips during load sequencing were not translated into and accounted for in the design basis for the In-Service Testing (IST) stroke time acceptance criteria for the Residual Heat Removal (RHR) system inboard and outboard shutoff valves and two core spray inboard isolation valves. This issue was entered into the licensee’s corrective action program.

The issue was more than minor because it was associated with the Mitigating System Cornerstone attribute of “Design Control,” and affected the cornerstone objective of ensuring the capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the MOV delays caused by voltage dips during Emergency Core Cooling System (ECCS) load sequencing were not accounted for in the licensee’s design basis. This introduced non-conservativisms in the margins for MOV IST acceptance criteria and also potentially for the acceptance criteria themselves. This finding was of very low safety significance, because the inspectors answered “no” to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. Specifically, even though the MOV delays were non-conservative, the actual MOV stroke times during the most recent IST testing for the valves in question were much less than the IST acceptance criteria. (Section 1R17.1.b.1)

## B. Licensee-Identified Violations

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 August 20 through September 7, 2007, the inspectors reviewed five evaluations performed pursuant to 10 CFR 50.59 to determine if the evaluations were adequate and that prior NRC approval was obtained as appropriate. The team could not review the minimum sample size of six evaluations, because only five evaluations were performed during the biennial sample period. The inspectors also reviewed 13 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 10 CFR 50.59 Evaluations for the Main Steam Line Tunnel High Temperature Instrumentation and the Electrohydraulic Control System Pressure Regulator

Introduction: The inspectors identified a 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 10 CFR 50.59 evaluation for bypassing a channel of the MSL tunnel high temperature instrumentation and for the failure to perform an adequate 10 CFR 50.59 evaluation for changing the license basis to allow operating the EHC System pressure regulator with only one channel in service.

Description: The licensee performed 10 CFR 50.59 Evaluation QC-E-2007-001 on June 5, 2007, to support Temporary Configuration Change Package (TCCP) Engineering Change (EC) 366160 initiated to install a temporary bypass jumper across the faulty MSL tunnel high temperature switch number TS 1-0261-15B. This bypass jumper rendered the faulty switch inoperable and changed the original design basis of two out of four open switches for trip/isolation to a two out of three open switches for trip/isolation. This same type of evaluation had been performed twice previously in June 2005 and July 2003. However, the licensee failed to provide a basis for why this action did not present more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety. Specifically, since less channels would be in service after this change, there was a reduction in channel diversity. Per NEI 96-07, "Guidelines for 10 CFR 50.59 Implementation," Section 4.3.2, a change that reduces system/equipment redundancy, diversity, separation, or independence results in more than a minimal increase in the likelihood of occurrence of a malfunction of a System, Structure, and Component (SSC) important to safety. The licensee entered this condition into their corrective action program as AR 00666761. The inspectors noted that if the MSL temperature channels had been treated as a nonconforming condition as per the NRC Part 9900 Technical Guidance, "Operability Determination and Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety," a 10 CFR 50.59 evaluation would only have been required to evaluate whether the temporary change/compensatory action of bypassing the MSL temperature channel impacted other aspects of the facility or procedures described in the Updated Final Safety Analysis Report (UFSAR). Because that process would address the reduction in channel diversity as a nonconforming condition, it would not have been necessary for the 10 CFR 50.59 to likewise do so. The licensee, however, did not evaluate the condition under this process. Also, while the jumper was installed only temporarily, the 10 CFR 50.59 did not contain that limitation.

Additionally, the inspection team identified a similar issue pertaining to a 10 CFR 50.59 evaluation performed in regard to the EHC System pressure regulator. On May 2, 2003, the licensee performed and approved a 10 CFR 50.59 evaluation that allowed operation of the EHC system with a pressure regulator out of service. Again, the 10 CFR 50.59 evaluation that was performed addressed the acceptability of operating the plant with only one pressure regulator channel in service. While the licensee maintained that this type of operation would only occur, and did only occur, when a pressure regulator channel needed to be worked on, the 10 CFR 50.59 evaluation allowed the continuous operation of the EHC system with only one channel in service.

Chapter 15.2 of the licensee's UFSAR addressed the Steam Pressure Regulator Malfunction event. In that event, the UFSAR states that if the pressure regulator were to fail low, the backup regulator would take over control of the turbine valves as soon as the failed regulator attempts to close the valves and pressure begins to rise. The inspectors noted that if the pressure regulator were to fail low with only one channel in service, there would be no backup regulator to take control. This would result in a Turbine Control Valve (TCV) closure event that would result in a more severe transient than what would be expected by the UFSAR description of a pressure regulator failure. The inspectors determined that this would result in an initiator or failure whose effects would not be bounded by those explicitly described in the UFSAR. As a result, the inspectors determined the change was a malfunction with a different result; however, as with the 10 CFR 50.59 evaluation for the MSL temperature channels, the inspectors noted that if a pressure regulator channel were out of service, this condition could be evaluated under the NRC Part 9900 Technical Guidance, "Operability Determination and Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to

Quality or Safety," or 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," as appropriate.

Because both of these violations have similar causes, misapplication of 10 CFR 50.59, they were treated as one violation of 10 CFR 50.59. The licensee entered this second condition into their corrective action program as AR 00666988. The licensee determined that had the 10 CFR 50.59 evaluations been performed to address the compensatory measures in place per NRC Part 9900 Technical Guidance for the two separate conditions, the evaluation could have been performed satisfactorily.

Analysis: This failure to perform adequate safety evaluations in accordance with 10 CFR 50.59 was a performance deficiency warranting a significance determination. Specifically, the licensee failed to perform an adequate 10 CFR 50.59 evaluation for bypassing a channel of the MSL high temperature instrumentation and failed to perform an adequate 10 CFR 50.59 evaluation to allow operating the EHC System pressure regulator with only one channel in service. The finding was determined to be more than minor because the inspectors could not reasonably determine that these 10 CFR 50.59 evaluations 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 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 IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and answered "no" to the Mitigating Systems screening questions in the Phase 1 Screening Worksheet. Specifically, while the licensee failed to perform an adequate 10 CFR 50.59 evaluation for bypassing a channel of the MSL tunnel high temperature instrumentation and for allowing operation of the EHC System pressure regulator with only one channel in service, the licensee would have been able to perform these same actions under the NRC Part 9900 Technical Guidance for Degraded or Nonconforming Conditions. 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. This finding did not have any cross-cutting aspects, because the initial 10 CFR 50.59 evaluations, (the initial evaluation for the MSL tunnel high temperature instrumentation was performed in July 2003) were performed greater than two years prior to the inspection team discovering the issue.

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 May 2003 and June 2007, performed 10 CFR 50.59 evaluations for bypassing a channel of the MSL tunnel high temperature instrumentation and for allowing operation of the EHC System pressure regulator with only one channel in service that did not provide an adequate basis for the determination that the change, test, or experiment did not require a license amendment. Even though the licensee did not intend to operate the plant permanently with a channel of the MSL tunnel high temperature bypassed or with only one EHC pressure regulator channel, the 10 CFR 50.59 evaluation that were performed allowed it. Because of this, the inspection team could not reasonably determine that these changes would not have required a license amendment, because the bypassing of the MSL tunnel high temperature channel could have resulted in more than a minimal increase in the likelihood of a malfunction of a structure, system, or component important to safety. Additionally, changing the license basis to allow operating the EHC System pressure regulator with only one channel in service could have created a possibility of a malfunction of an SSC important to safety with a different result. 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 (AR's 00666761 and 00666988), it is considered a Non-Cited Violation consistent with VI.A.1 of the NRC Enforcement Policy. (NCV 05000254/2007008-01; 05000265/2007008-01(DRS))

## 1R17 Permanent Plant Modifications (71111.17B)

### .1 Review of Permanent Plant Modifications

#### a. Inspection Scope

From August 20 through September 7, 2007, the inspectors reviewed six 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 reviewed to ensure the functionality of the modification, its associated system, and any support systems. 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 Failure to Account for Delays in ECCS MOV's Due to Voltage Dips during Load Sequencing

Introduction: The inspectors identified a NCV having very low safety significance (Green) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." Specifically, the licensee failed to account for delays in the ECCS response time for MOV stalling caused by momentary voltage dips during load sequencing.

Description: During review of calculation QDC-0000-E-0206, the inspection team determined that the following verbage was contained in the Design Input Data section of the calculation:

"Bus voltages based on running loads are used to calculate the MOV terminal voltages on the basis that the block starting of large motors is a transient condition that is overly conservative for the evaluation of MOV capability. In the worst case, the block starting of one or more large motors would momentarily delay the operation of an MOV until the voltage recovered to the point where sufficient torque could be developed."

Because the licensee was not able to quantify the amount of delay for the MOVs, the inspectors were concerned that ECCS response times could be affected. Specifically, MOVs that are required to reposition during a design basis Loss of Coolant Accident (LOCA) would now reposition at a slower rate causing ECCS injection water to be delayed in reaching the core. This delay would be caused by voltage dips during load sequencing that could potentially stall the MOVs until voltage eventually recovered.

In response to the team's concerns, the licensee was able to produce an historical evaluation that showed that for the valves of concern (the RHR inboard and outboard shutoff valves and two core spray inboard isolation valves), the maximum delay would be two seconds. Additionally, the licensee demonstrated that this 2-second delay would not adversely affect their ECCS response analysis since there was still substantial time for water to reach the core regardless of the delay.

However, the inspectors were concerned that these delays were not properly incorporated into and accounted for in the design basis for the IST acceptance criteria. While no calculation existed for the determination of the acceptance criteria, the licensee was able to determine what the margin was between the ECCS analysis requirement for the valves repositioning and the IST acceptance limit. This margin varied from three to five seconds for the valves. There appeared to be insufficient documentation that stated the basis for this margin; however, the LOCA input assumptions in the accident analysis included a 3-second time to account for initiation and surveillance time measurement uncertainties. It was unclear how this three seconds was factored into the time assumptions associated with the IST testing. This three seconds (assuming that it is not overly conservative), in addition to the two seconds for MOV delay caused by voltage drop during load sequencing would exceed the established margin currently in place between the LOCA assumptions and the IST acceptance criteria. The inspectors determined that this lack of a basis for margin affected the potential adequacy of the IST acceptance time limits. As a result, the inspectors determined that this lack of a design basis for the IST acceptance criteria for the ECCS valves, specifically the RHR inboard and outboard shutoff valves and two core spray inboard isolation valves, was a violation of 10 CFR 50, Appendix B, Criterion III, Design Control.

Analysis: The failure to account for MOV delays caused by voltage dips during ECCS load sequencing was a performance deficiency warranting a significance determination. The issue was more than minor because it was associated with the Mitigating System Cornerstone attribute of "Design Control," and affected the cornerstone objective of ensuring the capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the MOV delays caused by voltage dips during ECCS load sequencing were not accounted for in the licensee's design basis. This introduced non-conservativisms in the margins for MOV IST acceptance criteria and also potentially for the acceptance criteria themselves.

The finding screened as having 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. Specifically, even though the MOV delays were non-conservative, the actual MOV stroke times during the most recent IST testing for the valves in question were much less than the IST acceptance criteria. This finding did not have any cross-cutting aspects, because the failure to account for these MOV time delays was an old design issue that should have been addressed in the early 1990's when the licensee initially discovered that these time delays were possible.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control" states, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above, important design basis information relating to MOV delays caused by voltage dips during load sequencing were not translated into and accounted for in the specifications for the IST stroke time acceptance criteria for the RHR inboard and outboard shutoff valves and two core spray inboard isolation valves.

Because this failure to account for delays due to MOV stalling in the IST acceptance criteria was determined to be of very low safety significance and because it was entered in the licensee's corrective action program as AR 00668845, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000254/2007008-02; 05000265/2007008-02(DRS))

## 4. OTHER ACTIVITIES (OA)

### 4OA2 Identification and Resolution of Problems

#### .1 Routine Review of Condition Reports

##### a. Inspection Scope

From August 20 through September 7, 2007, 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.

4OA6 Meeting(s)

.1 Exit Meeting

The inspectors presented the inspection results to Mr. T. Tulon and others of the licensee's staff, on September 7, 2007. 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

Licensee
B. Adams, Engineering Director
K. Adlon, Design Engineer
C. Alguire, Design Engineer
W. Beck, Regulatory Assurance Manager
S. Bolimie, Design Engineering Manager
J. Campagna, Design Engineer
M. Humphrey, Engineering Programs
J. Taft, Design Engineering Supervisor
M. Wagner, Licensing Specialist

Nuclear Regulatory Commission
K. Stoedter, Senior Resident Inspector
D. Hills, EB1 Branch Chief

## ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Opened and Closed

| | | |
|---|---|---|
| 05000254/2007008-01; 05000265/2007008-01 | NCV | Inadequate 10 CFR 50.59 Evaluations for the Main Steam Line Tunnel High Temperature Instrumentation and the Electrohydraulic Control System Pressure Regulator |
| 05000254/2007008-02; 05000265/2007008-02 | NCV | Failure to Account for Delays in ECCS MOV's Due to Voltage Dips during Load Sequencing |

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.

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

10 CFR 50.59 Screenings

FPR-R17-006; Update NFPA Code References and Provide a Table with the Deviations; dated March 26, 2004

FPR-R17-008; Transfer Pumps Eliminated as Normal Fill Supply to Diesel Driven Fire Pumps; dated July 21, 2005

QC-S-2003-0205; Unit 1 Cycle 18A Core Operating Limits Report (COLR) Revision 1, UFSAR Update and Statistically Based Rod Withdrawal Error Analysis for Dresden and Quad Cities; Revision 0

QC-S-2004-0151; Tech Spec Bases Change Request # QC-BAS-04-003 and UFSAR Change #UFSAR-03-R8-048; Revision 0

QC-S-2005-0182; EC 357070 Install Switch Jumper in Auxiliary Relay 2-0590-102D Circuit to Eliminate Relay Chatter; Revision 00

QC-S-2005-0186; OCOA 6100-03 Loss of Offsite Power; Revision 0

QC-S-2006-0004; Circulating Water Pump Trip on a LOCA Signal; Revision 1

QC-S-2006-0009; UFSAR Change to Paragraph 7.3.2.6.1.A; Revision 0

QC-S-2006-0042; Scorpion II 360-Degree Work Platform; Revision 0

QC-S-2006-0072; EC 342196 Fuse Discrepancy for ADS Fuse FF-F9 at Panel 901-33; Revision 0

QC-S-2007-0013; Required Cold Weather Routines; Revision 0

QC-S-2007-0044; Update Minimum ECCS Room Cooler Flow Rates; Revision 0

QC-S-2007-0072; OCOA 1000-07 and 1000-08 Loss of 125VDC Control Power to RHR Channel 1(2) A and 1(2) B Initiation Logic; Revision 0

10 CFR 50.59 Evaluations

QC-E-2005-007; EC 355823 - Install Temporary Jumper Across MST High Temperature Switch; June 3, 2005

QC-E-2005-009; U2 250 VDC Battery Replacement/EC 356391 (WO 826101); Revision 0

QC-E-2006-002; EC 359006 and 359004/UFSAR-05-R9-036; Revision 0

QC-E-2006-003; Installation of Digital EHC System; Revision 1

QC-E-2007-001; EC 366160 Install Temporary Jumper Across Main Steam Line Tunnel High Temperature Switch; June 5, 2007

IR17 Permanent Plant Modifications (71111.17B)

Modifications

EC 22211; Replace SBGT Flow Instrumentation FT 0-7541-1A and FI 0-7540-13A; December 30, 2005

EC 339281; U2 Margin Improvement, MOV 2-2301-8, MOV 2-1201-2, MOV 2-0220-1, MOV 2-0220-2; Revision 2

EC 359787; Revise ATWS Instrument Loop [PT 1-0263-20(A, B, C, D) and PS 1-0263-22(A, B, C, D)] Scram Setpoint; May 15, 2007

EC/DCP 0000339279 001; MOV Margin Improvement By Installing Closing Torque Switch Bypass MO 1-0220-1; MO 1-0220-2, MO 1-2301-14; May 23, 2007

EC/DCP 00003350636 001; Install an Override Switch in the Auto Open Logic for RCIC Torus Valve 1-1301-25 and 26; November 8, 2005

EC/DCP 0000364951 001; Replace RHR Injection Valve 1-1001-28B Open Interlock Timer; dated May 6, 2007

EC/DCP 0000364951 000 and 001; TCCP Main Steam Tunnel Temperature Switch Bypass; June 13, 2007

Other Documents Reviewed During Inspection

Corrective Action Program Documents Generated As a Result of Inspection

AR 664159; 2007 Mods/50.59 Insp: Calc Did Not Check Stresses in Beam; dated August 24, 2007

AR 665556; 2007 Mods/50.59 Insp:  Incorrect EC Reference in Work Order; dated August 28, 2007

AR 666031; 2007 Mods/50.59 Insp:  Obsolete Calculation References; dated August 29, 2007

AR 666239; 2007 Mods/50.59 Insp:  Misleading Statement in 50.59 Screening; dated August 30, 2007

AR 666404; 2007 Mod/50.59 Insp:  RCIC Min Flow Vlv UFSAR Discussion; dated August 30, 2007

AR 666761; 2007 Mods/50.59 Insp:  50.59 not Identified as Comp Measure; dated August 31, 2007

AR 666988; 2007 Mods/50.59 Insp:  Additionalk detain Needed in 2003 50.59 Eval; dated August 31, 2007

AR 667824; 2007 Mods/50.59 Insp:  Concerns with Screening QC-S-2006-004; dated September 4, 2007

AR 667987; 2007 Mods/50.59 Insp:  Need to Clarify NES-MS-04.5; dated September 5, 2007

AR 667992; 2007 Mods/50.59 Insp:  Issue Identified in Calc QDC-5700-S-14; dated September 5, 2007

AR 668845; 2007 Mods/50.59 Insp:  MOV Testing Margin; dated September 7, 2007

AR 00668406; 2007 Mods/50.59 insp:  Administrative Error in Design Change; dated September 6, 2007

<u>Corrective Action Program Documents Reviewed During the Inspection</u>

AR 00167920; Half Group and RWCU Isolation; dated July 17, 2003

AR 00340740; Received Channel B Mn Stm Tunnel High Temp ½ Group 1; dated June 3,2005

AR 00551409; Post Fire Safe Shutdown Manual Actions (RIS 2006-10); dated October 31, 2006

AR 00636293; Relay 595-101B Dropping Out on Unit 1 - ½ Group 1 Isolation; dated June 1, 2007

AR 00639324; RWCU Isolation Due to 1-0261-15B Switch Actuation; dated June 12, 2007

Calculations

QDC-0000-E-0206; Motor Terminal Voltage Calculation for Quad Cities Unit 1 and Unit 2 GL 89-10 Motor Operated Valves; dated January 19, 2007

Drawings

4E - 1503B; Schematic Diagram PCI System Panel 901-17 Trip Logic; Revision AV

4E-1675B; Wiring and Schematic Diagram Reactor Bldg Essential Serv 480V MCC 18-1A, Part 2; Revision AV

4E-1675D; Wiring and Schematic Diagram Reactor Bldg Essential Serv 480V MCC 18-1A, Part 4; Revision AF

4E-1684E; Wiring Diagram Reactor Bldg 250V DC MCC 1B, Part 1; Revision AC

4E-1758A; Wiring Diagram panel 901-33,Part 1; Revision BK

4E-1765B; Wiring Diagram panel 901-47; Revision H

4E-1438F; Schematic Diagram RHR System Relay Logic DIV II, Sheet 6; Revision S

Procedures

CC-AA-102; Design Input and Configuration Change Impact Screening; Revision 14

Miscellaneous Documents

CHRON No.179514; DC Auxiliary Power System Guidance, Maximum Expected Electrolyte Temperature and Pre-Fault Voltage Considerations in Determining Available Short Circuit Current for DC Systems; dated January 27, 1992

EC 343713; Installation of Unit 1 Digital EHC; Revision 1

UFSAR-03-R8-022; Implementation of Pressure Regulator Out of Service as an Equipment Out of Service Option; dated May 5, 2003

WO 887321; Circulating Water Pump Trip on a LOCA Signal EC 358696; dated May 18, 2006

WO 859269; ECCS Simulated Automatic Actuation and DG Auto-Start Division 1; dated May 18, 2007

WO 859901; ECCS Simulated Automatic Actuation and DG Auto-Start Division 2; dated April 30, 2007

AEI Test No. 0591-1; Stationary Battery Short Circuit Test; dated May 15 and 16, 1991

## LIST OF ACRONYMS USED

| | |
|---|---|
| ADAMS | Agency-Wide Document Access and Management System |
| CFR | Code of Federal Regulations |
| DRP | Division of Reactor Projects |
| DRS | Division of Reactor Safety |
| EC | Engineering Change |
| ECCS | Emergency Core Cooling System |
| EHC | Electrohydraulic System |
| IMC | Inspection Manual Chapter |
| IR | Inspection Report |
| IST | In-Service Testing |
| LOCA | Loss of Coolant Accident |
| MOV | Motor Operated Valve |
| MSL | Main Steam Line |
| NCV | Non-Cited Violation |
| NEI | Nuclear Energy Institute |
| NRC | Nuclear Regulatory Commission |
| PARS | Publicly Available Records |
| RHR | Residual Heat Removal |
| SDP | Significance Determination Process |
| SSC | System, Structure, and Component |
| TCCP | Temporary Configuration Change Package |
| TCV | Turbine Control Valve |
| TS | Technical Specifications |
| UFSAR | Updated Final Safety Analysis Report |