June 29, 2004

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

SUBJECT: QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2
NRC SAFETY SYSTEM DESIGN AND PERFORMANCE CAPABILITY
INSPECTION 05000254/2004004(DRS); 05000265/2004004(DRS)

Dear Mr. Crane:

On May 28, 2004, the U.S. Nuclear Regulatory Commission (NRC) completed a baseline inspection at your Quad Cities Nuclear Power Station. The enclosed report documents the inspection findings which were discussed on May 28, 2004, with Mr. T. Tulon and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and to 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. Specifically, this inspection focused on the design and performance capability of the safe shutdown makeup pump and reactor core isolation cooling systems.

Based on the results of this inspection, there was one finding concerning the reactor core isolation cooling system torus suction isolation valve control logic, which would prevent the control room operator from isolating non-seismically qualified reactor core isolation cooling system discharge piping under certain design basis conditions. This finding did present an immediate safety concern. However, compensatory measures are in place while long-term corrective measures are being implemented. This issue is unresolved pending your staff's review of the seismic qualification of the reactor core isolation cooling system discharge piping. The safety significance of this finding will be determined once the NRC evaluates the results of these reviews in accordance with the agency's Significance Determination Process (Phase 3). In addition, one NRC-identified finding of very low safety significance was identified.

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Sincerely,

/RA/

Julio Lara, Chief
Electrical Engineering Branch
Division of Reactor Safety

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

Enclosure: Inspection Report 05000254/2004004(DRS);
05000265/2004004(DRS)

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
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Manager Licensing - Dresden and Quad Cities
Senior Counsel, Nuclear, Mid-West Regional
Operating Group
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Vice President - Law and Regulatory Affairs
Mid American Energy Company
Assistant Attorney General
Illinois Department of Nuclear Safety
State Liaison Officer, State of Illinois
State Liaison Officer, State of Iowa
Chairman, Illinois Commerce Commission
D. Tubbs, Manager of Nuclear
MidAmerican Energy Company

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Director Licensing - Mid-West Regional
Operating Group
Manager Licensing - Dresden and Quad Cities
Senior Counsel, Nuclear, Mid-West Regional
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Document Control Desk - Licensing
Vice President - Law and Regulatory Affairs
Mid American Energy Company
Assistant Attorney General
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MidAmerican Energy Company

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos: 50-254; 50-265
License Nos: DPR-29; DPR-30

Report No: 05000254/2004004(DRS);
05000265/2004004(DRS)

Licensee: Exelon Generation Co., LLC

Facility: Quad Cities Nuclear Power Station, Units 1 and 2

Location: 22710 206th Avenue North
Cordova, IL 61242

Dates: May 10 through 28, 2004

Inspectors: A. Dunlop, Senior Engineering Inspector
D. Jones, Engineering Inspector
G. O=Dwyer, Engineering Inspector
S. Sheldon, Engineering Inspector
C. Baron, Mechanical Contractor

Approved by: J. Lara, Chief
Electrical Engineering Branch
Division of Reactor Safety (DRS)

Enclosure

SUMMARY OF FINDINGS

IR 05000254/2004004(DRS); 05000265/2004004(DRS); 05/10/2004 - 05/28/2004; Quad Cities Nuclear Power Station; Safety System Design and Performance Capability.

The inspection was a three week baseline inspection of the design and performance capability of the shutdown makeup pump and reactor core isolation cooling systems. The inspection was conducted by regional engineering inspectors and a mechanical consultant. One issue of very low safety significance and one unresolved item with the potential safety significance greater than Green were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, ASignificance 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 finding of very low safety significance involving inadequate design control of the reactor core isolation cooling system. Specifically, the design of the reactor core isolation cooling system and plant operating procedures did not provide adequate minimum flow protection for the reactor core isolation cooling pump. As a result, the reactor core isolation cooling flow could be reduced below the minimum flow requirements for the pump, potentially resulting in pump damage. This finding applies to both units.

This finding was more than minor since it could have affected the mitigating system cornerstone objective of ensuring the availability of systems required to respond to initiating events. This finding was of low safety significance because it did not represent an actual degradation of the reactor core isolation cooling system. The licensee initiated appropriate corrective actions, including implementing a procedure change and obtaining formal minimum flow information from the pump vendor, to ensure continued operability. No violation of NRC requirements occurred. (Section 1R21.2.b.1)

$ TBD. The inspectors identified a violation of 10 CFR Part 50, Appendix B, Criterion III, ADesign Control.@ Specifically, the design of the reactor core isolation cooling system did not provide adequate capability to isolate the safety-related torus from the non-seismic reactor core isolation cooling system under all conditions. As a result, torus water could potentially drain into the reactor building following a seismic event that could rupture the non-seismically qualified reactor core isolation cooling piping. The loss of torus inventory could potentially affect the safety-related water supply for emergency core cooling systems and primary containment integrity. This finding applies to both units.

The inspectors considered that this finding could have affected the mitigating system cornerstone objective of ensuring the availability of mitigating systems required to respond to initiating events, as well as the barrier integrity cornerstone objective of maintaining the functionality of containment. Since this performance deficiency involves seismic qualification issues, the safety significance will be determined following NRC evaluation of the licensee's seismic analysis reviews and in accordance with a phase 3 evaluation. The licensee has initiated appropriate compensatory actions to ensure continued operability by initiating a procedure that could remotely bypass the control logic such that the operators could close the isolation valve when required for containment isolation. (Section 1R21.2.b.12)

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

None.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstone: Mitigating Systems and Barrier Integrity

1R21 Safety System Design and Performance Capability (71111.21)

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

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

The systems and components selected were the safe shutdown makeup pump (SSMP) and reactor core isolation cooling (RCIC) systems (two samples). These systems were selected for review based upon:

$ having high probabilistic risk analysis rankings;
$ considered high safety significant maintenance rule systems;
$ not having received recent NRC review; and
$ being complementary systems.

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

$ licensee technical specifications;
$ applicable updated final safety analysis report (UFSAR) sections; and
$ the systems' design documents.

The following system and component attributes were reviewed in detail:

System Requirements

Process Medium - water;
Energy Source - electrical power, steam, air;
Control Systems - initiation, control, and shutdown actions;
Operator Actions - initiation, monitoring, control, and shutdown; and

Heat Removal - ventilation.

System Condition and Capability

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

Component Level

Equipment Qualification - temperature and radiation; and
Equipment Protection - seismic and electrical.

.1 System Requirements

a. Inspection Scope

The inspectors reviewed the UFSAR, technical specifications, system notebooks, lesson plans, drawings, and other available design basis information, as listed in the attached List of Documents, to determine the performance requirements of SSMP, RCIC, and their associated support systems. The reviewed system attributes included process medium, energy sources, control systems, operator actions, and heat removal. The rationale for reviewing each of the attributes was:

Process Medium: This attribute required review to ensure that the SSMP and RCIC systems would supply the required amount of water to the reactor following normal transients and design basis events.

Energy Sources: This attribute needed to be reviewed to ensure that the SSMP and RCIC systems would start when called upon, and that appropriate valves would have sufficient power to change state when so required.

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

Operations: This attribute was reviewed because the emergency operating procedures permitted the operators to manually control RCIC operation to maintain desired reactor water level. The SSMP was a manually initiated system, which included several operator actions that were identified by the licensee as risk significant. Therefore, operator actions played an important role in the ability of the SSMP and RCIC systems to achieve their functions.

Heat Removal: This attribute was reviewed to ensure that the room coolers provided sufficient heat removal capability for the SSMP and RCIC systems.

b. Findings

No findings of significance were identified.

.2 System Condition and Capability

a. Inspection Scope

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

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

Installed Configuration: The inspectors confirmed that the installed configuration of the SSMP and RCIC systems met the design basis by performing detailed system walkdowns. The walkdowns focused on the installation and configuration of piping, components, and instruments; the placement of protective barriers and systems; the susceptibility to flooding, fire, or other environmental concerns; physical separation; provisions for seismic and other pressure transient concerns; and the conformance of the currently installed configuration of the systems with the design and licensing bases.

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

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

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

b. Findings

.1 Reactor Core Isolation Cooling Minimum Flow Valve

Introduction: The inspectors identified a finding of very low safety significance (Green) involving inadequate design control of the RCIC system. Specifically, the design of the RCIC system did not provide adequate minimum flow protection for the RCIC pump. The minimum flow valve, 1(2)-1301-60, would only automatically open on a low flow condition if a RCIC initiation signal was present. The RCIC system is not classified as safety-related, such that the finding was not considered a violation of regulatory requirements. This finding applies to both units.

Description: The inspectors reviewed the control logic for the RCIC minimum flow valve, 1(2)-1301-60. The design of the control logic included automatically opening the valve at a RCIC low flow setpoint of approximately 40 gpm and closing the valve at a setpoint of approximately 80 gpm. However, the inspectors determined that the valve would only automatically open if a RCIC initiation signal was present. Once the reactor vessel level was recovered after a transient, this valve would no longer function as an automatic minimum flow valve. This control logic was consistent with notes in RCIC operator training document LN-1300, AReactor Core Isolation Cooling System.@ Discussions with Quad Cities engineering personnel verified this operational feature as original design.

The inspectors were concerned that this design would not provide adequate minimum flow protection for the RCIC pump. If the operators reduced RCIC flow, as allowed by operating procedures, to prevent overfilling of the reactor vessel during a transient, this design would not provide automatic minimum flow protection. This could potentially result in pump damage due to pump deadheading. In response to this concern, the licensee reviewed plant operating procedures and did not identify any specific operator actions to either manually open the minimum flow valve or stop the RCIC pump under low flow conditions.

The licensee initiated Condition Report 221967, to address this issue. The condition report stated that the severity of the RCIC pump degradation due to deadhead operation was still being evaluated by the vendor, and that the operators had been made aware of this scenario in training and could manually control the system. The condition report also stated that procedure QCAN 901(2)-4 E-16, ARCIC Pump Flow Low,@ was being revised to ensure the operators would immediately trip the RCIC pump in the event of a low flow alarm (40 gpm) during surveillance testing activities. The condition report concluded that the RCIC system remained operable.

During the review of this issue, Quad Cities engineering personnel discovered a related design control issue. A letter from the RCIC pump vendor was being erroneously used as a reference for minimum flow requirements (40 gpm). A June 5, 1974, letter from

Bingham-Willamette Company was thought to be applicable to the Quad Cities RCIC pump and had been used as a reference for several engineering issues. However, the system engineer determined that the letter actually applied to the Clinton Station RCIC pump, which was larger than the Quad Cities pump. Condition Report 224355, was initiated to address this issue. The condition report included actions to obtain formal minimum flow guidance from the pump vendor, verify the minimum allowable flow for sustained RCIC pump operation, verify the minimum flow line orifice sizing, and verify the minimum flow setpoint values. Based on this information, the condition report included actions to initiate any required procedure, design, or plant changes. The vendor information had not been received during the period of the inspection.

Another issue related to the minimum flow valve identified by the inspectors concerned statements in test procedures QCOS 1300-01, APeriodic RCIC Pump Operability Test,@ and QCOS 1300-05, AQuarterly RCIC Pump Operability Test.@ The procedures stated that the RCIC system would not be considered inoperable due to minimum flow valve failing in either the open or closed position. However, the inspectors noted that the RCIC net positive suction head (NPSH) analyses did not account for the increased pump flow (~80 gpm) associated with an open minimum flow valve. Condition Report 224127 was initiated to address this concern. This condition report questioned whether it was appropriate to proceduralize the operability of the RCIC system in a condition outside of its design, because compensatory measures (such as modified allowable NPSH curves) may need to be implemented. The condition report concluded that the NPSH analyses included sufficient margin to ensure RCIC operability.

Analysis: The inspectors determined that failing to provide adequate minimum flow protection for the RCIC pump was a performance deficiency warranting a significance evaluation in accordance with IMC 0612, APower Reactor Inspection Reports,@ Appendix B, AIssue Disposition Screening,@ issued on April 29, 2002. The inspectors determined that the finding was more than minor because it could have affected the mitigating system cornerstone objective of ensuring the availability of systems required to respond to initiating events. Operation of the RCIC pump under low flow conditions could result in pump damage and render the system inoperable. The control logic for the RCIC minimum flow valve was not designed to provide automatic minimum flow protection, and the plant operating procedures did not provide specific directions to control room operators to protect the pump under low flow conditions.

The inspectors completed a significance determination of this issue using IMC 0609, Appendix A, ASignificance Determination of Reactor Inspection Findings for At-Power Situations.@ The inspectors answered Ayes@ to question 1 in the Phase 1 Screening Worksheet under the Mitigating Systems column, because it did not represent an actual degradation of the RCIC system. The specific low flow conditions that could have degraded the pump have not existed. The inspectors reviewed plant data during a past plant transient requiring prolonged RCIC operation and observed that the operators conservatively maintained a minimum RCIC flow of approximately 200 gpm. The inspectors determined that this was a design inadequacy that did not render the system inoperable per Generic Letter 91-18, AInformation to Licensees Regarding Two NRC

Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability.@ As a result, this issue was determined to be of very low safety significance. The licensee also initiated appropriate corrective actions to ensure continued operability and initiated several condition reports to address long-term issues and related concerns.

Enforcement: The RCIC system is classified as a non-safety-related system, and not subject to the design control requirements of 10 CFR Part 50, Appendix B. Therefore, no violation of regulatory requirements occurred. This issue was considered a finding of very low safety significance (FIN 05000254/2004004-01; 05000265/2004004-01). The licensee entered the issue into its corrective action system as Condition Reports 221967, 224355, and 224127.

.2 Reactor Core Isolation Cooling Torus Suction Valve

Introduction: The inspectors identified a violation of 10 CFR Part 50, Appendix B, Criterion III, ADesign Control.@ Specifically, the design of the RCIC system did not provide adequate capability to isolate the safety-related torus from the non-safety-related/non-seismic portions of the RCIC system under all conditions. This issue will be classified as an unresolved item pending completion of the licensee=s analysis. This finding applies to both units.

Description: The inspectors reviewed the control logic for the RCIC torus suction valves, 1(2)-1301-25 and 1(2)-1301-26. These motor-operated valves normally provide isolation of the flow path from the torus to the RCIC pump suction header. The RCIC normal water supply was from the contaminated condensate storage tank (CCST). These valves were designed to open in the event of either low CCST level or high torus level, providing automatic transfer of the RCIC pump suction source from the CCST to the torus. This automatic transfer was designed to occur whether the RCIC pump was operating or not. The inspectors noted that the installed design would not allow the operators to override the automatic transfer signal and manually close these valves from the control room. If the operators attempted to manually close the valves, they would automatically reopen based on the valves= control logic.

Valve 1(2)-1301-25 was identified as a primary containment isolation valve in UFSAR Table 6.2-7, and the RCIC piping and components downstream of 1(2)-1301-25 were classified as non-safety-related and non-seismic. The inspectors also noted that the CCST and related level instruments were classified as non-safety-related and non-seismically qualified.

The inspectors were concerned that this design could result in a scenario in which the safety-related torus could not be readily isolated from the non-safety-related/non-seismic portions of the RCIC system. Specifically, a seismic event could potentially result in an actual (or indicated) loss of CCST water level. As a result, valves 1(2)-1301-25 and 1(2)-1301-26 would automatically open as designed. If the seismic event also caused a failure in the non-seismically qualified portion of the RCIC system, water from the torus could potentially drain, or be pumped, into the reactor building RCIC room. Based on the valves= control logic, the operators could not manually close these valves from the control room thereby resulting in a potential significant loss of torus inventory. The loss of torus inventory could potentially degrade the safety-related water supply for emergency core cooling systems (ECCS) required to mitigate the transient. In addition, the operator=s inability to close these valves could adversely affect primary containment integrity. The inspectors noted that this concern had been previously identified by a vendor in the Component Classification Report for the RCIC system (CED-055559, dated January 20, 1998). The issue, however, was not entered into the corrective action program at that time, nor adequately resolved.

In response to these concerns, licensee engineering personnel investigated the design and determined that the control logic for these valves was changed in 1981 under modification M-4-1(2)-81-008. The modification added the automatic transfer feature to the RCIC design in response to a Three Mile Island (TMI) Action Item (NUREG-0737, Item II.K.3.22). The acceptance criteria for this action item stated, in part, A...the capability of remote manual containment isolation shall be retained.@ An NRC letter, dated August 5, 1982, requested verification from the licensee that the acceptance criteria for this action item had been satisfied. In response, a Commonwealth Edison letter, dated September 7, 1982, stated, AAll manual RCIC functions have been retained.@ Based on this information, the NRC determined that the Quad Cities design satisfied the requirements of NUREG-0737, Item II.K.3.22 and issued a safety evaluation on December 29, 1983. The results of this inspection indicated that the licensee lacked sufficient technical reviews to ensure that the remote manual containment isolation function was retained after installation of the RCIC automatic suction transfer modification in 1981.

Subsequent reviews by licensee engineering personnel determined that the RCIC pump suction piping had previously been analyzed and found to be seismically qualified. However, the RCIC pump discharge piping did not appear to have been designed for seismic loads.

The licensee initiated Condition Report 223815 to address this issue. In addition, Operability Evaluation 223815-08 was performed to address continued system operability. The operability evaluation included an action item to revise operating procedures to provide guidance to the operator to override the automatic transfer signal and close these valves if required. This action would require the operators to place finger blocks in relays in the auxiliary electrical equipment room. The action item was completed on May 28, 2004. The operability evaluation also concluded that based on engineering judgement, and by performing a hanger analysis using operability criteria, the RCIC discharge piping would not fail in a seismic event. The operability evaluation concluded that both the primary containment and RCIC system were operable. The licensee stated that additional evaluations were required to determine past operability for the period between 1981 and May 28, 2004, and provide a basis for the final resolution of the issue.

Analysis: The inspectors determined that the failure to maintain remote manual capability for the torus suction isolation valve was a performance deficiency. The loss of torus inventory could have potentially degraded safety-related ECCS systems required to mitigate the transient in that the torus was their safety-related water supply. In addition, the design could have prevented the operators from performing remote manual primary containment isolation, as required by the NUREG-0737, Item II.K.3.22 acceptance criteria. Because the finding affected the reactor safety mitigating system cornerstone objective, the finding is more than minor. The finding was also determined to have potential safety significance greater than very low significance because it could have affected the availability of ECCS systems required to respond to initiating events, as well as the barrier integrity cornerstone objective of maintaining the functionality of containment. The licensee has initiated appropriate compensatory actions to ensure continued operability by initiating an operating procedure to provide adequate guidance to the operator to override the automatic transfer signal and close these valves if required.

Enforcement: Criterion III of 10 CFR Part 50, Appendix B,ADesign Control,@ states, in part, that measures shall be established to assure that the design basis is correctly translated into specifications, procedures, and instructions. Contrary to the above, the modification to the control logic for valves 1(2)-1301-25 did not correctly implement the design basis requirement to maintain remote manual containment isolation capability. This design basis requirement was documented in the NUREG-0737, Item II.K.3.22 safety evaluation, dated December 29, 1983. After the identification of this issue by the inspectors, the licensee implemented appropriate compensatory actions to ensure continued operability. This performance deficiency is considered an unresolved item (URI 05000254/2004004-02; 05000265/2004004-02), pending completion of the licensee=s analysis on the seismic qualification of the RCIC system discharge piping. Since this performance deficiency involves seismic qualification issues, the safety significance will be determined following NRC evaluation of the licensee's review and in accordance with an SDP Phase 3 evaluation.

.3 Components

a. Inspection Scope

The inspectors examined the SSMP and RCIC systems to ensure that component level attributes were satisfied. Specifically, the following attributes of the SSMP and RCIC systems were reviewed:

Equipment/Environmental Qualification: This attribute verifies that the equipment is qualified to operate under the environment in which it expected to be subjected to under normal and accident conditions. The inspectors reviewed design information, specifications, and documentation to ensure that the SSMP and RCIC components were qualified to operate in within the temperatures and radiation fields specified in the environmental qualification documentation.

Equipment Protection: This attribute verifies that the SSMP and RCIC systems are adequately protected from natural phenomenon and other hazards, such as high energy line breaks, floods or missiles. The inspectors reviewed design information, specifications, and documentation to ensure that the SSMP and RCIC systems were adequately protected from those hazards identified in the UFSAR which could impact their ability to perform their safety function.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA2 Problem Identification and Resolution

.1 Review of Condition Reports

a. Inspection Scope

The team reviewed a sample of SSMP and RCIC system problems that were identified by the licensee and entered into the corrective action program. The inspectors reviewed these issues to verify an appropriate threshold for identifying issues and to evaluate the effectiveness of corrective actions related to design issues. In addition, condition reports written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problem 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

Section 1R21 describes that a vendor had identified the issue with the RCIC torus suction valve in 1998, but the licensee had not entered the issue in the corrective action program. Consequently, the concern was never fully evaluated or corrected.

4OA6 Meetings, Including Exits

.1 Exit Meeting

The inspectors presented the inspection results to Mr. T. Tulon and other members of licensee management at the conclusion of the inspection on May 28, 2004. The inspectors determined that proprietary information was reviewed during the inspection. The inspectors confirmed that the proprietary material had been returned to the licensee or indicated it would be handled in accordance with NRC policy on proprietary information.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
T. Tulon, Site Vice President
R. Gideon, Plant Manager
W. Beck, Regulatory Assurance Manager
D. Bolyes, Operations Support Manager
D. Dauzat, Design Engineering/I&C
B. Davenport, Corporate Licensing Engineer
S. Eldridge, Corporate Engineering
A. Fuhs, Regulatory Assurance
A. Lewis, SSMP System Engineer
B. Porter, Senior Engineering Manager
M. Perito, Operations Manager
T. Rush, RCIC System Engineer
T. Scott, Shift Operations Superintendent
P. Simpson, Corporate Licensing Manager
B. Strub, Engineering

Nuclear Regulatory Commission
J. Lara, Chief, Electrical Engineering Branch, Division of Reactor Safety
K. Stoedter, Senior Resident Inspector
M. Kurth, Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000254/2004004-01 05000265/2004004-01	FIN	Failure to Provide Adequate Minimum Flow Protection for the RCIC Pump (Section 1R21.2.b.1)
05000254/2004004-02 05000265/2004004-02	URI	Failure to Provide Adequate Capability to Isolate the Safety-Related Torus from the Non-Seismic Portions of the RCIC system (Section 1R21.2.b.2)

Closed

05000254/2004004-01 05000265/2004004-01	FIN	Failure to Provide Adequate Minimum Flow Protection for the RCIC Pump (Section 1R21.2.b.1)

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.

1R21 Safety System Design and Performance Capability

Calculations

Number	Title	Revision or Date
004-E-005-1301	Quad 2 - MOV Terminal Voltage Calculations (MO 2-1301-62)	Revision 5A
055559(CMED)	Reactor Core Isolation Cooling Component Classification Document (RCIC) CC-QC012	Revision 1