January 24, 2008

EA-06-295

Mr. Michael W. Rencheck
Senior Vice President and
Chief Nuclear Officer
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

SUBJECT: D. C. COOK NUCLEAR POWER PLANT, UNITS 1 AND 2 NRC INTEGRATED INSPECTION REPORT 05000315/2007006; 05000316/2007006

Dear Mr. Rencheck:

On December 31, 2007, the U. S. Nuclear Regulatory Commission (NRC) completed an inspection at your D. C. Cook Nuclear Power Plant, Units 1 and 2. The enclosed report documents the inspection results, which were discussed on January 10, 2008, with you and other members of your staff.

This inspection 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 this inspection, one Severity Level IV Non-Cited Violation and one finding of very low safety significance (Green) were identified. Because of the very low safety significance and because the issue was entered into your corrective action program, the NRC is treating the violation as a Non-Cited Violation in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

If you contest the subject or severity of a Non-Cited Violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector's Office at the D.C. Cook Nuclear Power Plant.

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/

Christine A. Lipa, Chief
Projects Branch 4
Division of Reactor Projects

Docket Nos. 50-315; 50-316
License Nos. DPR-58; DPR-74

cc w/encl: M. Peifer, Site Vice President
J. Gebbie, Plant Manager
G. White, Michigan Public Service Commission
L. Brandon, Michigan Department of Environmental Quality - Waste and Hazardous Materials Division
Emergency Management Division MI Department of State Police
State Liaison Officer, State of Michigan

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Letter to M. Rencheck from C. Lipa dated January 24, 2008

SUBJECT: D. C. COOK NUCLEAR POWER PLANT, UNITS 1 AND 2 NRC INTEGRATED INSPECTION REPORT 05000315/2007006; 05000316/2007006

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-315; 50-316
License Nos:	DPR-58; DPR-74
Report Nos.	05000315/2007006; 05000316/2007006
Licensee:	Indiana Michigan Power Company
Facility:	D. C. Cook Nuclear Power Plant, Units 1 and 2
Location:	Bridgman, MI 49106
Dates:	October 1, 2007, through December 31, 2007
Inspectors:	B. Kemker, Senior Resident Inspector J. Lennartz, Resident Inspector F. Tran, Reactor Engineer M. Holmberg, Senior Reactor Inspector J. Jacobson, Senior Reactor Inspector R. Jickling, Senior Emergency Preparedness Analyst D. Jones, Reactor Inspector J. Neurauter, Senior Reactor Inspector M. Phalen, Health Physicist D. Szwarc, Reactor Inspector A. Garmoe, Reactor Engineer M. Munir, Reactor Inspector
Approved by:	C. Lipa, Chief Projects Branch 4 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000315/2007006, 05000316/2007006; 10/01/2007 - 12/31/2007; D. C. Cook Nuclear Power Plant, Units 1 and 2; Evaluations of Changes, Tests, or Experiments, Event Response.

This report covers a three-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. One Severity Level IV Non-Cited Violation (NCV) and one Green finding were identified by the inspectors. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 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 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a finding of very low safety significance associated with a self-revealed event that resulted in a Unit 1 reactor trip. The licensee failed to correctly evaluate and incorporate the cooling needs of electrical equipment inside the Unit 1 main feedwater pump digital controls system cabinets into the design, which led to the loss of the east main feedwater pump due to overheated power supplies. Immediate corrective actions included replacement of affected power supplies and restoration of cooling to the cabinets. No violation of regulatory requirements was identified.

  The finding was of more than minor significance because this issue was associated with the Equipment Performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during power operations. Specifically, inadequate design consideration for equipment temperature limitations and cooling needs led to the main feedwater pump failure that caused the reactor trip. The finding was of very low safety significance because the finding: (1) did not contribute to the likelihood of a primary or secondary system loss-of-coolant-accident initiator, (2) did not contribute to both the likelihood of a reactor trip AND the likelihood that mitigation equipment or functions would not be available, and (3) did not increase the likelihood of a fire or internal/external flooding event. The inspectors did not identify a cross-cutting area component related to this finding. (Section 4OA3.4)

Cornerstone: Mitigating Systems

- Severity Level IV. The inspectors identified an NCV of 10 CFR 50.59(d)(1) associated with the licensee's failure to perform a 10 CFR 50.59 evaluation for operation of the plant with less than the design basis time allotted for ice condenser ice basket fusion. Specifically, the licensee failed to properly interpret design and licensing basis requirements associated with protection against external events (i.e., seismic) and as a result did not perform a 10 CFR 50.59 evaluation for plant operation with ice baskets that had less than the design basis time allotted for ice fusion. The licensee performed an evaluation of past operability and determined that the ice condenser would have continued to perform its pressure suppression function even with additional ice fall from the potentially unfused ice baskets.

Because this issue affected the NRC's ability to perform its regulatory function, the violation was reviewed under the traditional enforcement process; however, the underlying technical issue was evaluated using the Significance Determination Process. The violation was determined to be of more than minor significance because the inspectors could not reasonably determine that a 10 CFR 50.59 evaluation would not have ultimately required NRC prior approval. The inspectors reviewed the "Seismic, Flooding, and Severe Weather Screening Criteria" screening questions in Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations" and determined that Question No. 3 was applicable. The violation was of very low safety significance because the finding did not involve the total loss of a safety function identified by the licensee through Probabilistic Risk Assessment, Individual Plant Examination of External Events or similar analysis, that contributes to external event initiated core damage accident sequences. The inspectors did not identify a cross-cutting area component related to this finding. (Section 1R02)

REPORT DETAILS

Summary of Plant Status

Unit 1 was operated at or near full power during the inspection period.

Unit 2 was shut down and de-fueled at the beginning of the inspection period for the Cycle 17 refueling outage (U2C17). The licensee performed a reactor startup and synchronized the unit to the grid on November 6, 2007, upon completion of a 53 day refueling outage. Unit 2 reached full power on November 10, 2007, following testing of the new main generator digital control system. The unit was operated at or near full power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity [R]

1R01 Adverse Weather Protection (71111.01)

.1 Winter Seasonal Readiness Preparations

a. Inspection Scope

The inspectors reviewed and assessed activities conducted for the onset of cold weather. The inspectors verified that procedure 12-IHP-5040-EMP-004, "Plant Winterization and De-Winterization," requirements had been completed; toured the east and west main steam enclosure areas to verify that the winterization temporary heating and ventilation modifications were established as required; toured the outside water storage tank areas (refueling water storage tanks, primary water storage tanks, condensate storage tanks, and fire protection water storage tanks) and associated valve houses to verify that piping insulation was installed and not damaged, and that the associated heat trace circuits were operable; toured the Turbine Building, the Fire Pump House and the Lake Screen House to verify that winterization heaters were in service and ventilation modifications were established as required; and, toured the supplemental diesel generators to verify winterization heaters were in service and ventilation dampers sealed tightly. Additionally, the inspectors observed housekeeping conditions around the plant and in the switchyards to verify that materials capable of becoming airborne missile hazards during high wind conditions, or impacting snow removal, were appropriately located and restrained.

The inspectors reviewed selected action requests (AR) related to cold weather problems. The inspectors verified that identified problems were entered into the corrective action program with the appropriate significance characterization and that corrective actions were appropriate and implemented as scheduled.

This inspection constitutes one seasonal inspection sample as defined by Inspection Procedure 71111.01.

b. Findings

No findings of significance were identified.

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

.1 Review of Ice Condenser Ice Basket Ice Fusion Times

a. Inspection Scope

In May 2000, Region III requested the Office of Nuclear Reactor Regulation (NRR) to review an issue identified with the D.C. Cook ice condensers. Specifically, following a refueling outage, the ice in recently reloaded ice baskets did not have storage times demonstrated as sufficient to allow the ice particles to fuse together and prevent ice fallout during a seismic event at power. If sufficient ice were to fall out of the ice baskets during a seismic event, the lower inlet doors of the ice condenser could become blocked and degrade the ice condenser's capability to mitigate the post loss-of-coolant-accident (LOCA) containment pressure buildup. The NRR staff provided a response to Region III in "Donald C. Cook, Units 1 and 2 - TIA 2000-08 Seismic Qualification of Ice at the Donald C. Cook Plant," dated December 29, 2000. Further, the Region III and NRR staff discussed the NRC conclusions on this issue with the licensee's staff in September of 2000. The licensee documented corrective actions for this issue in condition report (CR) 00-04766.

From September 4 through November 5, 2007, the inspectors reviewed the licensee's corrective actions documented in CR 00-04766 for resolution of NRC concerns documented in TIA 2000-08. Because the licensee's activities potentially involved changes to the ice condenser design and licensing basis, the inspectors evaluated the licensee's corrective actions to determine if a safety evaluation pursuant to 10 CFR 50.59 was applicable, and to determine if prior NRC approval was required. The inspectors followed NRC Inspection Procedure 71111.02, with supplemental guidance from, Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, to determine acceptability of the licensee's activities.

Because the licensee did not complete any safety evaluations or screenings, no sample credit was taken toward the samples defined by Inspection Procedure 71111.02.

b. Findings

Lack of Safety Evaluation for Ice Condenser Operation with Insufficient Ice Fusion Time

Introduction

The inspectors identified a Severity Level IV Non-Cited Violation (NCV) of 10 CFR 50.59, "Changes, Tests, and Experiments," having very low safety significance. The licensee failed to perform a 10 CFR 50.59 evaluation for operation of Unit 1 and Unit 2 with ice baskets with less than the design basis time allotted for ice fusion. Specifically, the licensee failed to properly interpret design and licensing basis requirements associated with protection against external events (seismic) and as a result did not perform a 10 CFR 50.59 evaluation for plant operation with ice baskets which had less than the design basis time allotted for ice fusion.

Discussion

On September 27, 2007, the inspectors identified that the licensee had operated the ice condenser in a manner that was inconsistent with the Updated Final Safety Analysis Report (UFSAR) without a written evaluation that provided the basis for the determination that this test or experiment did not require a license amendment. Specifically, the licensee returned the ice condenser to operation following the previous Unit 1 and Unit 2 outages, with recently loaded ice baskets (approximately four weeks) that did not have at least five weeks for ice fusion.

The inspectors noted that WCAP-8110, Supplement 9, "Ice Fallout from Seismic Testing of Fused Ice Baskets," May 13, 1974, was incorporated into Section 5.3 of the UFSAR by reference. WCAP-8810, Supplement 9 stated, in part, that: "The objective of these tests was to determine the ice fallout from 1" x 1" perforated metal baskets, with 64 percent open area, as a result of simulated plant time history seismic disturbances after the baskets have had time for the ice to fuse." This testing demonstrated that ice fallout in excess of 1 percent would not occur from ice baskets subjected to a simulated seismic response spectra (e.g., shaker table testing) with a documented ice fusion time period of about seven weeks for one of two successful test baskets. Although not documented in WCAP-8110, Supplement 9, the licensee had other vendor records and a letter from the Atomic Energy Commission, which indicated that five weeks was the ice fusion time allotted for the other successful ice fallout test documented in the report. UFSAR Section 5.3.5.9.2, "Design Criteria and Codes," Paragraph (b) of the "Interface Requirements" states: "Sufficient clearance is required for the doors to open into the ice condenser. Items considered in this interface are floor clearance, lower support structure clearance and floor drain operation, and sufficient clearance (approximately 6 inches) to accommodate ice fallout in the event of a seismic disturbance occurring coincident with a LOCA [Loss of Coolant Accident]." The licensee's staff stated that based upon its calculations a 1 percent loss of ice from the ice baskets would not fill the 6 inch reservoir behind the lower ice condenser doors.

On March 27, 2000, the licensee documented in Condition Report (CR) 00-04766 that plant procedures had not considered the time required for ice to fuse in the ice baskets from the time that they are loaded until power ascension begins. In Action 3 of this CR, the licensee stated that a reasonable basis existed to conclude that ice basket fallout will remain within acceptable limits when the time allowed for fusion prior to power ascension is less than five weeks. This conclusion was based on calculation EVAL-SD-001009-001, "Evaluation of Seismic Fallout of Ice from Ice Baskets." However, this evaluation assumed only 4 ice baskets per bay (24 bays) were refilled with ice with less than five weeks of ice fusion wait time and a one time limit on the number of ice baskets reloaded was imposed for the Unit 1 ice condenser. In Action 4, the licensee concluded that no changes were needed to plant licensing basis documents or procedures based on Action 3, which relied on conclusions in EVAL-SD-001009-001 and ice basket reloading practice and history.

The licensee's acceptance criteria established in EVAL-SD-001009-001 for a limited number and distribution of ice baskets with less than five weeks ice fusion time was not consistent with the ice fusion time documented for ice basket tests, which met the 1 percent ice fallout criteria established in WCAP-8110, Supplement 9. Also, the licensee had not established limits in maintenance procedures to ensure that the ice condenser remained within this analyzed configuration during the prior Unit 1 and Unit 2 refueling outages. Therefore, the inspectors identified that the licensee failed to evaluate this condition as any activity where a structure, system, or component (SSC) is utilized or controlled in a manner which is either: (i) outside the reference bounds of the design bases as described in the UFSAR, or (ii) inconsistent with the UFSAR in accordance with 10 CFR 50.59. The licensee staff questioned why 10 CFR 50.59 applied to an ice basket reload, which was considered a maintenance activity. The inspectors noted that Section 4.1.2 of NEI 96-07 stated that maintenance procedures must not inadvertently alter the design performance requirements, operation or control of SSCs. In this case, the licensee's maintenance procedures failed to establish the minimum hold period for ice fusion necessary to ensure that the refilled ice baskets met the original design seismic performance criteria (i.e., less than 1 percent loss of ice). Because the licensee failed to properly interpret design and licensing basis requirements associated with protection against external events (i.e., seismic), a 10 CFR 50.59 evaluation had not been completed for plant operation with ice baskets that had less than the design basis time allotted for ice fusion. The licensee entered this issue into its corrective action program (AR 07270054) and stated that they would allow at least a five week wait period for ice fusion prior to restart of Unit 2 from the Cycle 17 refueling outage.

The inspectors reviewed the number and distribution of ice baskets refilled in the last Unit 1 and Unit 2 outages to determine if the licensee had exceeded the number and distribution of ice baskets which had less than a five week wait period for ice fusion. On May 6, 2006, the licensee started up Unit 2 after reloading 166 ice baskets and on November 10, 2006, the licensee started up Unit 1 after reloading 239 ice baskets. For these examples, the most recently filled ice baskets had approximately a four week wait period for ice fusion time. In Unit 2, 17 of 24 ice condenser bays had more than four ice baskets per bay with less than five weeks of ice fusion time at restart (a total of 144 ice baskets reloaded in these 17 bays). In Unit 1, 2 of 24 ice condenser bays had more than four ice baskets per bay with less than five weeks of ice fusion time at restart (a total of 23 ice baskets reloaded in these 2 bays). Therefore, the inspectors determined that the number and distribution of ice baskets for these outages with less than five week wait periods was not within that analyzed in EVAL-SD-001009-001 and thus, the licensee had potentially impaired the ice condenser function for the one week operating period following startup from these outages. For the one week operating period following these unit restarts, the licensee had not established compensatory measures to ensure that during a seismic event, ice fallout from the recently reloaded ice baskets would not block the lower inlet doors of the ice condenser and degrade the ice condenser's capability to mitigate post LOCA containment pressure buildup. The licensee performed an evaluation of past operability (reference AR 00819265), and determined that the ice condenser would have continued to perform its pressure suppression function even with additional ice fallout from these not fully fused ice baskets. The licensee's operability evaluation demonstrated that even with 20 percent ice fallout from the affected ice baskets, the ice buildup would not have blocked the lower inlet doors. The scope of this operability evaluation included evaluation of other potential impacts to the ice condenser function such as the reduction on total ice mass, change in ice distribution, and ice bed channel flow blocking. Overall, the inspectors concluded that the licensee evaluation of ice condenser past operability evaluation was conservative and comprehensive.

Analysis

The inspectors determined that the failure to perform a 10 CFR 50.59 evaluation for operation of the plant with insufficiently fused ice baskets was a performance deficiency because the potential hazards associated with a seismic event were not evaluated. In this case, the licensee failed to properly interpret design and licensing basis requirements associated with protection against external events (i.e., seismic), and as a result did not perform a 10 CFR 50.59 evaluation for periods of plant operation with insufficiently fused ice baskets. Operation with potentially unfused ice baskets represented a vulnerability which may have degraded the ice condenser's pressure suppression function following a seismic event. Based on the licensee's evaluation of past operability, the degraded ice condenser would have continued to perform its pressure suppression function even with additional ice fall from the affected ice baskets. Therefore, the degraded ice baskets did not impair the ice condenser's safety function.

The finding was determined to be of more than minor significance because the inspectors could not reasonably determine that a 10 CFR 50.59 evaluation would not have ultimately required NRC prior approval. Because violations of 10 CFR 50.59 are considered to be violations that potentially impede or impact the regulatory process, they are dispositioned using the traditional enforcement process instead of the Significance Determination Process (SDP). However, if possible, the underlying technical issue is evaluated under the SDP to determine the severity of the violation. In this case, the inspectors determined that the finding would affect the Mitigating Systems cornerstone and completed a significance determination of the underlying technical issue using Inspection Manual Chapter (IMC) 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The "Seismic, Flooding, and Severe Weather Screening Criteria" questions were completed. Because the ice condenser system does not involve a loss of equipment or function specifically designed to mitigate a seismic event, Screening Question No. 3 was applicable. The inspectors answered "no" to Question No. 3, which asked: "Does the finding involve the total loss of any safety function, identified by the licensee through PRA [Probabilistic Risk Assessment], IPEEE [Individual Plant Examination of External Events] or similar analysis, that contributes to external event initiated core damage accident sequences?" Therefore, the issue was screened as Green. Because the licensee's failure to evaluate this issue originated with the resolution CR 00-04766 in the 2000 time-frame, it did not reflect current licensee performance and no cross-cutting aspect was identified.

Enforcement

10 CFR 50.59(d)(1) requires, in part, that the licensee maintain records of changes in the facility, of changes in procedures, and of tests and experiments. These records must include a written evaluation which provides the bases for the determination that the change, test, or experiment does not require a license amendment.

10 CFR 50.59(a)(6) defines a test or experiment not described in the Final Safety Analysis Report to mean any activity where a structure, system, or component is utilized or controlled in a manner which is either: (i) outside the reference bounds of the design bases as described in the UFSAR, or (ii) inconsistent with the analyses or descriptions in the UFSAR.

10 CFR 50.59(c)(1) states, in part, that a licensee may make changes in the facility as described in the UFSAR and conduct tests or experiments not described in the UFSAR without obtaining a license amendment only if: (ii) the change test or experiment does not meet any criteria in paragraph (c)(2) of this section.

10 CFR 50.59(c)(2) states, in part, that the licensee shall obtain a license amendment pursuant to 10 CFR 50.90 prior to implementing a proposed test or experiment which would: (ii) result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR.

UFSAR Section 5.3.5.9.2, "Design Criteria and Codes," Paragraph (b) of the "Interface Requirements" states: "Sufficient clearance is required for the doors to open into the ice condenser. Items considered in this interface are floor clearance, lower support structure clearance and floor drain operation, and sufficient clearance (approximately 6 inches) to accommodate ice fallout in the event of a seismic disturbance occurring coincident with a LOCA."

WCAP-8110, Supplement 9, was incorporated into Section 5.3 of the UFSAR by reference. WCAP-8810, Supplement 9 stated, in part, that: "The objective of these tests was to determine the ice fallout from 1" x 1" perforated metal baskets, with 64 percent open area, as a result of simulated plant time history seismic disturbances after the baskets have had time for the ice to fuse." On March 18, 1974, this analysis recorded an ice basket shaker table test result, which met the 1 percent ice fallout criteria, and the test occurred approximately seven weeks after ice basket loading.

Contrary to the above, on May 6, 2006, for Unit 2, and on November 10, 2006, for Unit 1, the licensee operated the ice condenser in a manner that was inconsistent with the UFSAR without a written evaluation that provided the basis for the determination that this test or experiment did not require a license amendment. Specifically, the licensee returned the ice condenser to operation following these dates with recently loaded ice baskets (approximately four weeks), without waiting seven weeks for ice fusion as recorded for the successful ice fallout test documented in WCAP-8110, Supplement 9. Operation with insufficiently fused ice baskets was inconsistent with the UFSAR in that, it could result in a more than minimal increase in the likelihood of malfunction of the lower inlet ice condenser doors. Specifically, the lower inlet doors may not have opened as designed following a seismic event because excessive ice fallout from unfused ice baskets could have blocked the lower inlet doors, thereby invalidating the UFSAR Section 5.3.5.9.2 conclusion that sufficient lower inlet door clearance existed to accommodate ice fallout in the event of a seismic disturbance occurring coincident with a LOCA. Because the underlying technical issue was of very low safety significance, this violation is being treated as a Non-Cited Violation consistent with Section VI.A of the

NRC Enforcement Policy (NCV 05000315/316/2007006-01). The licensee entered this violation into its corrective action program as AR 07270054.

.2 Unit 2 Reactor Vessel Closure Head (RVCH) Replacement (71007)

a. Inspection Scope

From June 11 through June 15, 2007, and from June 25 through June 29, 2007, the inspector reviewed licensee documents for the design changes associated with the Unit 2 RVCH replacement to determine, for each change, whether the requirements of 10 CFR 50.59 had been appropriately applied. Specifically, the inspector reviewed modification 2-MOD-55516, "Replace Unit 2 Reactor Vessel Closure Head (2-OME-1)," which included a review of the function of each changed component, the change description, and the scope of one 10 CFR 50.59 screening for the following changes:

- new RVCH constructed from a single piece forging;
- new RVCH J-grove weld profile;
- elimination of twelve spare "dummy" penetrations;
- elimination of seven part length control rod drive mechanism (CRDM) penetrations;
- new CRDM mechanical assemblies;
- new thermocouple column sealing assemblies (TECSA) replace core exit thermocouple column assemblies;
- new dedicated reactor vessel head vent (RVHV) penetration nozzle;
- modification of o-ring retainer clip assembly; and
- the use of Inconel Alloy 600 was prohibited in fabrication of the new RVCH. For example, the penetration tube material was changed from Inconel Alloy 600 to Inconel Alloy 690 which is more resistant to primary water stress corrosion cracking.

The inspector also reviewed one 10 CFR 50.59 screening associated with changes to the Unit 2 enhanced service structure (ESS) to determine, for each change, whether the requirements of 10 CFR 50.59 had been appropriately applied. Specifically, the inspector reviewed modification 2-MOD-55002, "Install Unit 2 Replacement Reactor Vessel Closure Head (RVCH) and Modify the Existing Unit 2 Service Structure (2-OME-1)," which included a review of the function of each changed component, the change description, methods of analysis, and the scope of the 10 CFR 50.59 screening that included the following changes:

- integral radiation shield design with inspection doors;
- enhanced CRDM flow-path and ductwork;
- replacement CRDM rod position indicator cables;
- replacement RVCH cables;
- replacement RVCH resistance temperature detector cables;
- new replacement RVCH metallic reflective insulation;
- new seismic plates for removed part length CRDMs;
- modification of tripod clevis lift pins and installation of keeper plate;

- revised reactor vessel level instrumentation system and RVHV piping and valve layout; and
- additional fall protection attachment points.

The inspector also reviewed one 10 CFR 50.59 screening associated with removing the existing Unit 2 RVCH from the Containment Building and moving the replacement RVCH through the Auxiliary Building into the Containment Building.

The inspector used, in part, NEI 96-07, to determine acceptability of the completed pre-screenings and screening. 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." 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

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

The inspectors performed partial system walkdowns of the following risk-significant systems:

- Units 1 and 2 Spent Fuel Pool Cooling System following Unit 2 Core Offload
- Unit 2 West Charging System Train during maintenance on the East Charging System Train
- Unit 2 West Component Cooling Water System Train during maintenance on the East Component Cooling Water System Train

The inspectors selected these systems based on their risk significance relative to the reactor safety cornerstones. The inspectors reviewed operating procedures, system diagrams, Technical Specification (TS) requirements, and the impact of ongoing work activities on redundant trains of equipment. The inspectors verified that conditions did not exist that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components were aligned correctly and available as necessary.

In addition, the inspectors verified that equipment alignment problems were entered into the licensee's corrective action program with the appropriate characterization and significance. Selected action requests were reviewed to verify that corrective actions were appropriate and implemented as scheduled.

This inspection constituted three quarterly partial system walkdown inspection samples as defined by Inspection Procedure 71111.04.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

The inspectors performed fire protection tours in the following plant areas:

- Fire Zone 6A, Unit 1 and 2 Auxiliary Building Pipe Tunnel El. 601'
- Fire Zones 102 & 121, Unit 2 Containment Accumulator Enclosures
- Fire Zones 10 & 11, Unit 1 Cable Tunnel Quadrants 3M & 3S
- Fire Zones 24 & 25, Unit 2 Cable Tunnel Quadrants 3M & 3S
- Fire Zone 14, Unit 1 Transformer Room El. 591'
- Fire Zone 20, Unit 2 Transformer Room El. 591'
- Fire Zone 15, Unit 1 CD Diesel Generator Room
- Fire Zone 18, Unit 2 CD Diesel Generator Room

The inspectors verified that transient combustibles and ignition sources were appropriately controlled; and assessed the material condition of fire suppression systems, manual fire fighting equipment, smoke detection systems, fire barriers and emergency lighting units.

In addition, the inspectors verified that fire protection related problems were entered into the licensee's corrective action program with the appropriate characterization and significance. Selected action requests were reviewed to verify that corrective actions were appropriate and implemented as scheduled.

This inspection constituted eight quarterly inspection samples as defined by Inspection Procedure 71111.05.

b. Findings

No findings of significance were identified.

.2 Annual Fire Protection Drill Observation (71111.05A)

a. Inspection Scope

The inspectors observed an unannounced fire drill on December 8, 2007 in the Turbine Building 609' Elevation affecting the Unit 1 west main feedwater pump.

The inspectors assessed the licensee's readiness to respond to and mitigate fires by verifying whether the following aspects were properly performed:

- an appropriate number of fire brigade members arrived at the fire scene in a timely manner with self-contained breathing apparatus and protective clothing properly donned;
- the fire brigade brought sufficient fire-fighting equipment to the scene;
- the fire brigade leader demonstrated effective command and control at the fire scene by assigning tasks to individual brigade members and by providing fire attack strategies including discussing potential hazards in the fire area;
- the Control Room operators followed procedures for verification of the fire and initiation of response, including identification of fire location, dispatching the fire brigade, and sounding alarms;
- Emergency Action Levels were declared and notifications were made in accordance with NUREG 0654 and 10 CFR 50.72;
- the fire-fighting Pre-Fire Plan strategies were effectively utilized;
- fire hoses were laid out without flow restrictions and were of sufficient length to reach the fire area;
- appropriate fire extinguishing agents were effectively utilized;
- the fire brigade checked for fire victims and propagation into other plant areas;
- effective smoke removal operations were simulated in accordance with Pre-Fire Plans and strategies by aligning ventilation in the fire area;
- communications between fire brigade members and between the fire brigade leader and operations personnel were clear, efficient and effective; and
- the fire brigade members entered the fire area in a controlled manner utilizing the two-man rule.

The inspectors also verified whether the fire scenario was appropriately simulated, whether the licensee's pre-planned drill scenario was followed and whether the acceptance criteria for the drill objectives were met. The inspectors observed the post-drill critique to verify that the licensee evaluators appropriately identified performance deficiencies. The inspectors reviewed selected action requests related to fire drills to verify that identified problems were entered into the corrective action program with the appropriate significance characterization. Planned corrective actions were reviewed to verify they were appropriate for the circumstances.

This inspection constituted one annual inspection sample as defined by Inspection Procedure 71111.05.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

.1 Annual Resident Inspector Heat Sink Performance Inspection (71111.07A)

a. Inspection Scope

The inspectors reviewed the licensee's maintenance activities for the Unit 2 west component cooling water heat exchanger and the Unit 2 west containment spray heat exchanger. The inspectors assessed the as-found and as-left condition of the heat exchangers by direct observation and document reviews to verify that no deficiencies existed that would adversely impact the heat exchangers' ability to transfer heat to the essential service water system and to ensure that the licensee was adequately addressing problems that could result in initiating events that would cause an increase in risk. The inspectors observed portions of inspection and cleaning activities, and reviewed documentation to verify that the inspection acceptance criteria specified in procedure 12-EHP-8913-001-002, "Heat Exchanger Inspection," Revision 1 were satisfactorily met.

This inspection constituted two annual inspection samples as defined by Inspection Procedure 71111.07.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities (71111.08)

.1 Piping Systems ISI

a. Inspection Scope

From September 24 through October 4, 2007, the inspectors conducted a review of the implementation of the licensee's Risk-Informed Inservice Inspection Program (RI-ISI) for monitoring degradation of the reactor coolant system (RCS) boundary, and the risk significant piping system boundaries. The inspectors selected the licensee's RI-ISI Program components, and American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI required examination of Code components in order of risk priority as identified in Section 71111.08-03 of the inspection procedure, based upon the ISI activities available for review during the on-site inspection period.

The inspectors observed the following two types of nondestructive examination activities to evaluate compliance with the ASME Code Section XI and Section V requirements and to verify that the indications and defects (if present) were dispositioned in accordance with the ASME Code Section XI requirements.

- Ultrasonic Examination (UT) of feedwater elbow to pipe weld 2-FW-70-03S,
- UT of feedwater pipe to elbow weld 2-FW-71-02S,
- Visual Examination (VT-3) of main steam line support 2-MS-92-07S-PS,
- UT of safety injection elbow to pipe weld 2-SI-59-03, and
- UT of safety injection pipe to elbow weld 2-SI-59-02.

The inspectors requested examinations completed during the previous Unit 2 outage with relevant/recordable conditions/indications that were accepted for continued service to verify that the licensee's acceptance was in accordance with the Section XI of the ASME Code. No relevant indications accepted for continuous service from the previous outage were identified.

The inspectors reviewed pressure boundary welds for Class 1 or 2 Systems, which were completed since the beginning of the previous Unit 2 refueling outage to determine if the welding acceptance and pre-service examinations (e.g., VT, Liquid Penetrant Testing, and weld procedure qualification tensile tests) were performed in accordance with ASME Code, Sections III, V, IX, and XI requirements. Specifically, the inspectors reviewed documentation for welds associated with the following work activities:

- Installation of suction piping following centrifugal charging pump 2-PP-50E replacement; and
- Installation of discharge piping following centrifugal charging pump 2-PP-50E replacement.

This inspection constituted one sample as defined by Inspection Procedure 71111.08.

b. Findings

No findings of significance were identified.

.2 Unit 2 Reactor Vessel Closure Head Penetration Inspection Activities

The licensee replaced the Unit 2 RVCH during the refueling outage and therefore was not required to perform head examinations. Inspection of the RVCH replacement activities was performed per Inspection Procedure 71007, "Reactor Vessel Head Replacement Inspection," and is documented in Section 1R02.2 of this inspection report. Hence, this inspection sample was not available for review.

.3 Boric Acid Corrosion Control ISI

a. Inspection Scope

Following shutdown, the inspectors reviewed a sample of boric acid corrosion control visual examination activities through direct observation. This walkdown included the lower Containment Building inner volume and annulus, and was completed on September 15, 2007, with Unit 2 in Modes 4 and 5.

The inspectors reviewed the engineering evaluations performed for the following components to ensure that ASME Code wall thickness requirements were maintained:

- AR 06092012, Reactor Vessel Flange (2-OME-1),
- AR 06096042, Reactor Vessel Closure Head Penetration 42, and
- AR 06087056/04328019, Pressurizer Upper Shell Manway.

The inspectors also reviewed two boric acid leak corrective actions to determine if they were consistent with the requirements of the ASME Code and 10 CFR Part 50, Appendix B, Criterion XVI.

- Repair Leaking Elbow in Boric Acid Storage Tank Room (Work Order 55248356-05), September 26, 2007, and

- 2-RH-142 Replace Upstream Gasket and Bolting Material
  (Work Order 5256974), May 3, 2007.

This inspection constituted two samples as defined by Inspection Procedure 71111.08.

b. Findings

No findings of significance were identified.

.4 Steam Generator (SG) Tube Inspection Activities

a. Inspection Scope

The inspectors performed an on-site review of SG tube examination activities conducted pursuant to TS and the ASME Code Section XI requirements for Unit 2. The NRC inspectors observed acquisition of eddy current (ET) data, interviewed ET data analysts, and reviewed documents related to the SG ISI program to determine if:

- the in-situ SG tube pressure testing screening criteria and the methodologies used to derive these criteria answers were consistent with the Electric Power Research Institute (EPRI) TR-107620, "Steam Generator In-Situ Pressure Test Guidelines;"
- the numbers and sizes of SG tube flaws/degradation identified was bounded by the licensee's previous Unit 2 outage Operational Assessment predictions;
- the SG tube ET examination scope and expansion criteria were sufficient to identify the degradation based on-site and industry operating experience by confirming that the ET scope completed was consistent with the licensee's procedures, plant TS requirements, and EPI 1003138, "Pressurized Water Reactor Steam Generator Examination Guidelines," Revision 6;
- the licensee identified new tube degradation mechanisms;
- the SG tube ET examination scope included tube areas which represented ET challenges such as the tube sheet regions, expansion transitions, and support plates;
- the licensee's implemented repair methods were consistent with the repair processes allowed in the plant TS requirements;
- the required repair criteria were being adhered to;
- the licensee's primary-to-secondary leakage (e.g., SG tube leakage) was below the detection threshold during the previous operating cycle;
- the ET probes and equipment configurations used to acquire data from the SG tubes were qualified to detect the known/expected types of SG tube degradation in accordance with Appendix H, "Performance Demonstration for Eddy Current Examination," of EPRI 1003138, "Pressurize Water Reactor Steam Generator Examination Guidelines," Revision 6;
- the license identified deviations from ET data acquisition or analysis procedures; and
- retrieval attempts of foreign objects were made where practicable. For those objects that were unable to be retrieved, evaluations were performed for the

potential detrimental effects of the objects and appropriate repairs of the affected tubes were planned/taken.

The documents reviewed during this inspection are listed in the attachment to this report.

This inspection constituted two samples as defined by Inspection Procedure 71111.08.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of ISI/SG related problems that were identified by the licensee and entered into the corrective action program, conducted interviews with licensee staff and reviewed licensee corrective action records to determine if;

- the licensee had described the scope of the ISI/SG related problems;
- the licensee had established an appropriate threshold for identifying issues; and
- the licensee had evaluated operating experience and industry generic issues related to ISI and pressure boundary integrity.

The inspectors performed these reviews to ensure compliance with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspectors are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

The inspectors observed a crew of licensed operators during simulator training on November 27, 2007. The inspectors assessed the operators' response to the simulated events focusing on alarm response, command and control of crew activities, communication practices, procedural adherence, and implementation of Emergency Plan requirements. The inspectors also observed the post-training critique to assess the ability of licensee evaluators and operating crews to self-identify performance deficiencies. The crew's performance in these areas was compared to pre-established operator action expectations and successful critical task completion requirements.

This inspection constituted one quarterly inspection sample as defined by Inspection Procedure 71111.11.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

a. Inspection Scope

The inspectors reviewed the licensee's evaluation and management of plant risk for maintenance and emergent work activities affecting risk-significant and safety-related equipment listed below to verify that the appropriate risk assessments were performed prior to removing equipment for work:

- Unit 2 East and West Essential Service Water Pumps
- 765 Kilovolt Switchyard Transformer 4
- Unplanned Loss of Power to Unit 2 Train "A" Electrical Buses

These activities were selected based on their potential risk significance relative to the reactor safety cornerstones. As applicable for each of the above activities, the inspectors reviewed the scope of maintenance work in the plant's daily schedule, reviewed Control Room logs, verified that plant risk assessments were completed as required by 10 CFR 50.65(a)(4) prior to commencing maintenance activities, discussed the results of the assessment with the licensee's Probabilistic Risk Analyst and/or Shift Technical Advisor, and verified that plant conditions were consistent with the risk assessment assumptions. The inspectors also reviewed TS requirements and walked down portions of redundant safety systems, when applicable, to verify that risk analysis assumptions were valid, that redundant safety-related plant equipment necessary to minimize risk was available for use, and that applicable requirements were met.

In addition, the inspectors verified that maintenance risk related problems were entered into the licensee's corrective action program with the appropriate significance characterization. Selected action requests were reviewed to verify that corrective actions were appropriate and implemented as scheduled.

This inspection constituted three samples as defined by Inspection Procedure 71111.13.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following action requests:

- AR 07219065, "Evaluation of the Gaps in the HELB [High Energy Line Break] and Fire Protection Barrier Between the Turbine Building and the Screen House"
- AR 07242031, "Document the Position to Wait Time Specified in WCAP-8110"

- AR 00820364, "Modes 1-4 Aggregate Operability Determination Evaluation for Unit 2"
- AR 00821291, "Unit 2 Containment Spray Pump Breaker Time Was Out of Specification"

The inspectors selected these potential operability issues based on the risk-significance of the associated components and systems. The inspectors verified that the conditions did not render the associated equipment inoperable or result in an unrecognized increase in plant risk. When applicable, the inspectors verified that the licensee appropriately applied TS limitations, appropriately returned the affected equipment to an operable status, and reviewed the licensee's evaluation of the issues with respect to the regulatory reporting requirements. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations.

In addition, the inspectors verified that problems related to the operability of safety-related plant equipment were entered into the licensee's corrective action program with the appropriate characterization and significance. Selected action requests were reviewed to verify that corrective actions were appropriate and implemented as scheduled.

This inspection constituted four samples as defined by Inspection Procedure 71111.15.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17A)

.1 Annual Resident Inspector Review

a. Inspection Scope

The inspectors reviewed the engineering analyses, modification documents and design change information associated with the following permanent plant modifications:

- EC 0000047742, "RHR [Residual Heat Removal] Crosstie Modification for Unit 2"
- EC 0000047800, "Unit 2 Containment Sump Remote Strainer Modification"

During this inspection, the inspectors evaluated the implementation of the design modifications and verified that:

- the compatibility, functional properties, environmental qualifications, seismic qualification, and classification of materials and replacement components were acceptable;
- the structural integrity of the structures, systems and components would be acceptable for accident/event conditions;
- the implementation of the modifications did not impair key safety functions;

- no unintended system interactions occurred;
- the affected significant plant procedures, such as normal, abnormal, and emergency operating procedures, testing and surveillance procedures, and training were identified and necessary changes were completed;
- the design and licensing documents were either updated or were in the process of being updated to reflect the modifications;
- the changes to the facility and procedures, as described in the UFSAR, were appropriately reviewed and documented in accordance with 10 CFR 50.59;
- the system performance characteristics affected by the modification continued to meet the design basis;
- the modification test acceptance criteria were met; and
- the modification design assumptions were appropriate.

Completed activities associated with the implementation of the modifications, including testing, were also inspected and the inspectors discussed the modifications with the responsible engineering and operations staff.

<u>RHR Crosstie Modification</u>

This plant modification was implemented to resolve a long standing issue identified by the NRC in Bulletin 88-04, "Potential Safety-Related Pump Loss." The Bulletin identified a minimum flow design concern in some Westinghouse plants. Specifically, there was a possibility that piping system configurations existed that did not preclude pump-to-pump interaction during minimum flow operation. Therefore, the potential existed for the stronger centrifugal pump to deadhead the weaker pump during low flow operating conditions.

Previously, the licensee's solution to meet the requirements of Bulletin 88-04 was to operate with the RHR pump discharge crosstie valves normally closed. While acceptable, this configuration resulted in a significant reduction of available RHR injection flow under postulated post accident conditions. This plant modification installed check valves in each RHR train downstream of the minimum flow branch line and upstream of the RHR spray branch line. Installation of the check valves allows the crosstie valves to remain open, allowing a single RHR pump to feed all four injection lines and returning the plant to the original emergency core cooling system (ECCS) injection methodology. In addition, the motor operators and internals of the RHR heat exchanger bypass isolation valves were removed from their valve bodies and replaced with manual valve operators and internals. The manual valve operators and internals of the RHR discharge cross-connect line isolation valves were removed from their valve bodies and replaced with motor operators and internals.

<u>Containment Sump Remote Strainer Modification</u>

The inspectors completed this inspection in conjunction with the performance of Temporary Instruction (TI) 2515/166, "Pressurized Water Reactor Containment Sump Blockage (NRC Generic Letter (GL) 2004-02)." The licensee committed to completing plant modifications in its response GL 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors." Refer also to Section 4OA5.1 of this report.

During the fall 2006 Unit 1 refueling outage, the ECCS recirculation sump strainer was replaced with a larger, new design strainer. In addition, the licensee completed other associated physical plant modifications including: removal of calcium silicate insulation from the pressurizer relief tank, pressurizer safety and relief valve pipe, and pressurizer relief tank drain piping inside the crane wall; removal of qualified and unqualified labels in the Containment Building; extension of the front recirculation sump vents using collector boxes; installation of redundant, safety-related level instruments inside the recirculation sump; installation of debris interceptors to protect the drain paths from the containment equalization - hydrogen skimmer fan rooms and at the wide range containment level instrumentation; and, capping of the existing 8" diameter crossover pipe between the recirculation sump and the lower containment sump. The inspectors reviewed this modification and documented the results in NRC Inspection Report 05000315/316/2006007. These same plant modifications were installed in Unit 2 during the fall 2007 refueling outage.

As part of the overall plant design changes to address GL 2004-02, the licensee also installed a remote ECCS strainer in the Unit 2 Containment Building annulus during the fall 2007 refueling outage. The remote strainer is connected to the main recirculation sump via a waterway through the Containment Building crane wall and provides additional surface area for filtration of water during post-accident recirculation phase operation of the RHR and containment spray pumps. This portion of the design was deferred in Unit 1 until the spring 2008 refueling outage. In addition, the licensee completed other associated physical plant modifications in Unit 2 including: flood-up wall debris interceptors, flood-up wall radiation shielding, and the addition of a debris barrier/safety gate in the Containment Building annulus. The inspectors completed a review of the new remote strainer and waterway along with these other modifications during this inspection period.

This inspection constituted two annual inspection samples as defined by Inspection Procedure 71111.17.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed post maintenance testing activities on the following plant equipment to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- Unit 2 Degraded Voltage Protection System Backfit Modification
- Unit 2 AB Emergency Diesel Generator
- Unit 2 West Charging Pump
- Unit 2 CD Emergency Diesel Generator
- Unit 2 Rod Control System
- Unit 2 Pressurizer Power Operated Relief Valve 2-NRV-152

The inspectors reviewed the scope of the work performed and evaluated the adequacy of the specified post maintenance testing. The inspectors verified that the post maintenance testing was performed in accordance with approved procedures, that the procedures clearly stated the acceptance criteria, and that the acceptance criteria were met. The inspectors interviewed operations, maintenance, and engineering department personnel and reviewed the completed post maintenance testing documentation.

In addition, the inspectors verified that problems related to the conduct of post maintenance testing of safety-related plant equipment were entered into the licensee's corrective action program with the appropriate characterization and significance. Selected action requests were reviewed to verify that corrective actions were appropriate and implemented as scheduled.

This inspection constituted six samples as defined by Inspection Procedure 71111.19.

b. Findings

No findings of significance were identified.

1R20 Outage Activities (71111.20)

.1 Refueling Outage Activities

a. Inspection Scope

The inspectors evaluated the licensee's conduct of U2C17 refueling outage activities to assess the licensee's control of plant configuration and management of shutdown risk. The inspectors reviewed configuration management to verify that the licensee maintained defense-in-depth commensurate with the shutdown risk plan; reviewed major outage work activities to ensure that correct system lineups were maintained for key mitigating systems; and observed refueling activities to verify that fuel handling operations were performed in accordance with the TSs and approved procedures. Other major outage activities evaluated included the licensee's control of the following:

- containment penetrations in accordance with the TSs;
- SSCs that could cause unexpected reactivity changes;
- flow paths, configurations, and alternate means for RCS inventory addition and control of SSCs which could cause a loss of inventory;
- RCS pressure, level, and temperature instrumentation;
- spent fuel pool cooling during and after core offload;
- switchyard activities and the configuration of electrical power systems in accordance with the TSs and shutdown risk plan; and
- SSCs required for decay heat removal.

The inspectors observed portions of the plant cooldown, including the transition to shutdown cooling to verify that the licensee controlled the plant cooldown in accordance with the TSs. The inspectors observed operators drain the RCS to mid-loop conditions to accommodate vacuum fill of the RCS near the end of the refueling outage to verify that means of adding inventory to the RCS were available, sufficient indications of RCS water level were operable, and perturbations to the RCS were avoided. The inspectors also observed portions of the restart activities including plant heat up and initial criticality to verify that TS requirements and administrative procedure requirements were met prior to changing operational modes or plant configurations. Major restart inspection activities performed included:

- verification that RCS boundary leakage requirements were met prior to entry into Mode 4 and subsequent operational mode changes;
- verification that containment integrity was established prior to entry into Mode 4;
- inspection of the Containment Building, including the ice condenser, to assess material condition and search for loose debris, which if present could be transported to the containment recirculation sumps and cause restriction of flow to the ECCS pump suctions during LOCA conditions; and
- verification that the material condition of the Containment Building and ECCS recirculation sumps met the requirements of the TSs and was consistent with the design basis.

The inspectors interviewed operations, engineering, work control, radiological protection, and maintenance department personnel and reviewed selected procedures and documents.

In addition, the inspectors reviewed a sample of issues that the licensee entered into the corrective action program to verify that identified problems were being entered with the appropriate characterization and significance. The inspectors also reviewed the licensee's corrective actions for refueling outage issues documented in selected action requests.

This inspection constituted one refueling outage inspection sample as defined by Inspection Procedure 71111.20.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the test results for the following surveillance testing activities to determine whether risk-significant systems and equipment were capable of performing their intended safety function and to verify that the testing was conducted in accordance with applicable procedural and TS requirements:

- 12-EHP-4030-010-262, "Ice Condenser Surveillance and Operability Assessment"
- 2-EHP-4030-234-203, "Unit 2 LLRT [Local Leak Rate Testing]"
- 1-OHP-4030-156-017E, "East Motor Driven Auxiliary Feedwater System Test"

The inspectors observed selected portions of the test activities to verify that the testing was accomplished in accordance with plant procedures. The inspectors reviewed the test methodology and documentation to verify that equipment performance was consistent with safety analysis and design basis assumptions, and that testing acceptance criteria were satisfied.

In addition, the inspectors verified that surveillance testing problems were entered into the licensee's corrective action program with the appropriate characterization and significance. Selected action requests were reviewed to verify that corrective actions were appropriate and implemented as scheduled.

This inspection constituted one in-service testing sample, one containment isolation valve testing sample, and one ice condenser system testing sample as defined by Inspection Procedure 71111.22.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed temporary plant modifications implemented by the licensee using the following plant procedures:

- 12-THP-6010-RPP-004, "Installation of Temporary RP [Radiation Protection] Monitoring Equipment in Containment During Modes 3 & 4"
- 2-OHP-4021-002-013, "Reactor Coolant System Vacuum Fill"

The inspectors interviewed engineering and operations department personnel, and reviewed the design documents and applicable 10 CFR 50.59 evaluations to verify that TS and the UFSAR requirements were satisfied. The inspectors reviewed documentation and conducted plant walkdowns to verify that the modification was implemented as designed and that the modification did not adversely impact system operability or availability.

The inspectors also reviewed a sample of action requests pertaining to temporary modifications to verify that problems were entered into the licensee's corrective action program with the appropriate significance characterization and that corrective actions were appropriate.

This inspection constituted two samples as defined by Inspection Procedure 71111.23.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness [EP]

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspectors performed a screening review of Revisions 21, 22, 23, 24, and 25 of the D.C. Cook Nuclear Power Plant Emergency Plan to determine whether changes identified in Revisions 21, 22, 23, 24, and 25 decreased the effectiveness of the licensee's emergency planning for the D.C. Cook Plant. This review did not constitute an approval of the changes, and as such, the changes are subject to future NRC inspection to ensure that the emergency plan continues to meet NRC regulations.

This inspection constituted one sample as defined by Inspection Procedure 71114.04.

b. Findings

No findings of significance were identified.

1EP7 Force-on-Force Exercise Evaluation (71114.07)

.1 Force-on-Force Exercise Evaluation

a. Inspection Scope

The inspectors observed licensee performance during one site emergency preparedness drill in the Technical Support Center. This drill was in conjunction with a Force-on-Force inspection scheduled and observed by the NRC's Office of Nuclear Security and Incident Response and documented in NRC Inspection Report 05000315/316/2007201. The inspectors observed communications, event classification, and event notification activities by the simulated shift manager. The inspectors also observed portions of the post-drill critique to determine whether their observations were also identified by the licensee's evaluators. The inspectors verified that minor issues identified during this inspection were entered into the licensee's corrective action program.

This inspection constituted one sample as defined by Inspection Procedure 71114.07.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Public Radiation Safety [PS]

2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems (71122.01)

.1 Inspection Planning

a. Inspection Scope

The inspectors reviewed the current revision to the licensee's Offsite Dose Calculation Manual (ODCM) and the licensee's Annual Radioactive Effluent Release Reports for calendar years 2005 and 2006, along with selected radioactive effluent release data for year to date 2007. The inspectors reviewed anomalous results reported in those radioactive effluent release reports that were entered into the licensee's corrective action program and resolved. The inspectors determined whether evaluations were completed by the licensee to assess the potential radiological impact of any modifications made to the ODCM since the previous NRC inspection of the effluent control program in 2005. Similarly, the inspectors determined if the ODCM modifications necessitated changes to the effluent radiation monitor alarm setpoints, and if those changes were made, as warranted. The inspectors also reviewed, as applicable, audits, self-assessments and Licensee Event Reports (LERs) that involved unanticipated offsite releases of radioactive effluents. The effluent reports, effluent data, and licensee evaluations were reviewed to determine whether the radioactive effluent control program was implemented as required by the radiological effluent technical specifications (RETS) and the ODCM, to determine if public dose limits resulting from effluents were met, and to determine if any anomalies in effluent release data were adequately understood by the licensee, and were assessed and reported.

The inspectors evaluated the licensee's analyses of any effluent pathways resulting from spills, leaks or abnormal/unmonitored liquid and gaseous effluent discharges over the previous several years. The inspectors also determined whether the licensee had identified those systems and the associated equipment that were potentially vulnerable to leaks of contaminated fluids and whether the licensee had developed adequate mechanisms to identify spills/leaks should they occur. Moreover, the inspectors reviewed the licensee's recently developed plan for assessing the condition of buried piping and systems which carry radioactive fluids.

The inspectors reviewed the ODCM to identify the gaseous and liquid effluent radiation monitoring systems and associated effluent flow paths including in-line flow measurement devices, and reviewed the description of radioactive waste systems and effluent pathways provided in the UFSAR in preparation for the onsite inspection.

The inspectors reviewed the licensee's RETS/ODCM, and the licensee's procedures and/or surveillance activities, to determine whether a program was in-place for identifying and assessing potential spills/leaks.

This inspection constituted one sample as defined by Inspection Procedure 71122.01.

b. Findings

No findings of significance were identified.

.2 Walkdown of Effluent Control Systems, Review of System/Program Modifications, and Instrument Calibrations and Quality Control

a. Inspection Scope

The inspectors walked down the point of discharge liquid and gaseous effluent radiation monitors, particulate/charcoal samplers and the associated flow indicating devices to observe current system configuration with respect to the descriptions in the UFSAR and to determine if isokinetic sampling conditions existed. The inspectors also walked down selected high and locked high radiation areas of the Auxiliary Building, including the Auxiliary Building 573' elevation wall.

The inspectors reviewed the technical justification for changes made by the licensee to the ODCM, as well as changes to the liquid or gaseous radioactive waste system design or operation since the last NRC inspection, to determine whether these changes affected the licensee's ability to maintain effluents as low as reasonably achievable and whether changes made to monitoring instrumentation resulted in non-representative monitoring of effluents. Annual radioactive effluent release reports for the two years preceding the inspection were evaluated for any significant changes (factor of 5) in either the quantities or kinds of radioactive effluents and for any significant changes in offsite dose which could be indicative of problems with the effluent control program.

The inspectors reviewed records of the most recent instrument calibrations (channel calibrations) for each point-of-discharge effluent radiation monitor and for selected effluent flow measurement devices to determine if these monitors had been calibrated consistent with industry standards and in accordance with station procedures, TSs and the ODCM. Specifically, the inspectors reviewed calibration records for the following effluent radiation monitors and selected flow measuring devices:

- Unit 1 & 2 Steam Jet Air Ejector Vent Monitors (SRA 1900/2900);
- Unit 1 & 2 Vent Effluent Monitors (VRS 1500/2500);
- Unit 1 & 2 Gland Seal Exhaust Monitors (SRA 1800/2800);
- Common Unit Liquid Radwaste Effluent Monitor (RRS-1001);
- Unit 1 & 2 East (R-20) and West (R-28) Essential Service Water Monitors;
- Unit 1 & 2 Steam Generator Blowdown Treatment Monitors (R-24); and
- Unit 1 & 2 Steam Generator Blowdown Monitors (R-19).

The inspectors reviewed effluent radiation monitor setpoint bases and alarm values for the point of discharge gaseous effluent radiation monitors to assess their technical adequacy and for compliance with ODCM criteria. The inspectors selectively reviewed gaseous and liquid effluent monitor operational trend data, and discussed with system engineering staff. The trend data was reviewed and discussions were held to determine if the licensee had identified potential effluent monitoring system health issues and had taken actions or developed plans to address identified deficiencies.

The inspectors reviewed chemistry department quality control data for those instrumentation systems used to quantify effluent releases for indications of potential degraded instrument performance. Specifically, the inspectors reviewed the most recent efficiency calibration records and lower limit of detection determinations and selected other quality control data for Chemistry Department gamma spectroscopy systems and for the liquid scintillation counter.

This inspection constituted three samples as defined by Inspection Procedure 71122.01.

b. Findings

No findings of significance were identified.

.3 Effluent Release Packages, Abnormal/Unmonitored Releases, and Dose Calculations

a. Inspection Scope

The inspectors selectively reviewed selected batch liquid effluent release packages and gaseous effluent sampling data for selected periods in 2006 and 2007, including results of chemistry sample analyses, the application of vendor laboratory analysis results for difficult to detect nuclides, and the licensee's effluent release procedures and practices. Also, the inspectors reviewed the methods for calculating the projected doses to members of the public from these releases. These reviews were performed to determine if the licensee adequately applied analysis results in its dose calculations consistent with the methodologies in its ODCM, and to determine if appropriate treatment equipment was used and effluents were released in accordance with the RETS/ODCM requirements.

The inspectors also reviewed the licensee's practices for compensatory sampling during periods of effluent monitor inoperability including extended periods when radiation monitors were out-of-service, to determine if compliance with ODCM action statements was achieved.

The inspectors selectively reviewed monthly and quarterly dose calculations and projections to ensure that the licensee properly calculated the offsite dose from radiological effluent releases and to determine if any RETS/ODCM (i.e., Appendix I to 10 CFR Part 50) design objectives (limits) were exceeded. The inspectors reviewed the D.C. Cook source term data to determine if all applicable radionuclides that were released in effluents were included in the dose calculations, as applicable.

The inspectors reviewed the licensee's 10 CFR 50.75(g) file, which documented historical and more recent spills/leaks of contaminated liquids associated with its operating units that dated back to the site's early operating period. The inspectors selectively reviewed the site's historical spills/leaks with the potential for a radiological impact. The inspectors reviewed the licensee's evaluation of those incidents to assess the adequacy of the licensee's evaluations including the associated projected dose to the public, as applicable. The inspectors reviewed the 2007 study of the hydrogeologic characteristics of the site including the groundwater flow patterns. Additionally, the inspectors reviewed the licensee's recently expanded groundwater monitoring program for detecting potential leaks and spills. These reviews were performed to determine if the licensee had a program for early detection of spills/leaks, understood the site's groundwater flow characteristics and pathways to the environment, and to determine if the licensee had the capability to assess the radiological impact of a future spill/leak should it occur.

The inspectors reviewed the results of the radiochemistry inter-laboratory cross-check comparisons to validate the licensee's analyses capabilities. The inspectors reviewed the licensee's evaluation of any disparate inter-laboratory comparisons and the associated corrective actions for any deficiencies identified, as applicable. In addition, the inspectors reviewed quarterly inter-laboratory comparison data for the licensee's vendor laboratory to assess the analytical capabilities of the vendor laboratory for those difficult-to-detect nuclides specified in the ODCM.

This inspection constituted five samples as defined by Inspection Procedure 71122.01.

b. Findings

No findings of significance were identified.

.4 Ventilation Filter Testing

a. Inspection Scope

The inspectors reviewed the most recent results for both divisions of the Control Room and the Auxiliary Building emergency ventilation system filter testing to determine whether the test methods, frequency, and test results met TS requirements, as provided in ASME Standard N510-1980, "Testing of Nuclear Air Treatment Systems." Specifically, the inspectors reviewed the results of in-place high efficiency particulate air (HEPA) and charcoal absorber penetration/leak tests, laboratory tests of charcoal absorber methyl iodide penetration and in-place tests of pressure differential across the combined HEPA filters/charcoal absorbers.

This inspection constituted one sample as defined by Inspection Procedure 71122.01.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed a Chemistry Department self-assessment, Performance Assurance Department audits, and action requests generated in 2006 and 2007, which focused on the radioactive effluent treatment and monitoring program. The review was performed to determine if identified problems were entered into the corrective action program for resolution. The inspectors also determined if the licensee's problem identification and resolution program, together with its audit and self-assessment activities, were capable of identifying repetitive deficiencies or significant individual deficiencies in problem identification and resolution.

The inspectors reviewed various action requests related to the radioactive effluent treatment and monitoring program, interviewed staff, and reviewed associated licensee evaluations and corrective action documents to determine if the following activities were being conducted in an effective and timely manner commensurate with their importance to safety and risk:

- initial problem identification, characterization, and tracking;
- disposition of operability/reportability issues;
- evaluation of safety significance/risk and priority for resolution;
- identification of repetitive problems;
- identification of contributing causes;
- identification and implementation of effective corrective actions;
- resolution of Non-Cited Violations tracked in the corrective action system; and
- implementation/consideration of risk significant operational experience feedback.

The inspectors also reviewed the scope of the licensee's audit program relative to the occupational and public radiation safety cornerstones to verify that it meets the requirements of 10 CFR 20.1101(c).

This inspection constituted one sample as defined by Inspection Procedures 71121.02 and 711122.03

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Review of Submitted Quarterly Data

a. Inspection Scope

The inspectors performed a review of the data submitted by the licensee for the Third Quarter 2007 performance indicators for any obvious inconsistencies prior to its public release in accordance with IMC 0608, "Performance Indicator Program."

This inspection was not considered to be an inspection sample as defined by Inspection Procedure 71151.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution (71152)

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

As discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's corrective action system at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Some minor issues were entered into the licensee's corrective action system as a result of inspectors' observations but they are not discussed in this report.

This inspection was not considered to be an inspection sample as defined by Inspection Procedure 71152.

b. Findings

No findings of significance were identified.

.2 Semi-annual Trend Review

a. Inspection Scope

The inspectors reviewed repetitive or closely related issues documented in the licensee's corrective action program to look for trends not previously identified. The inspectors also reviewed action requests regarding licensee-identified trends to verify that corrective actions were effective in addressing the trends and implemented in a timely manner commensurate with the significance.

This inspection constituted one semi-annual trend inspection sample as defined by Inspection Procedure 71152.

b. Assessment and Observations

No findings of significance were identified.

.3 Annual In-Depth Sample Review: Review of Actions Taken to Comply with a Confirmatory Order (EA-06-295) to Address an Alternate Dispute Resolution Agreement

a. Inspection Scope

On April 4, 2007, the NRC issued a Confirmatory Order to the licensee as part of an alternate dispute resolution settlement between the licensee and the NRC regarding an apparent violation of 10 CFR 50.7, "Employee Protection," that was issued to the licensee on December 13, 2006. The inspectors reviewed the completion status of the actions taken by the licensee to comply with the Order.

This inspection constituted one annual review inspection sample as defined by Inspection Procedure 71152.

b. Findings and Observations

No findings of significance were identified. The Confirmatory Order remains open pending further licensee action and NRC review. The following actions from the Confirmatory Order were reviewed:

1. By no later than 180 calendar days after the issuance of this Confirmatory Order, the licensee agrees to complete an assessment of the D.C. Cook Plant's nuclear safety culture including its safety conscious work environment (SCWE).

Result: The licensee completed the safety culture assessment on April 20, 2007. The inspectors reviewed the assessment report and found no significant issues.

2. Within 60 calendar days after the completion of the assessment referenced in paragraph 1 above, the licensee shall make available to the NRC:

   A. A description of the tools/methods used to conduct that assessment including the survey questions;

   B. The results of the assessment and the licensee's analysis of the results; and

   C. The proposed actions, if any, the licensee would plan to take to address the results of the assessment in order to ensure that a thriving SCWE exists at the D.C. Cook Plant.

Result: The safety culture assessment was made available to the inspectors for review on June 13, 2007. The inspectors reviewed the assessment report and found no significant issues. The inspectors also reviewed the actions proposed by the licensee in response to the assessment report and found no significant issues.

3. As expeditiously as possible, but by no later than December 31, 2008, the licensee agrees to complete training of all D.C. Cook Plant's non-supervisory employees and long-term contractors on the topic of SCWE.

Result: (Open) Safety Culture Training (including SCWE) is on-going for all D.C. Cook Plant's non-supervisory employees. The licensee expects to complete the training by March 31, 2008. The inspectors will review this activity in a subsequent inspection.

4. By no later than 60 calendar days after the issuance of this Confirmatory Order, a member of the licensee's management at a level at least equal to the D.C. Cook Site Vice President will communicate with the D.C. Cook Plant workforce about the company's policy and his expectations of management regarding the maintenance and enhancement of a SCWE.

Result: The Site Vice President issued an email communication on May 17, 2007, to the D.C. Cook Plant workforce and also issued a plant newsletter article articulating the licensee's policy and his expectations of management regarding the maintenance of a SCWE. The inspectors reviewed the email communication and newsletter article and found no significant issues.

5. By no later than 90 calendar days after the issuance of this Confirmatory Order, the licensee agrees to implement a periodic assessment of its compliance with its work hour limitations program and evaluate the results of the assessment for trends.

Result: (Open) On April 4, 2007, the licensee updated its administrative procedure that implements its work hour limitations program to incorporate a periodic assessment and trend analysis of the work hour limitations program. The inspectors reviewed PMP-4010-WHL-001, "Working Hour Limitations," Revision 6 and found no significant issues. The inspectors noted, however, that while the licensee has been monitoring its compliance with its work hour limitations program on a monthly basis, it had not yet performed an evaluation of the results of the assessment for trends. The licensee informed the inspectors that the process has been implemented and that the evaluation was expected to be performed on an annual basis beginning in January 2008. The inspectors will review this activity in a subsequent inspection.

.4 Annual In-Depth Sample Review: Review of Selected Inspection Manual 0350 Closure Items

a. Inspection Scope

The inspectors reviewed documentation and selected condition reports related to previously documented IMC 0350 restart action matrix items. The inspectors verified that the documented corrective actions had been completed as prescribed.

This inspection constituted one annual review inspection sample as defined by Inspection Procedure 71152.

b. Findings and Observations

No findings of significance were identified.

4OA3 Followup of Events and Notices of Enforcement Discretion (71153)

.1 Operator Response to Loss of Pressurizer Backup Heaters

a. Inspection Scope

On October 2, 2007, a relay failed in the Unit 1 pressurizer level control circuit that resulted in several Control Room alarms and an RCS pressure transient due to letdown isolation and the loss of all pressurizer heaters. The loss of all pressurizer heaters was beyond the described conditions in TS 3.4.9. Therefore, TS 3.0.3 was entered, requiring restoration within 1 hour or entry into Mode 3 in seven hours. The loss of pressure control resulted in RCS pressure lowering to below the departure from nucleate boiling limit of 2050 pounds per square inch gage (psig). Therefore, TS 3.4.1 was entered, requiring restoration of pressure above the limit within two hours or entry into Mode 2. Operators reduced charging system flow and put excess letdown into service per their abnormal system response procedure to address the rising pressurizer level. Operators were able to energize one train of backup heaters by removing control power fuses from the power supply breaker and manually closing the breaker. RCS pressure had lowered to about 2010 psig when the heaters were energized and then was slowly restored to normal. Technical Specification 3.0.3 was exited when this train of backup heaters was restored. The failed relay is not part of the solid state protection system, but instead is part of the control system.

The inspectors evaluated Control Room operator performance during the event. This evaluation included direct observation in the Unit 1 Control Room, review of the Control Room operators' use of abnormal and normal plant operating procedures, and actions to mitigate the event. The inspectors interviewed plant personnel and reviewed applicable portions of the TSs, plant procedures, Control Room logs, and plant process computer data.

This inspection constitutes one sample as defined by Inspection Procedure 71153.

b. Findings

No findings of significance were identified.

.2 Operator Response to Broken Fuel Pin During Fuel Inspection Activities

a. Inspection Scope

On December 17, 2007, licensee and vendor personnel were conducting fuel inspection activities in the spent fuel pool area. While withdrawing fuel pin C1 from third burned fuel assembly Y-24, fuel handling personnel observed that the fuel pin was severed at approximately the 72-inch mark. However, fuel handling personnel did not observe any fuel pellets coming out of the fuel pin, and indicated radiation levels on all area and process radiation monitors remained normal and stable. In response, the fuel handling personnel stopped the fuel inspection activities and notified the Control Room operators as was planned during the pre-job briefing. Actions specified in abnormal operating procedures for fuel handling accidents were implemented in the spent fuel pool area and in the Control Room, which included evacuating the Auxiliary Building.

Planned recovery actions were developed, which included having radiation protection personnel verify that radiation levels in the spent fuel pool area of the Auxiliary Building remained normal. Fuel handling personnel subsequently placed the 72-inch piece of fuel pin into a storage container in the spent fuel pool. In addition, fuel handling personnel conducted additional inspections to verify that no fuel pellets had come out of the broken fuel pin.

The inspectors evaluated operator performance during the event. The inspectors observed Control Room operators and reviewed plant procedures to verify that the applicable abnormal operating procedures were implemented as required. The inspectors also observed plant meetings in which recovery plans were developed.

This inspection constitutes one sample as defined by Inspection Procedure 71153.

b. Findings

No findings of significance were identified.

.3 Off-Site Contaminated Worker Notification

a. Inspection Scope

The inspectors reviewed the circumstances involving the October 3, 2007, off-site transportation of a potentially contaminated worker to the Lakeland Medical Center (Event Notification Number 43684 and Nuclear Materials Event Database (NMED) Event Number 070605). The inspectors reviewed the incident to ensure that the licensee adequately reported the incident in accordance with the requirement contained in 10 CFR 50.72; that the licensee performed adequate radiological surveys of the individual, the ambulance, and any affected areas of the hospital; and that the licensee implemented adequate measures to ensure that radioactive material was not released into the public domain.

This inspection was not considered to be an inspection sample as defined by Inspection Procedure 71153.

b. Findings

No findings of significance were identified.

.4 (Closed) LER 50-315/2007-001-00: "Unit 1 Automatic Reactor Trip"

a. Inspection Scope

On August 28, 2007, an automatic reactor trip of Unit 1 occurred due to the loss of the east main feedwater pump. The inspectors reviewed the circumstances associated with this event, including the apparent cause evaluation and corrective actions.

This inspection constituted one sample as defined by Inspection Procedure 71153.

b. Findings

Introduction

The inspectors identified a finding of very low safety significance (Green) associated with this self-revealed event. The licensee failed to correctly evaluate and incorporate the cooling needs of electrical equipment inside the Unit 1 main feedwater pump digital controls system (DCS) cabinets into the design, which led to the loss of the east main feedwater pump due to overheated power supplies. No violation of regulatory requirements was identified.

Discussion

On August 28th, the Unit 1 reactor automatically tripped due to low steam generator water level coincident with low feedwater flow on the #11 steam generator (coincident with steam flow/feedwater flow mismatch). The cause was a loss of the east main feedwater pump due to a malfunction of power supplies in the non-safety related DCS. The main feedwater pump DCS cabinet temperature increased due to a loss of cooling and the power supplies for the main feedwater pump controller (two load-sharing power supplies) overheated. The licensee reported this event as a condition that resulted in the automatic actuation of the reactor protection system and auxiliary feedwater system in accordance with 10 CFR 50.73(a)(2)(iv)(A).

A new DCS was installed in Unit 1 for the main turbine generator and the main feedwater pumps during the fall 2006 refueling outage. The DCS electrical equipment cabinets are located on the main operating deck (633' elevation) of the Turbine Building. The Turbine Building is not air-conditioned and the cabinets were not originally provided with cooling to help maintain temperature. On February 9, 2007, the main feedwater pump DCS power supplies exhibited temperature susceptibility due to an elevated ambient temperature approaching 116°F. This resulted in the loss of ability to control voltage, which caused one of the two power supplies to shut down due to a high voltage condition. After the cabinet doors were opened, the affected power supply cooled and began to operate normally and all alarm conditions cleared. It was not recognized at that time that at least one of the two power supplies was left much degraded.

A temporary modification was installed in April 2007 to provide cooling to the Unit 1 DCS cabinets. The cooling system consisted of an air cooler unit with ducting to provide chilled air to the cabinets. A single air cooler unit supplied the main turbine generator DCS cabinets and another air cooler unit supplied the main feedwater pump DCS cabinets. On August 28th, operators responding to an alarm condition discovered that the air cooler unit for the main feedwater pump cabinets was blowing hot air. The condensate level in the bottom of the cooling unit was high, apparently due to a condensate pump outlet fitting failure. This caused the air conditioner to shutdown, but the fan remained running, blowing heated air into the cabinets. Just prior to the trip, the east main feedwater pump's speed decreased as the control valve failed closed. Operators attempted to raise speed, but were not able to do so and an automatic reactor trip occurred.

After the trip, engineers determined that east main feedwater pump DCS cabinet temperature increased from 87°F to 108°F. Visual examination of the power supplies revealed discoloration. The output voltage of the power supplies became erratic and increased due to the heat. At about 33 volts, the servo position controllers that receive power from the two power supplies shut off. This is an over-voltage protection design feature for the servo position controllers. The east main feedwater pump's control valve went shut because both of the servo position controllers shut off. After the cabinet door was opened and the power supplies cooled, the voltage returned to normal (about 28 volts). The power supplies were replaced for both the east and west main feedwater pumps' DCS cabinets. One of the west main feedwater pump's servo position controllers was also found to have shut off, but the other one had not.

The inspectors examined the licensee's apparent cause evaluation and concluded that the evaluation was sufficiently thorough and that corresponding immediate corrective actions appropriately addressed the cause. The licensee's evaluation concluded that design engineers did not adequately analyze the DCS power supply system fully and did not verify the vendor's promise of a robust design. The engineers did not fully understand the power supply operation or limitations; and, therefore did not fully evaluate the conditions under which the system was to operate. Thus, the engineers failed to properly identify the cooling needs of the electrical equipment inside the cabinets, with the power supplies being the most susceptible components. Furthermore, the engineers did not identify potential failure modes of the load-sharing power supply design.

Immediate corrective actions implemented to address this event included the following:

(a) Replacement of the main feedwater pumps DCS power supplies;
(b) Restoration of the temporary cooling unit to operation with changes made to the condensate drain line to prevent future backups; and
(c) Reduction of the cabinet high temperature alarm setpoint to provide earlier warning to operators and allow more time to take corrective action.

Analysis

The inspectors determined that the inadequate design review of the Unit 1 main feedwater pump digital controls system, which resulted in the Unit 1 reactor trip, was a licensee performance deficiency warranting a significance evaluation. The inspectors assessed this finding using the SDP. The inspectors reviewed the samples of minor issues in IMC 0612, "Power Reactor Inspection Reports," Appendix E, "Examples of Minor Issues," and determined that there were no examples related to this issue. Consistent with the guidance in IMC 0612, Appendix B, "Issue Screening," the inspectors determined that the finding was of more than minor significance because this issue was associated with the Equipment Performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during power operations since inadequate design consideration for equipment temperature limitations and cooling needs led to the main feedwater pump failure that caused the reactor trip. The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in IMC 0609, Appendix A, and determined that this finding was a licensee performance deficiency of very low safety significance (Green) because the finding: (1) did not contribute to the likelihood of a primary or secondary system loss-of-coolant-accident initiator, (2) did not contribute to both the likelihood of a reactor trip AND the likelihood that mitigation equipment or functions would not be available, and (3) did not increase the likelihood of a fire or internal/external flooding event. The inspectors did not identify a cross-cutting aspect related to this finding.

Enforcement

No violation of regulatory requirements was identified. This issue is considered to be a finding (FIN 05000315/2007006-02). The licensee entered this finding into its corrective action program as AR 00817858. LER 50-315/2007-001-00 is closed.

4OA5 Other Activities

.1 (Open) Pressurized Water Reactor Containment Sump Blockage (TI 2515/166)

a. Inspection Scope

On September 13, 2004, the NRC issued GL 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors," in response to evolving NRC staff concerns with the adequacy of pressurized water reactor recirculation sump designs. In GL 2004-02, the NRC requested that pressurized water reactor licensees evaluate the potential for post-accident debris to impede or prevent the recirculation functions of emergency core cooling and containment spray systems. The NRC also requested that addressees implement any needed plant modifications to ensure system functionality and stated that all actions should be completed by December 31, 2007.

The objective of Temporary Instruction (TI) 2515/166, "Pressurized Water Reactor Containment Sump Blockage (NRC Generic Letter 2004-02)," was to verify that the actions committed to by the licensee in its GL 2004-02 responses were completed, and where applicable, were programmatically controlled. Specifically, the inspection requirements were to:

(1) verify the implementation of the plant modifications and procedure changes committed to by the licensee in its GL 2004-02 responses;

(2) verify that changes to the facility or procedures, as described in the UFSAR, that were identified in the licensee's GL 2004-02 responses were reviewed and documented in accordance with 10 CFR 50.59; and

(3) verify that the licensee has obtained NRC approval prior to implementing those changes that require such approval as stated in 10 CFR 50.59.

During this inspection period, the inspectors reviewed the licensee's responses to GL 2004-02 to verify that the licensee had completed plant modifications and procedure changes, which it committed to accomplish for Unit 2 during the fall 2007 refueling outage. In addition, the inspectors performed a detailed review of the Unit 2 remote strainer modification and other associated plant modifications using Inspection Procedure 71111.17, "Permanent Plant Modifications." Refer to Section 1R17 of this report.

b. Observations

Summary

The inspectors did not identify any significant discrepancies based upon review of plant modifications and procedure changes completed for Unit 2 to address GL 2004-02.

Evaluation of Inspection Requirements

In accordance with the requirements of TI 2515/166, the inspectors evaluated and answered the following questions:

(1) Did the licensee implement the plant modifications and procedure changes committed to in its GL 2004-02 responses?

Yes. The licensee completed plant modifications and procedure changes for Unit 2 that it committed to accomplish during the fall 2007 refueling outage. This did not, however, complete all of the actions necessary for Unit 2 to achieve full compliance with the requirements in the Applicable Requirements section of GL 2004-02. Therefore TI 2515/166 will remain open for Unit 2. The inspectors reviewed the commitments specifically relevant to the scope of TI 2515/166. The following completed actions and remaining actions were reviewed:

(a) Containment Building walkdowns for determination and/or validation of debris sources including insulation and latent debris.

The initial Unit 2 Containment Building walkdowns were completed during the spring 2006 refueling outage. Final walkdowns were completed during the fall 2007 refueling outage.

(b) Replacement of containment recirculation sump strainers.

During the Unit 2 fall 2007 refueling outage, the main recirculation sump strainer was replaced with a larger, new design strainer. Installation of the new strainer resulted in an increase in surface area from about 85 square feet (ft^2) to about 900 ft^2 and an increase in available flow area through the strainer from about 37 ft^2 to about 270 ft^2. The new design consists of a pocket style strainer. The complex geometry of this type strainer should preclude the formation of a thin bed of fibrous debris that could increase head loss across the strainer. The new strainer should also provide increased margin against blockage or excessive wear of downstream components due to debris in the water and provide increased margin for emergency core cooling and containment spray systems pump suction head and vortexing. The replacement strainer has nominal 1/12" round openings; whereas, the previous strainer consisted of nominal 1/4" square openings in a vertical screen and grating arrangement. The reduction in opening size represents a 300 percent improvement in filtration capability.

During the Unit 2 fall 2007 refueling outage, the licensee also installed a remote strainer in the Unit 2 Containment Building annulus. The remote strainer uses the same pocket style design and is connected to the main recirculation sump via a waterway through the Containment Building crane wall. The remote strainer provides additional surface area (about 1072 ft^2) for filtration of water during post-accident recirculation phase operation of the RHR and containment spray pumps. The available flow area through the remote strainer is about 321 ft^2. When combined with the main recirculation sump strainer, the total available flow area for the containment recirculation sump strainers is about 591 ft^2.

(c) Installation of debris interceptor/trash rack modifications. This included debris interceptors to protect the drain paths from the containment equalization - hydrogen skimmer fan rooms, the existing flow holes from the loop compartment to the annulus through the vent well walls, the approach area to the strainer section in the annulus, and the area of the inlet nozzles for the containment wide range level instruments.

The following debris interceptor/trash rack modifications were installed during the Unit 2 fall 2007 refueling outage:

1. Installation of debris interceptors to protect the drain paths from the containment equalization - hydrogen skimmer fan rooms. This should reduce the potential for debris blockage of these design flow routes.

2. Installation of debris interceptors at the wide range containment level instrumentation. This should prevent plugging the bottom opening of the stilling well piping to ensure reliability of the level instruments.

3. Installation of a debris interceptor box in place of the existing lower containment sump cover plate. The containment equalization - hydrogen skimmer fan room drain lines are routed to the lower containment sump in Unit 2. With the crossover pipe between the recirculation sump and the lower containment sump capped, the new debris interceptor box should allow the flow of water from the fan room drains out of the lower containment sump.

4. Installations of debris interceptors at the flood-up overflow wall core holes. These debris interceptors cover the five flood-up wall core hole openings. This should ensure that larger debris does not block flow at the core hole openings.

5. Installation of an annular barrier with an access gate. The function of this gate is to act as a debris barrier for large pieces of transient debris in the Containment Building annulus to prevent its transport to the remote strainer.

The following additional modifications were also completed during the Unit 2 fall 2007 refueling outage:

1. Modification of the existing flood-up wall openings by chamfering the edges. This should reduce the hydraulic pressure drop across the openings.

2. Modification of the existing steel radiation shield outside the flood-up wall that limits shine through the existing 10" holes so that the bottom plate is cut (or lifted) 2" off the floor. This should provide a path to flush small debris that might otherwise tend to settle between the 10" holes and the shield.

(d) Installation of redundant, safety-related level instruments inside the recirculation sump to provide early indication of sump blockage.

During the Unit 2 fall 2007 refueling outage, redundant, safety-related level instruments were installed inside the recirculation sump to provide early indication of strainer blockage. An associated alarm was installed in the Control Room. This additional instrumentation should aid operators' identification of recirculation sump blockage or air entrainment earlier than solely relying on available indications of emergency core cooling and containment spray systems pump flow rate oscillations and motor amperage swings. Operators may then take action in accordance with their procedures to reduce flow, thus reducing the head loss across the strainer.

(e) Modification of recirculation sump vents to reduce debris screen openings to less than or equal to 1/8".

During the Unit 2 fall 2007 refueling outage, the front recirculation sump vents were extended using collector boxes. These were connected to the existing 6 inch vent line that comes from the rear recirculation sump area and vents above the maximum flood level of the containment. The vent path was also reconfigured to remove the former flat plate design. These changes provide margin against downstream effects by removing potential strainer bypass areas that had a nominal 1/4" opening. The openings are now smaller than the 1/12" opening of the new strainer. Reconfiguration of the front cover vent should also ensure that any air in this section of the sump will be vented outside of the sump.

(f) Modification of the existing cross-over pipe from the recirculation sump to the adjacent lower containment sump.

During the Unit 2 fall 2007 refueling outage, the existing 8" diameter crossover pipe between the recirculation sump and the lower containment sump was capped. This should prevent unfiltered water from bypassing the recirculation sump strainers and entering the recirculation sump. This removed a potential strainer bypass that had a nominal 1/4" opening.

(g) Modification of the Unit 2 lower containment sump to ensure sufficient flow openings exist to allow water to drain from the containment equalization - hydrogen skimmer fan rooms.

During the Unit 2 fall 2007 refueling outage, the containment equalization - hydrogen skimmer fan room drain line check valve internals were removed. This should eliminate a potential single failure mode that could prevent water from the fan rooms from draining to the lower containment sump. An actual modification to the lower containment sump was not needed to address this potential concern.

(h) Removal of the asbestos based calcium silicate insulation currently installed on the pressurizer relief tank and the pressurizer relief valve discharge pipe from the pressurizer enclosure to the pressurizer relief tank.

During the Unit 2 fall 2007 refueling outage, calcium silicate insulation was removed from the pressurizer relief tank, pressurizer safety and relief valve pipe, and pressurizer relief tank drain piping inside the crane wall. This removed 100 percent of the calcium silicate insulation assumed removed in the baseline analysis. No removal of fiberglass insulation was necessary since the Unit 2 containment is essentially fiberglass free.

(i) Removal of unqualified labels inside containment, consisting of vinyl letters for cable tray and conduit identification, and asbestos or asbestos free labels on piping systems.

During the Unit 2 fall 2007 refueling outage, the licensee removed a significant quantity of unqualified labels inside containment. Labels were removed to the extent practical, since some were not accessible without substantial personnel dose accumulation due to scaffold installation. The licensee estimated that 2072.75 in^2 of labels/tags remain and factored this quantity into its analyses.

(j) Implementation of programmatic, process, and procedural changes to ensure that potential sources of debris introduced into containment will be assessed for potential adverse effects on the post-accident recirculation phase operation of the RHR and containment spray pumps.

The licensee updated numerous plant processes and procedures to address debris introduction into the containment and to reflect the above plant modifications. The processes included work control, design change, configuration management, maintenance, and testing. The procedures included maintenance procedures, work control procedures, surveillance test procedures, alarm response procedures, normal operating procedures, and instrument calibration procedures. PMP-2220-SPP-002, "Evaluation and Control of Materials Affecting the Containment Recirculation Sump Protection Program," Revision 0 was issued to document the details associated with the licensee's containment recirculation sump protection program. An update was also made to one of the emergency operating procedures, 2-OHP-4023-ES-1.3, "Transfer to Cold Leg Recirculation," to provide guidance to operators in the event that sump blockage is indicated by the newly installed recirculation sump level instruments. Numerous other procedure changes were identified by the licensee to be completed prior to December 31, 2007. Because many of the procedure changes were completed just before the end of this year, the inspectors did not have the opportunity to review all of them in detail.

(2) Has the license updated its licensing bases to reflect the corrective action taken in response to GL 2004-02?

No. The licensee partially updated its licensing bases to reflect some of the corrective actions taken for Unit 2 in response to GL 2004-02. This did not, however, complete all of the licensing bases updates necessary to achieve full compliance with the requirements in the Applicable Requirements section of GL 2004-02. Additional licensing basis changes for Unit 2 are to be completed in accordance with an approved extension request by the end of May 2008.

The inspectors reviewed the changes identified by the licensee thus far to the UFSAR and the associated 10 CFR 50.59 screenings/evaluations and found no significant issues. The licensee identified changes to the Unit 2 TS and obtained a license amendment to implement those changes. License Amendment 282 was approved by the NRC on October 18, 2007, and was incorporated into the Unit 2 TS. The changes included: (1) revision to TS 3.3.3, "Post Accident Monitoring Instrumentation," to include containment recirculation sump level instrumentation, which will be used for indication of recirculation sump strainer blockage; (2) revision to TS 3.5.2, "ECCS – Operating," to replace the term "trash racks and screens" with the more descriptive term "strainers;" and, (3) revision to TS 3.6.14, "Containment Recirculation Drains," to include Limiting Conditions for Operation, Actions, and Surveillance Requirements to ensure the operability of flow paths credited in the evaluation of potential adverse effects of post-accident debris on the containment recirculation function. No other changes to the plant were identified by the licensee that would require NRC approval prior to implementing as stated in 10 CFR 50.59.

(3) Where an extension past the December 31, 2007 completion date has been approved, document what actions have been completed and what actions are outstanding.

All of the actions discussed above, other than completion of the downstream effects analysis, integrated chemical effects analysis, and GL 2004-02 licensing basis changes needed for final resolution of recirculation sump related issues have been completed for Unit 2. On December 6, 2007, the licensee submitted a request for extension of the completion date for these remaining actions. On December 26, 2007, the NRC staff granted approval of the extension request. Completion of the remaining actions is to be completed by the end of May 2008.

c. Findings

No findings of significance were identified.

.2 Unit 2 Reactor Vessel Closure Head Replacement (71007)

The original penetration nozzles were fabricated from Inconel Alloy 600 material. These nozzles were welded to the RVCH with a partial penetration weld fabricated from Inconel Alloy 600 weld filler metal. In recent years, several pressurized water reactors have experienced pressure boundary leakage caused by primary water stress corrosion cracking of this material.

The design of the Unit 2 replacement RVCH is similar to the original, with some notable exceptions including:

- the new RVCH is constructed from a single piece forging;
- the new RVCH design has an improved J-grove weld profile;
- the new RVCH design eliminates twelve spare "dummy" penetrations;
- the new RVCH design eliminates seven part length CRDM penetrations;
- new CRDM mechanical assemblies;

- new TECSAs replace core exit thermocouple column assemblies;
- the new RVCH design has a dedicated reactor RVHV penetration nozzle;
- the modification of the o-ring retainer clip assembly, and
- the use of Inconel Alloy 600 was prohibited in fabrication of the new RVCH. For example, the penetration tube material was changed from Inconel Alloy 600 to Inconel Alloy 690 which is more resistant to primary water stress corrosion cracking.

a. Inspection Scope

From June 11 through June 15, 2007, from June 25 through June 29, 2007, and from August 27 through August 31, 2007, the inspector reviewed the licensee's design changes associated with the Unit 2 RVCH, CRDM, and TECSA replacement effort. Licensee documents reviewed during the Unit 1 RVCH replacement (refer to NRC Inspection Report 05000315/316/2006007) that were applicable to the Unit 2 RVCH replacement were not reviewed as part of this inspection.

The inspector reviewed certified design specifications, certified design reports, ASME Code reconciliation reports, fabrication contract variation reports and non-conformance reports, and design calculations to confirm that the replacement RVCH, CRDMs, and TECSAs were in compliance with the requirements of Section III, Subsection NB (1995 Edition, including addenda through 1996 Addenda) of the ASME Code. The inspector confirmed that the design specifications and design reports were certified by registered professional engineers competent in ASME Code requirements. The inspector confirmed that adequate documentation existed to demonstrate the certifying registered professional engineers were qualified in accordance with the requirements of Section III (Appendix XXIII of Section III Appendices) of the ASME Code. The inspector also confirmed that the replacement RVCH and CRDM housings were provided as Code NPT stamped components.

b. Findings

No findings of significance were identified.

.3 Unit 2 Reactor Vessel Closure Head Replacement Inspection - Enhanced Service Structure (71007)

Concurrent with the Unit 2 RVCH replacement, the licensee installed enhancements to the existing service structure. Components and modifications implemented for the ESS include:

- integral radiation shield design with inspection doors;
- enhanced CRDM flow-path and ductwork;
- replacement CRDM rod position indicator cables;
- replacement RVCH cables;
- replacement RVCH resistance temperature detector cables;
- revised reactor vessel level instrumentation system (RVLIS) and RVHV piping and valve layout;

- new seismic plates for removed part length CRDMs;
- modification of tripod clevis lift pins and installation of keeper plate:
- new replacement RVCH metallic reflective insulation; and
- additional fall protection attachment points.

a. Inspection Scope

From June 11 through June 15, 2007, from June 25 through June 29, 2007, from August 27 through August 31, 2007, and from October 11 through October 17, 2007, the inspector reviewed the licensee's design changes associated with the installation of the ESS. Documents reviewed for the Unit 1 ESS installation (refer to NRC Inspection Report 05000315/316/2006007) that were also applicable to the Unit 2 ESS installation were not reviewed as part of this inspection. Specifically, the inspector reviewed the ESS design specification, design report, fabrication contract variation reports and non-conformance reports, and a representative sample of design calculations to confirm that ESS structures and components were in compliance with the requirements of the ESS design specification, applicable codes, and applicable standards.

b. Findings

No findings of significance were identified.

.4 Unit 2 Reactor Vessel Closure Head Replacement - Control of Heavy Loads (71007)

a. Inspection Scope

From August 27 through August 31, 2007, and from September 6 through September 19, 2007, the inspector reviewed rigging and load path calculations associated with lifting and moving the old RVCH from inside the Containment Building through the Auxiliary Building and lifting and moving the new RVCH through the Auxiliary Building to inside containment.

In addition, the inspector reviewed licensee documents associated with the RVCH removal and installation during refueling outages utilizing guidance from NRC's Operating Experience Smart Sample (OpESS) FY2007-03, "Crane and Heavy Lift Inspection, Supplemental Guidance for IP-71111.20." Specifically, the inspector reviewed the following:

- licensee submittals and commitments related to GL 80-113 and GL 81-07, "Control of Heavy Loads."
- licensee procedures related to polar crane preventative maintenance, testing, and inspection; the polar crane manufacturer's recommended maintenance, testing, and inspection; and a sample of licensee records of polar crane maintenance, testing and inspection completed prior to reactor disassembly and reactor head lift.
- licensee documents that establish the polar crane rated lift capacity and procedures that control the total weight lifted by the polar crane during reactor vessel head removal and installation during refueling operations.

- licensee calculations for rigging and special lifting devices used to remove and install the reactor vessel head during refueling operations.
- licensee documents that establish the safe load path and procedures that control the established safe load path for removal and installation of the reactor vessel head during refueling operations.
- licensee calculations related to a postulated reactor vessel head drop.
- licensee procedures that remove and install the reactor vessel head during refueling operations with respect to conformance to limiting parameters evaluated in the reactor head drop analysis, i.e., load drop weight, load drop height, and medium through which load drop occurs (air and/or water in the refueling cavity).

b. Findings

No findings of significance were identified.

.5 Unit 2 Reactor Vessel Closure Head Replacement Inspection - Reactor Vessel Head Lift Observations (71007)

a. Inspection Scope

On September 23, 2007, the inspectors observed portions of lifting and moving the old Unit 2 RVCH from the reactor vessel to the storage stand inside containment. The inspectors verified that the lift was conducted in accordance with plant procedures and that the vessel head traveled along the heavy load path specified in 12-OHP-4050-FHP-023, "Reactor Vessel Head Removal with Fuel in the Vessel."

The inspectors also observed portions of lifting and moving the old RCVH from containment to the auxiliary building, and portions of lifting and moving the new reactor vessel closure head from the auxiliary building to the containment equipment hatch. The inspectors verified that the vessel head traveled along the heavy load path as defined in the reactor vessel head modification package 2-MOD-55002, "Install Unit 2 Replacement Reactor Vessel Closure Head and Modify the Existing Unit 2 Service Structure." The inspectors also verified that plant conditions specified in the infrequently performed evolution briefing for lifting and moving the reactor vessel head were established. Specifically, the inspectors verified that the weir gate between the spent fuel pool and the fuel transfer canal was closed; that auxiliary building integrity was established; and, that the spent fuel building exhaust ventilation fans were operating.

For reactor vessel head post-installation inspections, the inspectors toured the reactor cavity on November 5, 2007, with the plant at normal operating temperature and pressure to look for evidence of leakage from the reactor vessel head penetrations; observed rod drop testing and reviewed test data to verify that test acceptance criteria specified in procedure 2-EHP-4030-202-386, "Multiple Rod Drop Measurements," were satisfied; reviewed documentation to verify that test acceptance criteria specified in 2-OHP-4030-212-015, "Full Length Control Rod Operability Test," were satisfied; and, reviewed various RCS leak rate calculations to verify that leak rates were within TS limits.

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

.1 Resident Inspectors' Exit Meeting

The inspectors presented the inspection results to Mr. M. Rencheck and other members of the licensee's staff at the conclusion of the inspection on January 10, 2008. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. Proprietary information was examined during this inspection, but is not specifically discussed in this report.

.2 Interim Exit Meetings

An interim exit meeting was conducted for the Ice Condenser Ice Basket Ice Fusion Time Inspection activities with Mr. J. Gebbie and other members of the licensee's staff on September 27, 2007. The inspectors returned proprietary information reviewed during the inspection and the licensee confirmed that none of the potential report input discussed was considered proprietary.

An interim exit meeting was conducted for the Inservice Inspection activities performed during the Unit 2 refueling outage with Mr. J. Gebbie and other members of licensee's staff on October 4, 2007. The inspectors returned proprietary information reviewed during the inspection and the licensee confirmed none of the potential report input discussed was considered proprietary.

An interim exit meeting was conducted for the Radiation Protection (RETS/ODCM) Inspection with Mr. J. Jensen and other members of the licensee's staff on November 9, 2007.

An interim exit meeting was conducted for Emergency Plan changes with Ms. C. Hutchinson on December 20, 2007.

An interim exit meeting was conducted for the Reactor Vessel Head Replacement Inspection with Mr. J. Jensen and other members of the licensee's staff on December 20, 2007. The licensee confirmed that contractor drawings, calculations, and design reports have been classified as proprietary. All paper and electronic copies of these proprietary documents will be shredded and deleted, respectively.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

S. Adkins, Regulatory Affairs/Licensing Coordinator
J. Anderson, GL 89-13 Program Owner
J. Beer, Staff Health Physicist
R. Crane, Regulatory Compliance Supervisor
P. Donavin, ISI Program Coordinator
B. Evans, Operations Senior License
J. Gebbie, Plant Manager
R. Guilfoyle, Operations
D. Hafer, Reactor Vessel Closure Head Project Manager
J. Harner, Environmental Manager
J. Jensen, Support Services Vice President
D. Fadel, Design Engineering Director
A. Feliciano, Design Engineer Mechanical
D. Foster, Environmental Supervisor
J. Kingseed, Reactor Vessel Closure Head Project
C. Lane, Engineering Programs Manager
Q. Lies, Assistant Plant Manager
R. Lingle, Systems Engineering Manager
W. Mammoser, Design Engineer Mechanical Supervisor
R. Meister, Regulatory Affairs Specialist
C. Moeller, Radiation Protection General Supervisor
P. Monk, Steam Generator Engineer
R. Niedzielski, Regulatory Affairs Specialist
J. Nimtz, Nuclear Regulatory Affairs Compliance Coordinator
M. Peifer, Site Vice President
J. Petro, Regulatory Affairs Manager
R. Pickard, Engineering Programs Supervisor
E. Ridgell, Engineering Programs Manager
S. Simpson, Support Services Director
S. Vasquez, Maintenance Manager
D. Walton, ALARA Supervisor
L. Weber, Plant Engineering Director
J. Zwolinski, Performance Assurance Director

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000315/2007006-01; 05000316/2007006-01	NCV	Lack of Safety Evaluation for Ice Condenser Operation with Insufficient Ice Fusion Time (Section 1R02)
05000315/2007006-02	FIN	Inadequate Design Review of the Unit 1 Main Feedwater Pump Digital Controls System (Section 4OA3.4)

Closed

05000315/2007006-01; 05000316/2007006-01	NCV	Lack of Safety Evaluation for Ice Condenser Operation with Insufficient Ice Fusion Time (Section 1R02)
05000315/2007006-02	FIN	Inadequate Design Review of the Unit 1 Main Feedwater Pump Digital Controls System (Section 4OA3.4)
50-315/2007-001-00	LER	Unit 1 Automatic Reactor Trip (Section 4OA3.4)

Discussed

TI 2515/166	TI	Pressurized Water Reactor Containment Sump Blockage (NRC Generic Letter 2004-02) (Section 4OA5.1)

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspector 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.

1R01 Adverse Weather Protection

- PMP-5055-SWM-001, "Severe Weather Guidelines," Revision 1
- 12-OHP-4022-001-010, "Severe Weather," Revision 5
- PMP-2291-SCH-002, "Work Control Seasonal Readiness Process," Revision 5
- PMI-5055, "Winterization/Summerization," Revision 3
- PMP-5055-001-001, "Winterization/Summerization," Revision 5
- 12-IHP-5040-EMP-004, "Plant Winterization and De-Winterization," Revision 11

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

- CR 00-04766, "Plant Procedures Have Not Considered The Time Required For Ice To Fuse In The Ice Baskets From The Time That They Are Loaded Until Power Ascension Begins"
- AR 07270054, "Potential Violation of 10 CFR 50.59"
- EVAL-SD-001009-001, "Evaluation of Seismic Fallout of Ice From Ice Baskets," Revision 0
- ARA 819265-02, "Ice Condenser Ice Fusion Issue for U2C16 and U1C21"
- AR 00804466, 10 CFR 50.59 Screening No. 2005-0548-01, October 21, 2006
- Design Change 2-MOD-55002, "Install Unit 2 Replacement Reactor Vessel Closure Head (RVCH) and Modify the Existing Unit 2 Service Structure (2-OME-1)," Revision 0
- Design Change 2-MOD-55516, "Replace Unit 2 Reactor Vessel Closure Head," Rev. 0
- 10 CFR 50.59 Screen No. 2005 -0548-01, Document No. ES-MECH-0908-QCN, "Reactor Vessel Closure Head - Licensing Considerations," Revision 0
- 10 CFR 50.59 Screen No.2006-0288-00, Document No. 2-MOD-55516, "Replace Unit 2 Reactor Vessel Closure Head (2-OME-1)," Revision 0
- 10 CFR 50.59 Screen No. 2006-0289-00, Document No. 2-MOD-55002, "D. C. Cook Unit - 2 Enhanced Service Structure (ESS)," Revision 0
- Procedure No. PMP-2350-SES-001, 10 CFR 50.59 Reviews, Revision 5

1R04 Equipment Alignment

- OP-2-5129A, "Flow Diagram OVCS-Reactor Letdown and Changing," Revision 37
- OP-2-5129, "Flow Diagram OVCS-Reactor Letdown and Changing," Revision 49
- OP-2-5135, "Flow Diagram CCW Pumps and CCW Heat Exchangers," Revision 37
- OP-2-5135A, "Flow Diagram CCW Safety Related Loads," Revision 40
- 12-OHP-4021-018-002, Sheet 1, "North Spent Fuel Pit Cooling Loop Alignment," Revision 17
- 12-OHP-4021-018-002, Sheet 2, "South Spent Fuel Pit Cooling Loop Alignment," Revision 17
- 12-OHP-4022-018-001, "Loss of Spent Fuel Pit Cooling," Revision 10
- OP-12-5136-21, "Flow Diagram Spent Fuel Pit Cooling & Clean-Up Unit 1 & 2," Revision 21

1R05 Fire Protection

- Fire Hazards Analysis, Fire Zones 6A, 10, 11, 14, 15, 18, 20, 24, 25, 102, & 121, Revision 13
- Fire Pre-Plan, Fire Areas E, H, I, L, M, V, X, BB, & CCC, Revision 4
- AR 07292016, "FHA Error for Fire Zones 10 and 11"
- AR 07348050, "Update Alarm Response Procedure for Announcing a Plant Fire"
- FB-DR-014, Fire Drill Pre-Plan, December 9, 2007

1R07 Heat Sink Performance

- AR 00818934, "Over Torqued Bolting on 2-HE-18E Hand Held Inspection Cover"
- Work Order 55255481-02, "2-HE-18W Inspect Heat Exchanger," December 12, 2007
- Work Order 55282596-02, "2-HE-15W Remove/Install End Bells, Clean and Inspect," November 13, 2007

1R08 Inservice Inspection (ISI) Activities

- ISI-UT-350, "Acquiring Material Thickness and Weld Contour," Revision 0
- ISI-PDI-UT-1, "PDI Generic Procedure for the UT of Carbon Steel Pipe Welds," Revision 4
- ISI-PDI-UT-2, "PDI Generic Procedure for the UT of Austenitic Pipe Welds," Revision 4
- 12-QHP-5050-NDE-006, "Visual Examinations: VT-1 and VT-3," Revision 3
- 2-EHP-4030-202-001, "Steam Generator Primary Side Surveillance," September 17, 2007
- 12-EHP-5037-SGP-004, "SG Secondary Visual Inspections," November 11, 2005
- Liquid Penetrant Examination Report for WO55248356-04, January 19, 2007
- SGP-DA-U2-C15, "SG Degradation Assessment-Unit 2 Cycle 15," October 12, 2004
- 51-9056553-000, "SG Degradation Assessment- Unit 2 Cycle 17," September 6, 2007
- AR 07271044, "2-GSI-R-660, Discrepancy with as found configuration," September 30, 2007
- AR 07274066" Failed Liquid Penetrant Examination," October 1, 2007
- AR00803744, "FME Found During SG13 Secondary Side Visuals," October 6, 2006
- CR04288004, "Loose Part (most likely a weld rod) Discovered Adjacent to Tube R3-C56 in SG 22 During Post-Sludge Lance Inspection," October 14, 2004
- CR02037035, "Steam Generator Eddy Current and Visual Inspections Identified Foreign Objects on the Secondary Side of Steam Generators 21 and 23," February 6, 2002

1R13 Maintenance Risk Assessments and Emergent Work Control

- PMP-2291-SCH-001, "Work Control Activity Scheduling Process," Revision 19
- PMP-2291-WAR-001, "Work Activity Risk Management Process," Revision 14
- PMP-2291-OLR-001, "On-Line Risk Management," Unit 1 and Unit 2 Part 1 Configuration Risk Assessment, October 8, October 14, and October 25-27, 2007
- PMP-4100-SDR-001, Plant Shutdown Safety and Risk Management, Revision 15
- Control Room Logs, October 8, October 14, and October 25-27, 2007
- Daily Work Activity Schedules, October 8, October 14, and October 25-27, 2007
- ORAM-SENTINAL Risk Management Program
- AR 07288010, "Train A ESF Buses Deenergized for Unknown Reasons"
- 12-OHP-4022-018-001, "Loss of Spent Fuel Pit Cooling," Revision 1

1R15 Operability Evaluations

- AR 00820364, "Modes 1-4 Aggregate Operability Determination Evaluation for Unit 2"
- AR 008210087, "Operability Determination Evaluation for the Aggregate Effects of Non-conservative Values Impacting Control Room Habitability and Offsite Dose Analyses"
- AR 00818959, "Accumulator Temperature Impacting LOCA Analysis"
- AR 00808762, "Offsite and Control Room Dose"
- AR 00814362, "Maximum Operating Temperatures During Recirculation Phase Are Not Accounted for in Calculations for Various Safety Related Systems"
- AR 00820572, "Aggregate Divider Barrier Seal Bypass Evaluation for the Conditions Identified During U2C17"
- AR 00089572, "Water Hammer on Essential Service Water"
- AR 00813401, "Calculation TH-95-01 Does Not Model Gap at Column Line A"
- AR 07219065, "Evaluation of the Gaps in the HELB and Fire Protection Barrier Between the Turbine Building and the Screen House"
- AR 00821291, "U2 CTS Breaker Time Was Out of Spec"

1R17 Permanent Plant Modifications

- EC 0000047742, "RHR [Residual Heat Removal] Crosstie Modification for Unit 2," Revision 0
- EC 0000047800, "Unit 2 Containment Sump Remote Strainer Modification," Revisions 0 and 1
- D. C. Cook Updated Final Safety Analysis Report, Revision 21
- MD-12-DG-011-N, "Effect of EDG Frequency on Pump Break Horsepower," Revision 0
- MD-02-ECCS-005-N, "Unit 2 ECCS Pumps Net Positive Suction Head Analysis," Revision 2
- 2-OHP-4030-208-008R, "ECCS Check Valve Test," Attachment 1, "Residual Heat Removal Suction and Discharge Check Valve Test," Revision 9
- 12-MHP-4030-031-001, "Inspection of the Recirculation Sump," Revision 9
- 2-OHP-4030-001-002, "Containment Inspection Tours," Revision 21
- PMP-4010-CAC-001, "Containment Access Control," Revision 9

1R19 Post Maintenance Testing

- Letter from C. Haney U.S. NRC, to M. Nazar, Indiana Michigan Power Company, "Donald C. Cook Nuclear Plant, Units 1 and 2 - Imposition of Facility Specific Backfit Re: Degraded Voltage Protection System (TAC Nos. MC5735 and MC5736), November 9, 2005
- OP-2-98050-D047412, "Reserve Bus Tran. and Auxiliary Buses Low Voltage Protection Elementary Diagram," December 19, 2006
- OP-2-98041-E047412, "4 KV Aux Transformers 2AB & 201AB Elementary Diagram," December 19, 2006
- OP-2-980411-EO47412, "4 KV Aux Transformers 2AB & 201AB Elementary Diagram," December 19, 2006
- OP-2-98045-D047412, "4 KV/600V Auxiliary Transformers 21A & 21C Elementary Diagram Sheet 1 of 2," December 19, 2006
- OP-2-98046-D047412, "4 KV/600V Auxiliary Transformers 21B & 21D Elementary Diagram Sheet 1 of 2," December 19, 2006
- OP-2-98042-E047412, "4 KV Aux. Transformers 2 CD & 201CD Elementary Diagram," December 19, 2006
- EC-0000047412, "Degraded Voltage Back-Fit Modification," Revision 0 (Description of Change, Section 1.1, Modification Test Plan Summary, Attachment 5)

- EC-0000047412-TP-01, "Unit 2 Degraded Voltage Back-Fit Order Post Modification Test," Revision 0
- WO 55256304-10, 2-OME-150-AB/EN, Perform Diesel Surveillance, October 1, 2007
- WO 55286723-21, 2-CMM-55054-RO, WMO-722 to 724 VT-2, October 3, 2007
- WO 55245807-05, 2-NRV-152, Replace Valve, October 20, 2007
- WO 55294770-01, Replace 2CD EDG Fuel Injector Pumps, October 20, 2007
- 2-OHP-4030-203-052W, Attachment 2, "West CCP Operability Test - West CCP Not Running," Revision 5
- 2-EHP-4030-202-386, "Multiple Rod Drop Measurements," Revision 8, November 5, 2007
- 2-OHP-4030-232-027CD, "CD Diesel Generator Operability Test (Train A)," Attachment 2, "Fast Speed Start," Revision 3, October 23, 2007
- AR 00819723, "2-NRV-152 Failed Its Actuator Drop Test"
- AR 00820892, "2CD EDG Number 1 Front Cylinder Running Cold"
- AR 00820978, "U2 CD EDG Failed to Synchronize Across T21D8 Breaker"
- AR 07295034, "2CD EDG High Cylinder Delta Temperature"
- AR 07295037, "2CD EDG Did Not Meet Acceptance Criteria for 600KW Load Reject"

1R20 Outage Activities

- PMP-4100-SDR-001, "Plant Shutdown Safety and Risk Management," Revision 14
- 2-OHP-4021-002-005, "RCS Draining," Revision 29
- 2-OHP-4030-227-041, "Refueling Integrity," Revision 10
- 2-OHP-4021-001-002, "Reactor Start-Up," Revision 34
- 12-EHP-4040-SNM-302, "Physical Inventory of SNM in ICA II (ICA III)," Revision 2a
- 12-OHP-4050-FHP-001, "Refueling Procedure Guidelines," Revision 16
- 2-OHP-4030-001-002, "Containment Inspection Tours, Revision 21
- 2-OHP-4030-227-037, "Refueling Surveillance," Revision 10
- 2-OHP-4030-214-030, "Daily and Shiftly Surveillance Checks," Data Sheet 20, "LTOP Verification," Revision 12
- 2-OHP-5030-001-002, "Outage Risk and Technical Specification Monitoring," Revision 08
- Unit 2 Control Room Logs, September 15 through 6, 2007
- AR 00819375, "2-IMO-325 As-Found Close Over-thrust Condition"
- AR 00819266, "As-Found Flow Blockage Greater Than 15 percent"
- AR 07304034, "Foreign Material Found in the Unit 2 Ice Condenser"
- AR 00820188, "2-WCR-923 - Found Incorrectly Assembled"
- AR 00820914, "FME Found in Core Area Prior to Core Reload"
- AR 00821164, "FME Found in Core Twice During U2C17 Core Reload"
- AR 00820687, "1.25 Inch Plug in CTS System Leakage"
- AR 00820963, "Excessive Boric Acid Deposits on Reactor Vessel Bottom"
- AR 00819656, "Battery Out of Spec"
- AR 00819793, "Demux Power Supply Failed"
- AR 00819434, "2-DCR-206 Will Not Open"
- AR 00819722, "Non-Accessible Pipe Support"
- AR 00819792, "Leaking Fitting at 2-NFP-210-IH"
- AR 00820984, "Springcan 2-RC-2231F All-Thread is Too Long"
- AR 00821232, "2-NSO-63 Failed Stroke Time Open"
- AR 00819219, "2-CCM-430 Failed to Meet Diagnostic Testing Criteria"
- AR 00818905, "Degraded Voltage Timing Relay Failed to Actuate"
- AR 00821646, "Turbine Casing Safety Valve Lifted During Surveillance"

- AR 00821574, "2-N-43, Signal Cable for Detector A Tested Bad Per IMP.011"
- AR 00819768, "Wire Size for QFR Power Supplies"
- AR 07311057, "Missed Surveillance Requires Use of SR 3.0.3"

1R22 Surveillance Testing

- 12-EHP-4030-010-262, "Ice Condenser Surveillance and Operability Evaluation," Revision 6
- 12-MHP-4030-010-002, "Ice Condenser Flow Channel Surveillance," Revision 7
- AR 07296051, "As-Left Flow Blockage Exceeding 10 percent"
- AR 07269033, "Stuck Ice Baskets"
- AR 07257018, "Ten Heavy Ice Baskets"
- 02-EHP-4030-234-203, "Unit 2 LLRT," Revision 7
- AR 00821237, "2-WCR-942 Will Not Open"
- 2-TDB-2-19.1, "Power Operated Valve Stroke Time Limits," Revision 70
- AR 06358484, "2-WCR-946 Will Not Fully Open and Will Not Stay Open"
- 02-EHP-4030-234-203, Data Sheet 1, "Test Sketch Leak Rate Data Sheet," Revision 7
- 1-OHP-4030-156-017E, "East Motor Driven Auxiliary Feedwater System Test," Revision 2
- AR 00811911, "Flow Switch Requires Calibration"
- AR 00815917, "Flow Switch Reading 80 gpm Without Pump Running"
- OP-1-98217-23, "Motor Driven Auxiliary Feedwater Supply Elementary Diagram," Revision 23
- OP-1-5106A-59, "Flow Diagram Auxiliary Feedwater," Revision 59

1R23 Temporary Plant Modifications

- 12-EHP-5040-MOD-001, "Temporary Modifications," Revision 12
- Drawing OP 2-5663-11, "Unit 2 RCS Loop Details," Revision 11
- 2-OHP-4021-002-013, "Reactor Coolant System Vacuum Fill," Revision 11
- 2-TM-06-07-R0, "Temporary Installation of Vacuum Skid for RCS Fill," Revision 0
- 12-THP-6010-RPP-004, "Installation of Temporary RP Monitoring Equipment in Containment During Modes 3 & 4," Revision 0
- 2-TM-03-15-R0, "Installation of Radiation Protection Related Communications Equipment Inside the Unit 2 Containment While the Unit is in Mode 3," Revision 0
- AR 00818912, "NRC Found Items in Containment Upon Mode 5 Entry"
- AR 07252004, "Containment Foreign Materials Exclusion Issue"
- Work Order 55255975-30, "Install RP Monitoring Cables/Antenna"
- Work Order 55290165-34, "Engineering Review and Provide Requirements for Performing the Final Cross Calibration Checks on the Resistance Thermal Detectors in Lower Containment During Modes 2, 3, and 4"

1EP4 Emergency Action Level and Emergency Plan Changes

- D. C. Cook Nuclear Power Plant Emergency Plan, Revisions 20, 21, 22, 23, 24 and 25

1EP7 Force-on-Force Exercise Evaluation

- EMD-32a Michigan State Police, Nuclear Plant Event Notification, December 12, 2007
- PMP-2080-EPP-100, "Emergency Response," Attachment 8, "Notification of Off-Site Agencies," Revision 9
- PMP-2080-EPP-101, "Emergency Classification," Revision 11

2PS1 Radioactive Gaseous And Liquid Effluent Treatment And Monitoring Systems

- Annual Radioactive Effluent Release Report for Calendar Years 2005 and 2006
- Chemistry Self Assessment (AR 819214, "Radiological Effluent Control Program," October 24, 2007
- Data Sheet of 12-THP-6010-RPI-805, "RMS High Alarm Setpoint for Monitors R-19, R-28, (various periods of 2006 - 2007)
- Efficiency Calibration, LLD Determinations and Quality Control Data for Gamma Spectroscopy Systems No. 2, and 5, various periods of 2007
- "Hydro-geological Evaluation of the DC Cook Nuclear Plant," July 2007
- Monthly Dose Calculations and Dose Projections Due to Liquid and Gaseous Effluents for Calendar Year 2007
- "Performance Assurance Audit PA-07-06," Chemistry, June 15, 2007
- PMP-6010-OSD-001, "Offsite Dose Calculation Manual," Revision 22
- "Results of Radiochemistry Cross Check Program," Fourth Quarter 2006 - Second Quarter 2007
- THI-6020-ADM-001, "Quality Control," Revision 4
- THP-6020-ADM-010, "Analytical Results," Revision 6
- 10 CFR 50.75.g Files, Various
- 12-THP-6010-RPC-817, "Eberline Radiation Monitoring System Mid and High Range Noble Gas Calibration," Revision 5
- 12-THP-6010-RPC-815, "Eberline Radiation Monitoring System Particulate and Low Range Noble Gas Calibration," Revision 0
- 12-THP-6010-RPP-30, "Radiation Protection Actions for Radioactive Material Control," Revision 006
- 12-THP-6010-RPP-601, "Preparation of the Annual Radioactive Effluent Release Report," Revision 007
- 12-THP-6010-RPP-704, "Loss of Control of Radioactive Material," Revision 7
- 12-THP-6020-CHM-308, "Effluent Batch Releases," Revision 8
- 12-THP-6020-CHM-322, "Vent Stack Gaseous Sampling," Revision 9
- 12-THP-6020-INS-026, "Gamma Spectrometry System," Revision 3
- 12-THP-6020-INS-525, "Liquid Scintillation Counter," Revision 7
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- 01/02-IHP-4030-113/213-028, "Steam Jet Air Ejector Monitor Channel Functional Test," 2007
- 01/02-IHP-4030-125/225-001, "Steam Generator Blowdown Liquid Process Radiation Monitor Channel Functional Test," August 2007
- 12-IHP-6030-IMP-040, "Waste Liquid Sample and Discharge Flow Radiation Indication Calibration," May 2006
- 01/02-IHP-4030-119/219-002, "West Essential Service Water Process Radiation Monitor Channel Functional Test," 2007
- 01/02-IHP-4030-119/219-001, "East Essential Service Water Process Radiation Monitor Channel Functional Test," 2007
- 01-IHP-4030-125-002, "Steam Generator Blowdown Treatment Process Radiation Monitor Channel Functional Test," February 2007
- 01/02-IHP-4030-STP-091/191, "Gland Steam Condenser Vent Monitor Channel Functional Test," August 2007

- 1/2-EHP-4030-128-229, "Main Control Room Emergency Ventilation System Surveillance," September and October 2007
- 12-EHP-4030-128-230, "Spent Fuel Pool Exhaust Ventilation System Surveillance," May 2007
- "NCS Corporation Radio-iodine Retention/Penetration Test Reports," October 2006 and May 2007

4OA2 Problem Identification and Resolution

- Letter from C. Carpenter, U.S. NRC, EA-06-295, to M. Nazar, Indiana Michigan Power Company, "Confirmatory Order (Effective Immediately), April 4, 2007
- American Electric Power Donald C. Cook Plant Nuclear Safety Culture Assessment, Conducted by Utilities Service Alliance, March 12 through 16, 2007
- PMP-4010-WHL-001, "Working Hour Limitations," Revision 6
- OHI-4000, "Conduct of Operations," Revision 33
- Generic Letter 82-12, "Nuclear Power Plant Staff Working Hours," June 15, 1982
- Information Notice 91-36, "Nuclear Plant Staff Working Hours," June 10, 1991
- Regulatory Issue Summary 2002-07, "Clarification of NRC Requirements Applicable to Worker Fatigue and Self-Declarations of Fitness-for-Duty," May 10, 2002
- AR 02136071, "NRC Regulatory Issue Summary 2002-07 -Clarification of NRC Requirements Applicable to Worker Fatigue and Self-Declarations of Fitness-for-Duty – Industry Operating Experience"
- AR 00095631, "Tracking for Self-Assessment SA-2005-QAD-001-F, "Bi Annual Safety Culture Assessment"
- CR P-99-07866, "The GL 89-13 Thermal Performance Monitoring Program Needs Upgraded"
- CR P-98-05574, "The GL 89-13 Program Has Been Implemented Thru Individual Systems But Not on a Formal Programmatic Basis"
- CR 97-3523, "The Maximum Fouling Factor Acceptance Criteria for CCW/ESW Heat Exchangers"
- CR P-00-03602, "The RHR, SI and CS Piping Included in the Scope of Large Bore Piping Reconstitution Project"

4OA3 Followup of Events and Notices of Enforcement Discretion

- AR 00819969, "Worker Injured By Falling Clevis During U2R17," October 3, 2007
- AR 00819978, "Medic 1 and First Responders into Protected Area," October 3, 2007
- Shift Manager Log, October 3, 2007
- 12-THP-6010-RPP-900, Revision 19; Shipment Number ENV07-072
- AR 00817858, "Reactor Trip From Apparent DCS Issue"
- Unit 1 Control Room Logs, October 2, 2007
- AR 07276051, "Unit 1 Letdown Isolation Without Pressurizer Heater Control"
- AR 07351046, "Broken Fuel Pin in Spent Fuel Pit"
- 12-OHP-4022-018-006, "Irradiated Fuel Handling Accident in Spent Fuel Storage Area - Control Room Actions," Revision 3
- 12-OHP-4022-018-005, "Irradiated Fuel Handling Accident in Spent Fuel Storage Area - Local Actions," Revision 0

4OA5 Other Activities

TI 2515/166:

- Letter AEP:NRC:5054-11 from J. Jensen, Indiana Michigan Power, to U.S. NRC, "Donald C. Cook Nuclear Plant Units 1 and 2, NRC GL 2004-02 - Information Requested by September 1, 2005," August 31, 2005
- Letter AEP:NRC:5054-14 from J. Jensen, Indiana Michigan Power, to U.S. NRC, "Donald C. Cook Nuclear Plant Units 1 and 2, NRC GL 2004-02 Revision of Commitments," December 19, 2005
- Letter AEP:NRC:6054-05 from J. Jensen, Indiana Michigan Power Company, to U.S. NRC, "Donald C. Cook Nuclear Plant Units 1 and 2, Update to Response to NRC GL 2004-02: Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors," June 27, 2006
- Letter AEP:NRC:6054-06 from J. Jensen, Indiana Michigan Power Company, to U.S. NRC, "Donald C. Cook Nuclear Plant Unit 1, Request for Extension of Completion Date for Unit 1 Actions in Response to GL 2004-02, 'Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors,'" June 27, 2006
- Letter AEP:NRC:6054-07 from J. Jensen, Indiana Michigan Power Company, to U.S. NRC, "Donald C. Cook Nuclear Plant Units 1 and 2, Revision of Commitment for Update of Response to RAI Regarding NRC GL 2004-02," December 19, 2006
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- Letter from P. Tam, U.S. NRC, to M. Nazar, Indiana Michigan Power Company, "Donald C. Cook Nuclear Plant, Unit 1 - Extension of Completion Date for Actions in Response to GL 2004-02," July 28, 2006
- Letter from P. Tam, U.S. NRC, to M. Rencheck, Indiana Michigan Power Company, "Donald C. Cook Nuclear Plant, Units 1 and 2 – GL 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors," Extension Request Approval (TAC NOS. MC4679 and MD4680)," December 26, 2007
- EC 0000047800, "Unit 2 Containment Sump Remote Strainer Modification," Revisions 0 and 1
- 12-MHP-4030-031-001, "Inspection of the Recirculation Sump," Revision 9
- 2-OHP-4030-001-002, "Containment Inspection Tours," Revision 21
- PMP-4010-CAC-001, "Containment Access Control," Revision 9
- 2-OHP-4023-ES-1.3, "Transfer to Cold Leg Recirculation," Revision 7
- PMI-2220, "Foreign Material Exclusion, Cleanliness, and Housekeeping/Material Condition," Revision 15
- PMP-2220-SPP-002, "Evaluation and Control of Materials Affecting the Containment Recirculation Sump Protection Program," Revision 0

Replacement Reactor Vessel Closure Head (IP 71007):
- Design Change 2-MOD-55516, "Replace Unit 2 Reactor Vessel Head (2-OME-1)," Revision 0
- Design Information Transmittal, DIT-B-03153, "Load Conditions and Stress Limits for RVCH Project," Revision 0
- Document No. 08-5051897, "Certified Design Specification: Reactor Vessel Closure Head Replacement, D.C. Cook Units 1 & 2," Revision 4
- Document No. 08-5066942, "Design Specification: Control Rod Drive Mechanism Pressure Housing Assembly for Integrated Latch Housing, Appurtenances ASME III Class 1, D. C. Cook Nuclear Plant Units 1 & 2," Revision 5
- Document No. 08-5070166, "Design Specification: Thermocouple Column Sealing Assembly, D. C. Cook Units 1 & 2," Revision 5
- Document No. 18-2500006, "Technical Document: Reactor Vessel Closure Head Replacements, Control Rod Drive Replacements and Enhanced Service Structure Modifications, Donald C. Cook - Units 1 & 2," Revision 3
- Document No. 32-9001581, "Calculation: D. C. Cook Units 1 & 2, CRDM Nozzle to Adapter Weld and Taper," Revision 3
- Document No. 32-9001581, "Calculation: D. C. Cook Units 1 & 2, CRDM Nozzle to Adapter Weld and Taper" Revision 4
- Document No. 32-9003111, "Calculation: D. C. Cook Units 1 and 2, RVCH Lifting Lug and Support Lug Qualification," Revision 3
- Document No. 32-9003111, "Calculation: D. C. Cook Units 1 and 2, RVCH Lifting Lug and Support Lug Qualification," Revision 6
- Document No. 32-9003249, "Calculation: D. C. Cook Units 1 and 2, RVCH Sizing Calculation," Revision 3
- Document No. 32-9003249, "Calculation: D. C. Cook Units 1 and 2, RVCH Sizing Calculation," Revision 4
- Document No. 32-9003276, "Calculation: D. C. Cook Units 1 and 2, Reactor Vessel Head Vent/RVLIS Nozzle Connection," Revision 3
- Document No. 32-9003276, "Calculation: D. C. Cook Units 1 and 2, Reactor Vessel Head Vent/RVLIS Nozzle Connection," Revision 4
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- Revision 1 Document No. 32-9012285, "Calculation: D. C. Cook Units 1 & 2 Thermocouple Sealing Assembly (TECSA) Analysis," Revision 4
- Document No. 32-9012285, "Calculation: D. C. Cook Units 1 & 2 Thermocouple Sealing Assembly (TECSA) Analysis," Revision 5
- Document No. 32-9024117, Calculation: Cook Unit 2, Design Basis RVCH Limits, Revision 2
- Document No. 32-9024122, "Calculation: D. C. Cook Unit 2, Section XI Appendix G RV Closure Head Limits," Revision 2
- Document No. 33-9009036, "ASME Design Report: D. C. Cook - Units 1&2, Replacement Reactor Vessel Closure Head," Revision 1
- Document No. 33-9013947, "Design Report Addendum: Control Rod Drive Mechanism Pressure Housing Assembly, Appurtenances ASME III Class 1, D. C. Cook Nuclear Plant Units 1 & 2," Revision 4

- Document No. 33-9023774, "ASME Design Report: Thermocouple Column Sealing Assembly (TECSA), D. C. Cook - Units 1 & 2," Revision 5
- Document No. 38-9014278, "ASME Code Reconciliation: Modified L-106A CRDM Appurtenances, D. C. Cook Units 1 & 2," Revision 1
- Document No. 51-9026727, "Engineering Information Record: D. C. Cook Unit 2, Reactor Vessel Closure Head Code Reconciliation," Revision 0
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- Document No. 87-9038405, "Contract Variation Approval Request (CVAR)," Revision 1
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- Document No. JQA-03-361, "Certified Material Test Report: Japan Steel Works, ASME SA-508, Gr. 3, Cl. 1, Heat No. 03W98-1-1, D. C. Cook Unit 2," November 21, 2003
- Drawing No. 02-5022544C, "RV Head Forging Dimensional Outline, Unit 2," Revision 2
- Drawing No. 02-5030807D, "Replacement RVCH - Closure Head Forging Procurement Drawing, D. C. Cook Unit 2," Revision 0
- Drawing No. 02-5051034C, "RV Head Cladding, D. C. Cook Unit 2," Revision 1
- Drawing No. 02-5055842E, "Specification Drawing: Replacement Reactor Vessel Closure Head, D. C. Cook Unit 2," Revision 6
- Drawing No. 02-5056727C, "Thermal Sleeve Assembly, D. C. Cook Units 1 & 2," Revision 4
- Drawing No. 02-5065861D, "CRDM; I.L.H. Assembly Details," D. C. Cook Unit 2 Revision 1
- Drawing No. 02-5068645D, "Thermocouple Nozzle, D. C. Cook Unit 2," Revision 0
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- Drawing No. 02-9008398D, "Thermocouple Column Sealing Assembly, Sheet 3," Rev. 5
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- AR 07294006, 12-OHP-4050-FHP-034 Needs Correction and Further Enhancement"

Replacement Enhanced Service Structure (71007)

- Design Change 2-MOD-55002, "Install Unit 2 Replacement Reactor Vessel Closure Head (RVCH) and Modify the Existing Unit 2 Service Structure (2-OME-1)," Revision 0
- Design Information Transmittal No.: DIT-B-03153, "Load Conditions and Stress Limits for RVCH Project," Revision 0
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- Document No. 18-2500006, "Technical Document: Reactor Vessel Closure Head Replacements, Control Rod Drive Replacements and Enhanced Service Structure Modifications, Donald C. Cook - Units 1 & 2," Revision 3
- Document No. 32-5068858, "Design Report: D. C. Cook Unit 2, ESS," Revision 2
- Document No. 32-9020603, Cook Unit 2 - Reactor Head Vent Line Piping Analysis, Rev. 1
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- Document No. 32-9020605, D.C. Cook U-2 RVHVS Support Design (Head Vent Piping Per 2-MOD-55002), Rev. 0
- Document No. 32-9020608, "D.C. Cook U-2 - RV Head Vent Plant Side Support Evaluation (2-ARC-R5004 & 2-ARC-R665)," Revision 0
- Document No. 32-9020644, "D.C. Cook Unit 2 Evaluation of RVLIS Piping," Revision 1
- Document No. 32-9020650, "D.C. Cook U-2 RVLIS Support Design (2-ARC-R5322, 2-ARC-R5323, 2ARC-R5324 & 2-ARC-R5325)," Revision 0
- Document No. 32-9024311, Cook Unit 2 RVHVS Piping Loads - DC Cook Unit 2, Revision 0
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Control of Heavy Loads (71007)

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- Document No. 32-9016197, "Rigging Calculation: Upend/Downend Frame," Revision 2
- Document No. 32-9016198, "Rigging Calculation: Lifting New/Old heads with Bigge Spreader Bar and Rigging Trusses." Revision 2
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- Document No. 32-9018873," Rigging Calculation: Runway," Revision 2
- Document No. 32-9019055, "Lifting New/Old Heads Horizontal Rigging System," Rev. 6

- Document No. 32-9026717," Rigging Calculation: Lifting New Head w/ Shipping Container," Revision 2
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- Document No. 51-5068632, "Engineering Information Record: D. C. Cook TRD - RVCH Replacement Heavy Rigging & Lifting - Bigge," Revision 2
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- Document No. 51-9019515, "Engineering Information Record: D. C. Cook Unit 2, RVCH Weight Evaluation," Revision 0
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- Job Order No. R0096044, Activity 01; Polar Crane: 1-OME-25, "Perform Visual and Surface Exams," January 16, 2006
- Job Order No. R0231215, Activity 01; Polar Crane: 2-OME-25, "NDE of Lift Rig Critical Welds," June 28, 2002
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- Job Order No. 05342079, Activity 01; Polar Crane: 2-QM-4, "Correct Material Deficiencies," January 25, 2007
- Job Order No. 05342079, Activity 05; Polar Crane: 2-QM-4, "Replace #4 Bridge Drive Brakes," February 3, 2006
- Job Order No. 05342079, Activity 07; Polar Crane: 2-QM-4, "Remove Grease Fittings, Plug, and Label," January 26, 2006
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- Job Order No. 06040075, Activity 01; Polar Crane: 2-QM-4, "Replace Auxiliary Hook," March 16, 2006
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- Work Order No. 55280323 02, "Polar Crane: 1-QM-4, Perform Main Hoist Load Brake Load Test," August 16, 2006
- AR 00125386; Safety Classification for Control Rod Drive; April 20, 2006
- AR 00803829; Incorrect References to ANSI-B30.2.-2001 Versus
- ANSI-B30.2-1976; October 9, 2006
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- AR 00807806; Unit 2 Reactor Head Drop Analysis; January 19, 2007
- AR 00814389; OE: NRC Regulatory Issue Summary 2005-25, Supplement 1, Clarification of NRC Guidelines for Control of Heavy Loads; June 5, 2007
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- AR 00814814, "Administrative Error in Calculation 32-9003276-03," June 15, 2007
- AR 00815812, "U1 Enhanced Service Structure - Calculation Error for Bolt Stress," July 9, 2007
- AR 00818962, "Section Modulus Z Equation for Piping," September, 19, 2007
- AR 00818962-01, "Operability Determination Evaluation (ODE) for the Unit 1 Reactor Head Vent Calculation 32-9004408-003," September 20, 2007

LIST OF ACRONYMS USED

ADAMS	Agency Documents Access and Management System
ALARA	As-Low-As-Is-Reasonably-Achievable
AR	Action Request
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
CR	Condition Report
CRDM	Control Rod Drive Mechanism
DCS	Digital Control System
DRP	Division of Reactor Projects
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EPRI	Electric Power Research Institute
ESS	Enhanced Service Structure
ET	Eddy Current
ft^2	Square Feet
GL	Generic Letter
HEPA	High Efficiency Particulate Air
HELB	High Energy Line Break
IMC	Inspection Manual Chapter
IPEEE	Individual Plant Examination of External Events
ISI	Inservice Inspection
LER	Licensee Event Report
LLRT	Local Leak Rate Testing
LOCA	Loss of Coolant Accident
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NMED	Nuclear Materials Event Database
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
ODCM	Offsite Dose Calculation Manual
PARS	Publicly Available Records
PRA	Probabilistic Risk Assessment
psig	Pounds Per Square Inch Gauge
RCS	Reactor Coolant System
RETS	Radiological Effluent Technical Specification
RHR	Residual Heat Removal
RI-ISI	Risk-Informed Inservice Inspection
RP	Radiation Protection
RVCH	Reactor Vessel Closure Head
RVHV	Reactor Vessel Head Vent
RVLIS	Reactor Vessel Level Instrumentation System
SCWE	Safety Conscious Work Environment
SDP	Significance Determination Process
SG	Steam Generator
SSC	Structure, System, or Component
TECSA	Thermocouple Column Sealing Assembly
TI	Temporary Instruction

TS	Technical Specification
U2C17	Unit 2 Cycle 17
UFSAR	Updated Final Safety Analysis Report
UT	Ultrasonic Examination
VT	Visual Examination