February 12, 2007

EA-06-134

Mr. Mano K. Nazar
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 05000315/2006007;
05000316/2006007

Dear Mr. Nazar:

On December 31, 2006, 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 9, 2007 with Mr. J. Jensen 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 Violation and one finding of very low safety significance (Green), which also involved a violation of NRC requirements, were identified. However, because of the very low safety significance and because the issues were entered into your corrective action program, the NRC is treating the violations as Non-Cited Violations 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

Enclosure: Inspection Report 05000315/2006007; 05000316/2006007
w/Attachment: Supplemental Information

cc w/encl: M. Peifer, Site Vice President
L. Weber, Plant Manager
S. Simpson, Regulatory Affairs 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

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

Enclosure: Inspection Report 05000315/2006007; 05000316/2006007
w/Attachment: Supplemental Information

cc w/encl: M. Peifer, Site Vice President
L. Weber, Plant Manager
S. Simpson, Regulatory Affairs 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

DOCUMENT NAME: C:\FileNet\ML070450022.wpd
☐ Publicly Available ☐ Non-Publicly Available ☐ Sensitive ☐ Non-Sensitive
***See previous concurrence**
To receive a copy of this document, indicate in the concurrence box "C" = Copy without attach/encl "E" = Copy with attach/encl "N" = No copy

OFFICE	RIII		RIII		RIII/EICS	Y	RIII	
NAME	CLipa for BKemker:jb		CLipa		PPelke for GShear*			
DATE	02/12/07		02/12/07		02/07/07			

OFFICIAL RECORD COPY

DISTRIBUTION:
TEB
MMT
RidsNrrDirsIrib
KGO
BJK1
CAA1
LSL
C. Pederson, DRS
DRPIII
DRSIII
PLB1
TXN
ROPreports@nrc.gov
ADAMS (PARS)
SECY
OCA
W. Kane, DEDR
C. Carpenter, OE
D. Solorio, OE
D. Starkey, OE
J. Caldwell, RIII
G. Grant, RIII
L. Chandler, OGC
B. Jones, OGC
J. Dyer, NRR
S. Richards, Chief, IIPB, NRR
M. Tschiltz, Chief, SPSB, NRR
D. Merzke, NRR
J. Stang, NRR
D. Holody, Enforcement Officer, RI
C. Evans, Enforcement Officer, RII
G. Shear, Enforcement Officer, RIII
K. Fuller, Enforcement Officer, RIV
R. Pascarelli, Enforcement Coordinator, NRR
E. Brenner, OPA
H. Bell, OIG
G. Caputo, OI
J. Schlueter, OSTP
P. Pelke, RIII:EICS
V. Mitlyng, RIII:PA
R. Lickus, RIII
J. Lynch, RIII
OEWEB
OEMAIL

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos.:	50-315; 50-316
License Nos.:	DPR-58; DPR-74
Report Nos.:	05000315/2006007; 05000316/2006007
Licensee:	Indiana Michigan Power Company
Facility:	D. C. Cook Nuclear Power Plant, Units 1 and 2
Location:	Bridgman, MI 49106
Dates:	October 1 through December 31, 2006
Inspectors:	B. Kemker, Senior Resident Inspector J. Lennartz, Resident Inspector T. Bilik, Reactor Engineer A. Garmoe, Reactor Engineer M. Holmberg, Reactor Engineer J. Neurauter, Reactor Engineer M. Phalen, Radiation Specialist W. Slawinski, Senior Radiation Specialist R. Walton, Operations Engineer
Approved by:	C. Lipa, Chief Projects Branch 4 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000315/2006-007, IR 05000316/2006-007; 10/01/2006-12/31/2006; D. C. Cook Nuclear Power Plant, Units 1 and 2; Evaluation of Changes, Tests, or Experiments; Inservice Inspection Activities.

The report covered a 13-week period of inspection by the resident inspectors and announced inspections by regional inspectors. One Severity Level IV Non-Cited Violation (NCV) and one Green finding, which had an associated NCV, were identified. 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 3, dated July 2000.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Initiating Events

- Severity Level IV. The inspectors identified a NCV of 10 CFR 50.59 because the licensee failed to perform an adequate safety evaluation review as required by 10 CFR 50.59 for changes made to the Updated Final Safety Analysis Report (UFSAR). In 10 CFR 50.59 Screen No. 2006-0041, "Replace Unit 1 Reactor Vessel Closure Head (1-OME-1)," Revision 0, the licensee evaluated an UFSAR change that removed the emergency load condition specified in UFSAR Tables 2.9-1 and 2.9-2. Within the 10 CFR 50.59 screen, the licensee failed to identify that the proposed activity involved revising or replacing an UFSAR described evaluation methodology that is used in establishing the design bases. As a result, a 10 CFR 50.59 evaluation for the UFSAR change was not performed. The licensee entered the issue into its corrective action program (AR 00803398 and AR 00803828).

  The finding was determined to be more than minor because the inspectors, at the time of the inspection, could not reasonably determine that the UFSAR change, which adversely affected equipment important to safety, would not have ultimately required NRC approval. Because the issue affected the NRC's ability to perform its regulatory function, this finding was evaluated using the traditional enforcement process. However, where possible, the underlying technical issue is evaluated under the Significance Determination Process (SDP) to determine the severity of the violation. In this case, the finding screened as having very low safety significance (Green) using IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," because the inspectors answered "no" to LOCA initiators question 1 under the Initiating Events Cornerstone column of the Phase 1 worksheet. Specifically, since the licensee had evaluated the faulted loading condition as part of the design basis for the replacement reactor vessel closure head, the finding was not a design or qualification deficiency that was confirmed to result in a loss of operability or functionality per Part 9900 Technical Guidance, "Operability Determination Process for

Operability and Functional Assessment." Therefore, the finding would not have likely affected other mitigating systems resulting is a total loss of their safety function. (Section 1R02.1)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a NCV of 10 CFR Part 50.55(a)(g)(4), for failure to accept for continued service, by correction, or evaluation or test, a chemical and volume control system (CVCS) support (2-ACS-R-913) whose examination detected a condition (loose anchor plate nut) unacceptable for continued service in accordance with American Society of Mechanical Engineers (ASME) Section XI Code. The licensee, having instead dispositioned the condition in accordance with operability screening procedure ES-PIPE-1002-QCN, subsequently completed an analysis to confirm that the support was operable with this configuration and entered this issue into their corrective action program.

  This finding was of more than minor significance because the licensee neither corrected this condition (e.g., tighten the loose nut) nor completed an engineering evaluation or test to confirm the ability of this support to carry design loads as required by ASME Code prior to returning it to service. The failure to repair or to perform an engineering evaluation that demonstrated this degraded CVCS support would carry design loads, increased the likelihood of a component failure and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. This finding was of very low safety significance because the licensee subsequently completed an evaluation which confirmed that the support was operable in this configuration. In particular, it did not affect the availability or function of the mitigating system.

  This has a cross-cutting aspect in the area of human performance because the licensee’s screening procedure was not adequate and directed the licensee to perform actions contrary to the requirements of the ASME Code. (Section 1R08.5)

B. Licensee Identified Violations

One violation of very low safety significance, which was identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. The violation and the corrective action tracking number are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 1 was shut down and defueled at the beginning of the inspection period for the Cycle 21 refueling outage (U1C21). The licensee performed a reactor startup on November 10, 2006, and synchronized the unit to the grid on November 13, 2006, upon completion of a 58 day refueling outage. Unit 1 reached full power on November 20, 2006, following extensive testing of the new low pressure turbine and main generator digital control system. The unit operated at or near full power for the remainder of the inspection period.

Unit 2 was operated at or near full power during the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

1R01 Adverse Weather Protection (71111.01)

.1 Cold Weather Protection

a. Inspection Scope

The inspectors completed one inspection sample regarding adverse weather protection by reviewing and assessing activities conducted for the onset of cold weather.

The inspectors reviewed documentation to verify 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; and, toured the lake screenhouse to verify that winterization heaters were in service.

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

b. Findings

No findings of significance were identified.

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

.1 Reactor Vessel Closure Head (RVCH) Replacement (71007)

a. Inspection Scope

From September 5 through September 8, 2006, from September 25 through September 29, 2006, and from October 10 through October 13, 2006, the inspectors reviewed the licensee's screening documents for the design changes associated with the Unit 1 RVCH replacement to determine, for each change, whether the requirements of 10 CFR 50.59 had been appropriately applied. Specifically, the inspector reviewed 1-MOD-55520, "Replace Unit 1 Reactor Vessel Closure Head (1-OME-1)," which included a review of the function of each changed component, the change description and 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 one spare thermocouple penetration;
- 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 columns;
- new dedicated reactor vessel level instrumentation system (RVLIS) penetration nozzle;
- new dedicated reactor vessel head vent (RVHV) penetration nozzle; 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 the Unit 1 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 1-MOD-55003, "Install Unit 1 Replacement RVCH and Modify the Existing Unit 1 Service Structure (1-OME-1)," which included a review of the function of each changed component, the change description, methods of analysis, and scope of the 10 CFR 50.59 screen 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 RVHV cables;
- replacement RVHV resistance temperature detector cables;
- revised RVLIS and RVHV piping and valve layout;
- new handrail assembly on existing CRDM platform; and
- additional fall protection attachment points.

The inspector also reviewed one 10 CFR 50.59 screening and one 10 CFR 50.59 evaluation associated with removing the existing RVCH from containment and moving the new RVCH through the auxiliary building into containment.

The inspector used, in part, Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Implementation," 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

Failure to Perform 10 CFR 50.59 Evaluation to Remove Design Basis Requirement from UFSAR

Introduction: On September 30, 2006, the inspectors identified a Severity Level IV Non-Cited Violation (NCV) of very low safety significance for failing to perform a safety evaluation in accordance with 10 CFR 50.59. The 10 CFR 50.59 Screening No. 2006-0041-00, "Replace Unit 1 Reactor Vessel Closure Head (1-OME-1)," evaluated an Updated Final Safety Analysis Report (UFSAR) change that removed the emergency load combination specified in UFSAR Tables 2.9-1 and 2.9-2. Within the 10 CFR 50.59 screen, the licensee failed to identify that the proposed activity involved revising or replacing an UFSAR described evaluation methodology that is used in establishing the design bases. As a result, a 10 CFR 50.59 evaluation for the UFSAR change was not performed.

Description: In 10 CFR 50.59 Screen No. 2006-0041, the licensee evaluated a change to the UFSAR that removed the emergency load condition specified in UFSAR Tables 2.9-1 and 2.9-2. Specifically, UFSAR Table 2.9-1 defines the emergency condition as the combination of the normal condition and the design basis earthquake (DBE) load, and UFSAR Table 2.9-2 specifies the stress intensity acceptance limits for the emergency loading condition.

The inspectors identified that the replacement RVCH design specification and design reports did not evaluate the emergency loading condition. The DBE load case was only evaluated in the faulted loading condition, the combination of the normal condition and the DBE load and the pipe rupture load. The inspectors further identified that the UFSAR Table 2.9-2 stress intensity acceptance limits for the emergency loading condition were lower than the stress intensity acceptance limits for the faulted loading condition.

Using NEI 96-07 Section 4.3.8.1 for guidance related to changing a method of evaluation, the inspectors determined that evaluating the DBE load case only as part of the faulted load condition could result in gaining margin with respect to the emergency loading condition. Therefore, removal of the emergency condition from UFSAR Tables 2.9-1 and 2.9-2 is considered to be a non-conservative change and thus a departure from a method of evaluation for purposes of 10 CFR 50.59 that would require NRC approval.

Analysis: The inspectors determined that this issue was a performance deficiency, since the licensee permanently changed the facility as described in the UFSAR without providing the necessary justification under 10 CFR 50.59 for the activity that created a possibility for a malfunction of a system, structure, or component important to safety with a different result than previously evaluated in the UFSAR. The finding was determined to be more than minor because the inspectors, at the time of the inspection, could not reasonably determine that the UFSAR change, which adversely affected equipment important to safety, would not have ultimately required NRC 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, where possible, the underlying technical issue is evaluated under the SDP to determine the severity of the violation. In this case, the finding screened as having very low significance (Green) using IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," because the inspectors answered "no" to LOCA Initiator question 1 under the Initiating Events Cornerstone column of the Phase 1 worksheet. Specifically, since the licensee had evaluated the faulted loading condition as part of the design basis for the replacement RVCH, the finding was not a design or qualification deficiency that was confirmed to result in a loss of operability or functionality per Part 9900, Technical Guidance, "Operability Determination Process for Operability and Functional Assessment." Therefore, the finding would not have likely affected other mitigating systems resulting in a total loss of their safety function. Based upon this Phase 1 screening, the inspectors concluded that the issue was of very low safety significance (Green). In accordance with the Enforcement Policy, the finding was therefore classified as a Severity Level IV violation.

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

Contrary to the above, the licensee failed to perform an adequate safety evaluation review as required by 10 CFR 50.59 for the change that removed the emergency load condition specified in UFSAR Tables 2.9-1 and 2.9-2. Within the 10 CFR 50.59 Screen No. 2006-0041, the licensee failed to identify that the proposed activity involved revising or replacing a UFSAR described evaluation methodology that is used in establishing the design bases. As a result, a 10 CFR 50.59 evaluation for the UFSAR change was not performed. In accordance with the Enforcement Policy, this violation of the requirements of 10 CFR 50.59 was classified as a Severity Level IV Violation because the underlying technical issue was of very low safety significance. Because this violation was not willful or not repetitive, and was entered into the licensee's corrective action program (AR Nos. 00803398 and 00803828), it is considered a NCV consistent with VI.A.1 of the NRC Enforcement Policy. (NCV 05000315/2006007-01)

1R04 Equipment Alignment (71111.04)

.1 Partial System Walkdowns

a. Inspection Scope

The inspectors completed three partial equipment walkdown inspection samples for the following risk significant systems:

- Unit 1 Makeup Sources to the Spent Fuel Pool;
- Unit 2 East Containment Spray System Train; and
- Unit 2 South Safety Injection 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, 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 were 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.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours

a. Inspection Scope

The inspectors completed six quarterly fire protection inspection samples by performing walkdowns in the following plant areas:

- Unit 1 Reactor Head Enclosure 567' Elevation (Zone 103);
- Unit 2 Refueling Water, Condensate, Primary Water Tank Area Pipe Tunnel 593' Elevation (Zone 117);
- Unit 1 Refueling Water, Condensate, Primary Water Tank Area Pipe Tunnel 593' Elevation (Zone 116);
- Unit 1 Containment Regenerative Heat Exchanger Room (Zone 118);
- Unit 1 Containment Spray Pump Rooms (Zones 1A and 1B); and
- Unit 2 Containment Spray Pump Rooms (Zones 1E and 1F).

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.

b. Findings

No findings of significance were identified.

.2 Annual Fire Drill Observation

a. Inspection Scope

The inspectors completed one annual inspection sample by observing an unannounced fire drill that was conducted on November 30, 2006, in the Auxiliary Building 609' elevation snubber room.

The inspectors assessed the fire brigade's readiness to respond to and mitigate fires by verifying the following:

- 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 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;
- fire hoses were laid out without flow restrictions and were of sufficient length to reach the fire area;
- communications between fire brigade members and between the fire brigade leader and operations personnel were clear, efficient and effective; and
- fire brigade members entered the fire area in a controlled manner utilizing the two-man rule.

The inspectors also verified that the fire scenario was appropriately simulated, that the licensee’s pre-planned drill scenario was followed and that 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 condition reports 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.

b. <u>Findings</u>

No findings of significance were identified.

1R08 <u>Inservice Inspection (ISI) Activities</u> (71111.08)

.1 <u>Piping Systems ISI</u>

a. <u>Inspection Scope</u>

From September 25, 2006, through October 11, 2006, the inspectors conducted a review of the implementation of the licensee's ISI program for monitoring degradation of the reactor coolant system (RCS) boundary and the risk significant piping system boundaries. The inspectors selected the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI required examinations and Code components in order of risk priority as identified in Section 71111.08-03 of IP 71111.08, "Inservice Inspection Activities," based upon the ISI activities available for review during the onsite inspection period.

The following nondestructive examination (NDE) activities were observed by the inspectors to evaluate compliance with the ASME Code Section XI and Section V requirements and to verify that indications and defects (if present) were dispositioned in accordance with the ASME Code Section XI requirements. Specifically, the inspector observed the following examinations:

- Ultrasonic Examination (UT) of the boric acid injection tank upper head-to-shell weld 1-BIT-B; and
- Visual Examination of main steam pipe support snubber 1MS08007S1 and component cooling system pipe support snubber 1CC24013S.

The inspectors reviewed examinations completed during the previous outage with relevant/recordable conditions/indications that were accepted for continued service to observe that the licensee's acceptances were in accordance with the Section XI of the ASME Code. Specifically, the inspectors reviewed the following records:

- UT examination of pressurizer (1-PRZ-26) vessel support. Several recordable indications were evaluated against ASME Section XI, 1989 Edition, no Addenda Table IWB-3510-2, and were found to be acceptable; and
- UT examination of pressurizer (1-PRZ-15) upper shell to upper head. Several recordable indications were evaluated against ASME Section XI, 1989 Edition, no Addenda Table IWB-3510-1, and were found to be acceptable.

The inspectors reviewed a pressure boundary weld for a Class 2 system which was completed since the beginning of the previous refueling outage to determine if the welding acceptance and preservice examinations (e.g., visual, dye penetrant, and weld procedure qualification tensile tests) were performed in accordance with ASME Code Sections III, V, IX, and XI requirements. Specifically, the inspectors reviewed a weld associated with the following work activity:

- Repair (welding; OW-1, 1-CS-34) of ISI Class 2, 6" suction piping for centrifugal charging pump 1-PP-50W for the CVCS.

The reviews discussed above counted as one inspection sample.

b. Findings

No findings of significance were identified.

.2 Pressurized Water Reactor Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

The licensee replaced the reactor pressure vessel head during this refueling outage and hence were not required to complete vessel upper head examinations. Therefore, this section of the baseline inspection procedure was not available for inspection and did not count as a completed inspection sample. (See Section 4OA5 of this report and IR 2006-006 for documentation of related NRC Inspection Activities of the vessel head replacement per IP 71007.)

b. Findings

No findings of significance were identified.

.3 Boric Acid Corrosion Control ISI

a. Inspection Scope

Following shutdown, the inspectors reviewed a sample of boric acid corrosion control walkdown visual examination activities through direct observation. This walkdown was completed with Unit 1 in Mode 3 and included the lower containment building inner volume and annulus. The inspectors verified that the visual inspections emphasized locations where boric acid leaks can cause degradation of safety significant components.

The inspectors reviewed the following boric acid leak corrective action to confirm that it was consistent with the requirements of the ASME code and 10 CFR Part 50, Appendix B, Criterion XVI. The inspectors also reviewed the engineering evaluation performed for this same corrective action document. The evaluation was verified, as applicable, to ensure that ASME Code wall thickness requirements were maintained:

- Condition Report 06063012 (Unit 1), "1-ICM-311, East RHR to RC Loops Isolation Valve Leakage."

The inspectors also reviewed a number of boric acid leak corrective actions to confirm that they were consistent with the requirements of the ASME Code and 10 CFR Part 50, Appendix B, Criterion XVI. The documents reviewed during this inspection are listed in the Attachment to this report.

The reviews discussed above counted as one inspection sample.

b. Findings

No findings of significance were identified.

.4 Steam Generator (SG) Tube ISI

a. Inspection Scope

From October 6, 2006 through October 12, 2006, the inspectors performed an on-site review of SG tube examination activities conducted pursuant to Technical Specification (TS) and the ASME Code Section XI requirements. 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:

- in-situ SG tube pressure testing screening criteria and the methodologies used to derive these criteria 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 outage Operational Assessment predictions;
- the SG tube ET examination scope and expansion criteria were sufficient to identify tube 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 EPRI 1003138, "Pressurized Water Reactor Steam Generator Examination Guidelines: Revision 6;"
- the SG tube ET examination scope included tube areas which represent ET challenges such as the tubesheet regions, expansion transitions, and support plates;
- the licensee identified new tube degradation mechanisms;
- the licensee implemented repair methods which were consistent with the repair processes allowed in the plant TS requirements;
- the licensee 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, "Pressurized Water Reactor Steam Generator Examination Guidelines," Revision 6; and
- the licensee identified deviations from ET data acquisition or analysis procedures.

The inspectors performed a review of SG ISI 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 SG related problems;
- the licensee had established an appropriate threshold for identifying issues;

- the licensee had evaluated industry generic issues related to SG tube integrity; and
- the licensee implemented appropriate corrective actions.

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

The reviews as discussed above counted as one inspection sample.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of ISI 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 related problems;
- the licensee had established an appropriate threshold for identifying issues;
- the licensee had evaluated industry generic issues related to ISI and pressure boundary integrity; and
- the licensee implemented appropriate corrective actions.

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. In addition, the inspectors verified that the licensee correctly assessed operating experience for applicability to the ISI group.

b. Findings

.1 Failure to Verify Adequacy of Degraded CVCS Support

Introduction: The inspectors identified a Green NCV of 10 CFR Part 50.55(a)(g)(4), for failure to accept for continued service, by correction, or evaluation or test, a CVCS support (2-ACS-R-913) whose examination detected a condition (loose anchor plate nut) unacceptable for continued service in accordance with ASME Section XI Code.

Description: On January 16, 2006, during an ASME Code required VT-3 visual examination of CVCS support 2-ACS-R-913, the licensee identified a loosened support item, which is a Code rejectable condition. Specifically, one of the two support baseplate anchor bolt nuts was backed off (loose). ASME Section XI states that component support conditions which are unacceptable for continued service shall include loosened support items. The licensee, as allowed by operability screening procedure ES-PIPE-1002-QCN, neither corrected this condition (e.g., tighten the loose nut) nor completed an engineering evaluation or test to confirm the ability of this support to carry design loads as required by ASME Code prior to returning it to service. Specifically, the procedure allowed the licensee to disposition the problem without sufficient justification. The inspectors' questions on how this Code rejectable condition was accepted, prompted the licensee's staff to enter this issue into the corrective action system (AR 00803738) and to complete an evaluation (incorporated into AR 00803738) to confirm that the support was operable in this configuration.

Analysis: The inspectors determined that the failure to correct or evaluate or accept the nonconformance for support 2-ACS-R-913 in accordance with ASME Code was a performance deficiency that warranted a significance evaluation. This finding was of more than minor significance in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," because the finding was associated with the Mitigating System cornerstone attribute of "Equipment Performance" and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The failure to correct this condition, or to perform an engineering evaluation to confirm that this degraded CVCS support would carry design loads, increased the likelihood of a component failure that would affect CVCS operability. Because the inspectors answered "No" to each of the phase 1 screening questions for "Determining the Significance of Reactor Inspection Findings for At-Power Situations," and specifically because the licensee's staff completed an evaluation (incorporated into AR00803738) to confirm that the support was operable in this configuration, this finding was of very low safety significance. This finding has a cross-cutting aspect in the area of human performance, because the licensee's screening procedure was not adequate, and directed the licensee to perform actions contrary to the requirements of the ASME Code.

Enforcement: The inspectors identified a NCV of 10 CFR Part 50.55(a)(g)(4), "Inservice Inspection Requirements," having a very low safety significance (Green), related to the acceptance of a component support for continued service without being dispositioned in accordance with the ASME Code.

Title 10 CFR 50.55a(g)4 requires, in part, that throughout the service life of a pressurized water-cooled nuclear power facility, components must meet the requirements set forth in the ASME Code Section XI.

The ASME Code 1989 Edition, no Addenda, Section XI, Article IWF-3410, "Acceptance Standards - Component Support Structural Integrity," states in part that component support conditions which are unacceptable for continued service shall include loosened support items.

ASME Code Section XI, IWB-3122, requires that component supports which do not meet the acceptance standards of IWF-3410 shall be corrected in accordance with the provisions of IWF-3122.2 (acceptance by correction), or IWF-3122.3 (acceptance by evaluation or test) to permit acceptance for continued service.

Contrary to the above, the licensee failed to accept an unacceptable CVCS component support condition for support 2-ACS-R-913 by correction, evaluation, or test prior to accepting the component for continued service. Because of the very low safety significance of this finding, and because the licensee subsequently completed an evaluation to confirm that the support was operable with this configuration and entered this issue into their corrective action program as AR 00803738, it is being treated as a NCV consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000315/2006007-02).

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review

a. Inspection Scope

The inspectors completed one quarterly inspection sample of licensed operator requalification training by observing a crew of licensed operators during simulator training on November 21, 2006. 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.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Resident Inspector Quarterly Review

a. Inspection Scope

The inspectors completed two quarterly maintenance effectiveness inspection samples by evaluating the licensee's handling of selected degraded performance issues involving the following risk-significant structures, systems, and components (SSCs):

- Non-essential Service Water Supply to Unit 1 Upper Containment Vent Unit #3 Train 'A' Containment Isolation Valve 1-WCR-929; and
- Unit 1 and Unit 2 Ice Condensers.

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the SSCs. Specifically, the inspectors independently verified the licensee's handling of SSC performance or condition problems in terms of:

- appropriate work practices;
- identifying and addressing common cause failures;

- scoping of SSCs in accordance with 10 CFR 50.65(b);
- characterizing SSC reliability issues;
- tracking SSC unavailability;
- trending key parameters (condition monitoring);
- 10 CFR 50.65(a)(1) or (a)(2) classification and reclassification; and
- appropriateness of performance criteria for SSCs/functions classified (a)(2) and/or appropriateness and adequacy of goals and corrective actions for SSCs/functions classified (a)(1).

In addition, the inspectors verified that problems associated with the effectiveness of plant maintenance 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.

b. Findings

1. Unit 1 and Unit 2 Ice Condensers

The inspectors reviewed a sample of equipment performance issues associated with the Unit 1 and Unit 2 ice condensers and found one example where a Maintenance Rule Evaluation (MRE) was not completed when the performance criteria for ice bed flow blockage was exceeded. The inspectors reviewed action request (AR) 00124195 from the Unit 2 refueling outage in March 2006, which was written to evaluate whether the TS surveillance requirement limit of 15 percent ice blockage in a particular bay was exceeded. While the licensee's analysis found the actual blockage to be less than 15 percent, the flow blockage (11 percent actual flow blockage) exceeded the Maintenance Rule performance criteria of 10 percent and no MRE was completed. In response to the inspectors' questions, the licensee wrote AR 06361030 to perform an MRE for this example.

Based on the identification that no MRE was completed for this one example where the performance criteria was exceeded, the inspectors expanded the scope of their review to determine if there were additional examples of ice blockage that were not properly evaluated. The inspectors interviewed the system engineer to determine whether there were historical issues with ice bed flow blockage in the ice condensers or other ice condenser performance criteria issues (e.g., lower inlet door test failures), where the Maintenance Rule performance criteria may have been exceeded while the TS surveillance requirement limit was not exceeded and MREs were not completed. Based on this discussion, the inspectors requested additional documents to review. This issue is considered to be an Unresolved Item (URI 05000315/316/2006007-03) pending additional review.

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

a. Inspection Scope

The inspectors completed seven inspection samples regarding maintenance risk assessments and emergent work evaluations for the following maintenance activities:

- Unit 1 and Unit 2 Significant Switchyard Maintenance Activities Concurrent with Unit 1 CD Emergency Diesel Generator (EDG) Maintenance and High Wind Condition;
- Unit 1 Control Room Instrumentation Distribution Inverter #1 Emergent Maintenance;
- 345 kilo-Volt Switchyard Maintenance Activities and Unit 1 AB EDG Surveillance Testing;
- Unit 1 Control Air Compressor Maintenance, Unit 1 CD EDG and West Motor Driven Auxiliary Water Pump Surveillance Testing, and Unit 2 East Component Cooling Water Pump and Heat Exchanger Preventive Maintenance;
- Unit 1 RCS Drain to Mid-loop During the Refueling Outage;
- Unit 2 Instrument Isolation Valve 2-NPS-121-II Emergent Maintenance; and
- Unit 2 AB EDG Maintenance.

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.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors completed ten inspection samples associated with operability evaluations by reviewing the following action requests (ARs):

- AR 06259019, "Overboration of RCS [Reactor Coolant System];"
- AR 06271003, "Investigate Apparent Fatigue Failure Exhaust Manifold Bolt;"
- AR 06275019, "1-OHP-4021-082-001 Allows Backfeed in Modes 5 and 6;"
- AR 00802716, "Valves 1-SI-158-L1 and 1-SI-158-L4 Had a Combined Leak Rate of 21.029 GPM [Gallons-per-Minute];"
- AR 06276085, "Missing Bolts on Divider Barrier Seal;"

- AR 06284020, "The Minimum Wall Thickness on Main Steam Line Is Higher Than Pipe Initial Thickness;"
- AR 06285023, "Boron Injection Tank Telltale Drains;"
- AR 00802749, "Flood-up Tube #9 at 1-CEP-218 Has a Thru-wall Hole As a Result of an Arc Strike;"
- AR 06352129, "1-ICM-311 Is Over-thrusting Open and Closed;" and
- AR 00801035, "Accumulator Water Temperatures Exceeding LBLOCA [Large Break Loss-of-Coolant Accident] Analysis Assumption."

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 and appropriately returned the affected equipment to an operable status.

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.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors completed one annual inspection sample by reviewing the engineering analyses, modification documents and design change information associated with the following permanent plant modification:

1-MOD-65754, "Containment Sump Debris Accumulation Modifications"

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 these 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 Unit 1 refueling outage, the emergency core cooling system (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 containment; 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 inch diameter crossover pipe between the recirculation sump and the lower containment sump.

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 SSCs 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.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors completed ten inspection samples pertaining to post maintenance testing by assessing testing activities that were conducted on the following plant equipment:

- Unit 1 Steam Generator Power Operated Relief Valve 1-MRV-223;
- Unit 1 East Residual Heat Removal System to RCS Loops 1 and 4 Isolation Valve 1-ICM-311;
- Unit 1 RCS Resistance Temperature Detector Bypass Manifold Removal and Reorientation of Vessel Level Instrument Lines;
- Unit 1 East Charging Pump;
- Unit 1 Main Generator Digital Controls Upgrade;
- Unit 1 Control Rod Drop Measurements;
- Unit 1 CD EDG Voltage Regulator;

- Unit 1 CD EDG 2R Fuel Injector Pump;
- Unit 1 AB EDG Air Start Valve XRV-222; and
- Unit 1 AB EDG Digital Reference Unit.

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.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

.1 Unit 1 Refueling Outage (U1C21)

a. Inspection Scope

The inspectors evaluated the licensee's conduct of U1C21 activities to assess the licensee's control of plant configuration and management of shutdown risk. This represented one inspection sample. 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 which 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 loss-of-coolant accident 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 the issues that the licensee entered into the corrective action program to verify that identified problems were being entered into the program with the appropriate characterization and significance. The inspectors also reviewed the licensee's corrective actions for refueling outage issues documented in selected action requests.

b. Findings

.1 Review of Unit 1 RCS Boundary Leakage Requirements

The inspectors reviewed the Unit 1 RCS boundary leakage requirements from entry into Mode 4 on November 5, 2006, until after the first RCS leakrate calculation was performed with the unit at full power on November 21st, and noted that the licensee had not performed an RCS inventory balance for Unit 1 since November 9th. The inspectors asked the licensee how it complied with TS Surveillance Requirement 3.4.13.1, which required verification that RCS operational leakage is within limits by performance of an RCS water inventory balance every 72 hours with the unit in Modes 1 through 4. There is a note in the TSs that states that the leakrate calculation is not required to be performed until 12 hours after establishment of steady state operation. Steady state operation is defined in the TS Bases as steady RCS pressure, temperature and power level. The inspectors reviewed Unit 1 plant power history since November 9th and noted that there were several periods of time when it appeared that the plant was stable, at steady state conditions, during the power ascent. In response to the inspectors' questions, the licensee wrote an action request to evaluate the processes and procedures for ensuring that the RCS boundary leakage requirements are met during plant startup. This issue is considered an Unresolved Item (URI 05000315/2006007-04) pending the inspectors' review of compliance with the surveillance test requirement.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors completed five inspection samples regarding surveillance testing by reviewing the following activities. This included one Inservice Testing (IST) sample, one RCS leakrate detection sample, one ice condenser system sample, and one local leak rate test (LLRT) sample.

- 1-EHP-4030-134-001, "Unit 1 Primary Containment Leak Rate Running Total" (LLRT sample);
- 12-EHP-5030-001-008, "Recirculation Loop Total Leak Rate" (RCS Leakrate sample);
- 12-MHP-4030-010-003, "Ice Condenser Lower Inlet Door Surveillance, 40 Degree Force Test" (Ice Condenser sample);
- 2-OHP-4030-216-020W, "West Component Cooling Water Loop Surveillance Test" (IST sample); and
- 1-OHP-4030-132-217A, "EDG 1CD Load Sequencing and ESF [Engineered Safety Features] Testing."

The inspectors observed 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 being entered into the licensee's corrective action program with the appropriate characterization and significance.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors completed two inspection samples regarding emergency preparedness drill evaluations by observing simulator training evolutions for licensed operators on November 29 and December 12, 2006, which required emergency plan implementation. Licensee emergency preparedness personnel had pre-designated that the opportunities for the Shift Manager to classify the event and make required notifications would be evaluated and included in performance indicator data regarding drill and exercise performance.

The inspectors verified that the Shift Manager classified the emergency condition and completed the required notifications to state and local police authorities in an accurate and timely manner as required by the Emergency Plan implementing procedures. The inspectors also observed the post-training critique to verify that licensee evaluators appropriately identified performance deficiencies.

b. Findings

No findings of significance identified.

2. RADIATION SAFETY

2OS1 Access Control to Radiologically Significant Areas and Reactor Vessel Head Replacement Inspection (71121.01 and 71007)

.1 Review of Licensee Performance Indicators for the Occupational Exposure Cornerstone

a. Inspection Scope

The inspectors reviewed Licensee Event Reports (LERs), corrective action documents, electronic dosimetry transaction data for radiologically controlled area egress, internal dose assessment summary information, and data reported on the NRC's web site relative to the licensee's occupational exposure control performance indicator to determine whether or not the conditions surrounding any actual or potential performance indicator (PI) occurrences had been evaluated, and identified problems had been entered into the corrective action program for resolution. Also, performance indicator data collection and analysis methods were evaluated for adequacy by the inspectors as described in Section 4OA1.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.2 Plant Walkdowns/Boundary Verifications and Radiation Work Permit Reviews

a. Inspection Scope

The inspectors identified the following five radiologically significant work areas within high and locked high radiation areas of the plant and other potentially exposure significant work activities, and selectively reviewed radiation work permit (RWP) packages and radiation surveys for these areas. The inspectors evaluated the radiological controls to determine if these controls including postings and access control barriers were adequate:

- Reactor Vessel Head Replacement in Unit 1 Upper Containment;
- Reactor Coolant Pump Motor Replacement in Unit 1 Lower Containment;
- Resistance Temperature Detector Bypass Removal in Unit 1 Lower Containment;
- Pressurizer Weld Overlay in Unit 1 Lower Containment; and
- SG Platform Activities in Unit 1 Lower Containment.

The inspectors reviewed the RWPs which governed activities in these radiologically significant areas to identify the work control instructions and control barriers that had been specified. For these activities, electronic dosimeter alarm set points for both integrated dose and dose rate were evaluated for conformity with survey indications and plant procedures. Workers were interviewed to determine if they were aware of the radiological conditions in their work areas and the actions required when their electronic dosimetry malfunctioned or alarmed.

The inspectors walked down and surveyed numerous high and locked high radiation area boundaries in the Auxiliary and Unit-1 Containment Buildings to determine if the prescribed radiological access controls were in place, that licensee postings were complete and accurate, and that physical barricades/barriers were adequate. During the walkdowns, the inspectors challenged access control boundaries to determine if high radiation area and locked high radiation area (LHRA) access was controlled in compliance with the licensee's procedures, Technical Specifications, the requirements of 10 CFR 20.1601, and were consistent with Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants."

The inspectors reviewed the licensee's practices and programmatic controls which prohibited the temporary storage of highly activated and/or contaminated materials (non-fuel) within the spent fuel pool attached to cables/lanyards and consequently more readily removable from the pool. Radiation protection staff were interviewed and a walkdown of the refuel floor was performed to verify the licensee's practices.

The inspectors reviewed the RWPs for those work activities with the potential to generate airborne radioactivity to determine whether adequate engineering controls (e.g., use of ventilation systems, surface wetting, vacuuming, etc.) were provided to reduce the potential for worker internal exposure. Work activities with the potential for airborne transuranic radioactivity such as work in the SG bowls and work associated with the reactor vessel head replacement were evaluated to determine if the licensee had performed surveys to identify whether alpha emitters were present.

The inspectors reviewed the licensee's procedures and its methods for the assessment of internal dose as required by 10 CFR 20.1204, to determine if methodologies were technically accurate and would include the impact of hard to detect radionuclides such as pure beta or alpha emitters, if applicable. No worker intakes that resulted in a committed effective dose equivalent in excess of 50 millirem occurred during the period reviewed by the inspectors (September 2005 - September 2006). However, internal dose assessments which resulted in exposures less than 50 millirem committed effective dose equivalent were reviewed by the inspectors for adequacy.

These reviews represented five inspection samples.

b. Findings

No findings of significance were identified.

.3 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed LERs and Special Reports, as applicable, and the AR database along with individual ARs related to the radiological access and exposure control programs to determine if identified problems were entered into the corrective action program for resolution. In particular, the inspectors reviewed radiological issues which occurred over approximately the 6 month period that preceded the inspection including the review of any high radiation area radiological incidents (non-PI occurrences identified by the licensee in high and locked high radiation areas) to determine if follow-up activities were conducted in an effective and timely manner commensurate with their importance to safety and risk based on the following:

- 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; and
- Identification and implementation of corrective actions.

The inspectors evaluated the licensee's process for problem identification, characterization, and prioritization, and determined if problems were entered into the corrective action program and were being resolved in a timely manner. For potential repetitive deficiencies or possible trends, the inspectors determined if the licensee's self-assessment activities were capable of identifying and addressing these deficiencies, if applicable.

The inspectors reviewed the licensee's documentation for all potential PI events occurring since the NRC's last review of these areas in September 2005, to determine if any of these events involved dose rates greater than 25 Rem/hour at 30 centimeters or greater than 500 Rem/hour at 1 meter or involved unintended exposures greater than 100 millirem total effective dose equivalent (or greater than 5 Rem shallow dose equivalent or greater than 1.5 Rem lens dose equivalent). None were identified.

These reviews represented four inspection samples. Specifically, the samples pertained to the problem identification and resolution program for radiological incidents, a review of the licensee's ability to identify and address repetitive deficiencies, and a review of those radiological incidents and potential PI occurrences of greatest radiological risk.

b. Findings

No findings of significance were identified.

.4 <u>Job-In-Progress Reviews and Review of Work Practices in Radiologically Significant Areas</u>

a. <u>Inspection Scope</u>

The inspectors observed aspects of the following five ongoing jobs that were being performed in radiologically significant areas to evaluate work activities that presented the greatest radiological risk to workers. This review was conducted in conjunction with Inspection Procedure 71121.02, "ALARA Planning and Controls," and is documented further in Section 2OS2.3 of this report.

- Reactor Vessel Head Replacement;
- Reactor Coolant Pump Motor Replacement;
- Resistance Temperature Detector Bypass Removal;
- Pressurizer Weld Overlay; and
- SG Manway Removal and Platform Activities.

The inspectors reviewed the radiological job requirements for these activities, including the RWP requirements, and attended an ALARA pre-job briefing for SG diaphragm removal. Radiation survey information to support these work activities was reviewed and the radiological job requirements and the access control provisions were assessed for conformity with TSs and with the licensee’s procedures. During job observations, the inspectors determined if radiological controls and radiation protection job coverage were adequate including audio/video remote job coverage surveillance.

The inspectors reviewed the licensee’s procedures and discussed with RP staff its practices for access into high and potentially very high radiation areas, and for areas with the potential for changing radiological conditions such as the Reactor Pit Hatch and the Aux Building 573' CVCS Middle Hold-Up tank Room. The inspectors evaluated the adequacy of the radiological controls and the radiological hazards assessment associated with such entries.

The inspectors also reviewed the licensee’s procedure and practices associated with dosimetry placement (both whole body and extremity dosimetry) and with the use of multiple whole body dosimetry for work in high radiation areas having significant dose gradients for compliance with the requirements of 10 CFR 20.1201(c) and applicable industry guidelines. Work in areas where dose rate gradients were subject to significant variation, including pressurizer weld overlay activities and SG diaphragm removal, were reviewed to evaluate the licensee’s practices for dosimetry placement.

These reviews represented three inspection samples.

b. <u>Findings</u>

No findings of significance were identified.

.5 High Risk Significant, LHRA and Very High Radiation Area (VHRA) Access Controls

a. Inspection Scope

The inspectors reviewed the licensee's procedures and evaluated RP practices for the control of access to radiologically significant areas (high, locked high, and very high radiation areas). The inspectors assessed compliance with the licensee's Technical Specifications, procedures, the requirements of 10 CFR Part 20, and the guidance contained in Regulatory Guide 8.38. In particular, the inspectors evaluated the RP staff's control of keys to LHRAs and VHRAs, the use of access control guards during work in these areas, and methods and practices for independently verifying proper securing of access doors upon area egress. The inspectors selectively reviewed key issuance/return and door lock verification records and key accountability logs for selected periods in 2006 to determine the adequacy of accountability practices and documentation. The inspectors also evaluated the RP staff's practices for radiation protection manager and station management approval for access into VHRAs and for the use of flashing lights in lieu of locking areas to verify compliance with procedure requirements and those of 10 CFR 20.1602.

The inspectors discussed with RP staff the controls that were in place for areas that had the potential to become high radiation areas during normal plant operation and plant unit shutdowns, to verify that appropriate radiological controls were embedded into plant procedures and outage schedule activities. Additionally, the inspectors determined that selected activities that required communication before-hand with the RP group, were embedded into plant scheduling process so as to allow corresponding timely actions to properly post and control the radiation hazards.

The inspectors conducted plant walkdowns to verify the posting and locking of entrances to numerous HRAs and LHRAs throughout the Auxiliary and Unit 1 Containment Buildings, including the barriers and flashing lights used to control access to the LHRA present on the underside of the old reactor vessel head.

These reviews represented three inspection samples.

b. Findings

No findings of significance were identified.

.6 Radiation Worker Performance

a. Inspection Scope

During job performance observations, the inspectors evaluated radiation worker performance with respect to stated radiation protection work requirements and determined whether workers were aware of the radiological conditions, the RWP controls and limits in place, and that their performance had accounted for the level of radiological hazards present.

The inspectors also reviewed radiological problem reports generated primarily in 2006 (year to date) which found that the cause of the event was due to radiation worker errors to determine if there was an observable pattern traceable to a similar cause, and to determine if this matched the corrective action approach taken by the licensee to resolve the identified problems.

These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

.7 Radiation Protection Technician (RPT) Staffing, Training and Proficiency

a. Inspection Scope

During job observations and plant walkdowns, the inspectors evaluated radiation protection staff performance with respect to radiation protection work requirements, conformance with procedures and those requirements specified in the RWP, and assessed proficiency with respect to radiation protection requirements, station procedures and health physics practices.

The inspectors reviewed selected radiological problem reports generated primarily in 2006 (year to date) to determine the extent of any specific problems or trends that may have been caused by deficiencies with RPT work control, and to determine if the corrective action approach taken by the licensee to resolve the reported problems, if applicable, was adequate.

Additionally, the inspectors reviewed RPT contractor staffing to support the reactor vessel head replacement project and the training provided to these supplemental RP staff. The staffing and training was reviewed to determine if the licensee supported the head replacement project with sufficiently qualified and trained RP staff.

These reviews represented two inspection samples for Inspection Procedure 71121.01 and one inspection sample for Inspection Procedure 71007.

b. Findings

No findings of significance were identified.

2OS2 As-Low-As-Reasonably-Achievable (ALARA) Planning and Controls and Reactor Vessel Head Replacement Inspection (71121.02 & 71007)

.1 Radiological Work Planning

a. Inspection Scope

The inspectors reviewed the licensee's list of refueling outage work ranked by estimated exposure and reviewed the following radiologically significant work activities:

- Reactor Coolant Pump Motor Replacement and Seal Activities (RWP 1151/1152);
- Pressurizer Weld Overlay Activities (RWP 1190);
- Reactor Head Replacement (RWP 1107); and
- SG Manway/Diaphragm Activities and SG Platform Activities (RWP 1147 and 1148).

For each of the activities listed above, the inspectors reviewed the RWP and the ALARA Plan, and the associated total effective dose equivalent (TEDE) ALARA evaluations (i.e., respirator evaluations), as applicable. The reviews were performed in order to verify that the licensee had established radiological engineering controls and dose mitigation criteria that were based on sound radiation protection principles in order to achieve occupational exposures that were ALARA.

The inspectors compared the exposure results achieved for the initial 17-days of the scheduled 45-day outage including the dose rate reductions and person-rem expended with the doses projected in the licensee's ALARA planning for the above listed work activities. Reasons for inconsistencies between intended (projected) and actual work activity doses as well as time/labor differences were examined to determine if the activities were planned reasonably well and to ensure the licensee was cognizant of and evaluated any work planning deficiencies.

Work-In-Progress ALARA Reports were reviewed by the inspectors for those above listed outage jobs that approached their respective dose estimates or that were otherwise generated to document problems, to identify changes in work scope or to document variances in estimated versus actual doses. These reports were reviewed to verify that the licensee could identify problems and address them as work progressed.

These inspection samples were credited in NRC Inspection Report 05000315/2006004; 05000316/2006004.

b. Findings

No findings of significance were identified.

.2 Verification of Dose Estimates and Exposure Tracking Systems

a. Inspection Scope

The inspectors reviewed the licensee's assumptions for its collective refueling outage exposure estimate and for individual outage job estimates focusing on the estimate for the reactor head replacement, and evaluated the methodology and practices for projecting work activity specific exposures. This included evaluating both dose rate and time/labor estimates for adequacy compared to historical station specific or industry data.

The inspectors reviewed the licensee's process for adjusting outage exposure estimates when unexpected changes in scope, emergent work or other unanticipated problems were encountered which could significantly impact worker exposures.

The licensee's exposure tracking system was examined to determine whether the level of exposure tracking detail, exposure report timeliness, and exposure report distribution was sufficient to support control of outage work exposures. Radiation work permits were reviewed to determine if they covered an excessive number of work activities to ensure they allowed work activity specific exposure trends to be detected and controlled. During the conduct of exposure significant work, the inspectors evaluated if licensee management was aware of the exposure status of the work and would intervene if exposure trends increased significantly beyond exposure estimates.

These inspection samples were credited in NRC Inspection Report 05000315/2006004; 05000316/2006004.

b. Findings

No findings of significance were identified.

.3 Job Site Inspections and ALARA Controls

a. Inspection Scope

The inspectors observed aspects of the following four jobs being performed in high radiation and potentially airborne radioactivity areas for work activities that presented the greatest radiological risk to workers:

- Reactor Vessel Head Replacement;
- Reactor Coolant Pump Motor Replacement;
- Pressurizer Weld Overlay; and
- SG Manway Removal.

The licensee's use of ALARA controls for these work activities was evaluated to determine whether:

- The licensee developed and effectively used engineering controls to achieve dose reductions and to verify that the controls were consistent with the licensee's ALARA reviews; and
- Workers were cognizant of work area radiological conditions and utilized low dose waiting areas when subjected to temporary work delays.

Job performance was observed to determine if radiological conditions in the work areas were adequately communicated to workers through one of the pre-job briefings attended by the inspectors. The inspectors also evaluated the adequacy of the oversight provided by the radiation protection staff and the administrative and physical controls used over ingress/egress into these areas.

These inspection samples were credited in NRC IR 05000315/2006004; 05000316/2006004.

b. Findings

No findings of significance were identified.

.4 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed the results of Radiation Protection (RP) department self-assessments and the results of Performance Assurance Department field observations and audits of the radiation protection program to assess the licensee's ability to identify and correct problems.

The inspectors verified that identified problems were entered into the corrective action program for resolution, and that they had been properly characterized, prioritized, and were being addressed. This included ALARA program critique items and lessons learned from the licensee's previous Unit 2 refueling outage completed in the spring of 2006.

Corrective action requests generated over the 6-month period that preceded the inspection that were related to the ALARA program were selectively reviewed by the inspectors and licensee staff members were interviewed to verify that follow-up activities were being conducted in a timely manner commensurate with their importance to safety and risk using the following criteria:

- 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; and
- Identification and implementation of effective corrective actions.

The licensee's corrective action program was also reviewed to determine if repetitive deficiencies in problem identification and resolution had been addressed, as applicable.

These inspection samples were credited in NCR IR 05000315/2006004; 05000316/2006004.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety, Occupational Radiation Safety

2PS2 Radioactive Material Processing and Transportation (71122.02)

.1 Radioactive Waste System Description and Waste Generation

a. Inspection Scope

The inspectors reviewed the liquid and solid radioactive waste system descriptions in the UFSAR and the 2004 and 2005 Annual Radioactive Effluent Release Reports for information on the types and amounts of radioactive waste (radwaste) generated and disposed. The inspectors reviewed the scope of the licensee's audit/self-assessment activities with regard to radioactive material processing and transportation programs to determine if those activities satisfied the requirements of 10 CFR 20.1101©, and the quality assurance audit requirements of Appendix G to 10 CFR Part 20 and of 10 CFR 71.137, as applicable.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

.2 Radioactive Waste System Walkdowns

a. Inspection Scope

The inspectors walked down portions of the liquid and solid radwaste processing systems to verify that these systems were consistent with the descriptions in the UFSAR and in the Process Control Program, and to assess the material condition and operability of those systems. No changes were made to the radwaste processing systems in the last several years. However, the inspectors reviewed the status of available radioactive waste process equipment that had either never been utilized or otherwise not operated for well over a decade yet not declared as abandoned in-place and isolated. These systems included the waste solidification/drumming equipment, the radwaste evaporator system, and the radwaste concentrates system. The inspectors discussed with the licensee the absence of administrative and/or physical controls preventing the inadvertent use of this equipment and of a specific safety consequence evaluation to determine the impact of any inadvertent use of this equipment such as an unmonitored release or a source of unnecessary personnel exposure, which was entered into the licensee's corrective action program.

The inspectors reviewed the licensee's processes for transferring waste resin into shipping containers to determine if appropriate waste stream mixing and sampling was performed so as to obtain representative waste stream samples for analysis. The inspectors reviewed the licensee's practices for the collection of area smear surveys to represent the dry-active waste (DAW) stream and the methods used for determining the radionuclide mix of various filter media to ensure they were representative of the intended radwaste stream. Additionally, the inspectors reviewed the methodologies for quantifying gamma emitting radionuclide waste stream content, for determining waste stream tritium concentrations and for waste concentration averaging to ensure that representative samples of the waste products were provided for the purposes of waste classification pursuant to 10 CFR 61.55.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

.3 Waste Characterization and Classification

a. Inspection Scope

The inspectors reviewed the licensee's methods and procedures for determining the classification of radioactive waste shipments including the use of scaling factors to quantify difficult-to-measure radionuclides (e.g., pure alpha or beta emitting radionuclides and those that decay by electron capture). The inspectors reviewed the licensee's most recent radiochemical sample analysis results for each of the licensee's waste streams, and the associated calculations used to account for difficult-to-measure radionuclides. These waste streams consisted of primary system resins, radwaste demineralizer resins, various filter media and dry-active-waste (DAW). The licensee had not made any shipments of activated metals since the last inspection in this area, so this waste stream was not reviewed by the inspectors. The inspectors also reviewed the minimum detectable concentrations achieved for each waste stream as determined by the licensee's contract analytical laboratory compared to the corresponding radionuclide groupings in 10 CFR 61.55 to determine whether the concentration values satisfied the NRC Branch Technical Position on Radioactive Waste Classification. These reviews were conducted to determine if the licensee's program assured compliance with 10 CFR 61.55 and 10 CFR 61.56, as required by Appendix G of 10 CFR Part 20. The inspectors also reviewed the licensee's waste characterization and classification program to determine if reactor coolant chemistry data was periodically evaluated to account for changing operational parameters that could potentially affect waste stream classification and thus validate the continued use of existing scaling factors between sample analysis updates.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

.4 Shipment Preparation and Shipment Manifests

a. Inspection Scope

The inspectors reviewed the documentation of shipment packaging, surveying, package labeling and marking, vehicle inspections and placarding, emergency instructions, and licensee verification of shipment readiness for six non-excepted radioactive material and radwaste shipments made between June 2004 and October 2006. No shipments in Type B casks were made since the last NRC inspection in this area. The shipment documentation reviewed included:

- DAW in a Shielded Cask Shipped as Low Specific Activity (LSA) - II;
- Primary System Spent Resin Shipped as LSA-II;
- Primary System Spent Resin Shipped as LSA-II;
- Secondary System Spent Resin Shipped as LSA-II;
- Filters in Shielded Casks Shipped as LSA-II; and
- Contaminated Equipment Shipped as Radioactive Material Type A.

For each shipment, the inspectors determined if the requirements of 10 CFR Parts 20 and 61, and those of the Department of Transportation (DOT) in 49 CFR Parts 170-189 were met. Specifically, records were reviewed and staff involved in shipment activities were interviewed to determine if packages were labeled and marked properly, if package and transport vehicle surveys were performed with appropriate instrumentation and whether survey results satisfied DOT requirements, and if the quantity and type of radionuclides in each shipment were determined accurately. The inspectors also determined whether shipment manifests were completed in accordance with DOT and NRC requirements, if they included the required emergency response information, if the recipient was authorized to receive the shipment, and if shipments were tracked as required by 10 CFR Part 20.

Selected staff involved in shipment activities were interviewed by the inspectors to determine if they had adequate skills to accomplish shipment related tasks, and to determine if the shippers were knowledgeable of the applicable regulations to satisfy package preparation requirements for public transport with respect to NRC Bulletin 79-19, "Packaging of Low-Level Radioactive Waste for Transport and Burial," and 49 CFR Part 172 Subpart H. Also, the inspectors observed a technician conduct surveys on several packages containing DAW in preparation for their planned shipment to a waste processor later that same day. Additionally, the lesson plans for Safety Training and for General Awareness/Familiarization Training for technicians and warehouse staff were reviewed for compliance with the hazardous material training requirements of 49 CFR 172.704.

These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems for Radwaste Processing and Transportation

a. Inspection Scope

The inspectors reviewed LER (as applicable), selected action requests, self-assessment and audit reports along with field observation reports that addressed the radioactive waste and radioactive materials shipping program since the last inspection to determine if the licensee had effectively implemented the corrective action program and that problems were identified, characterized, prioritized, and corrected. The inspectors determined whether the licensee's oversight mechanisms collectively were capable of identifying repetitive deficiencies or significant individual deficiencies in problem identification and resolution.

The inspectors also selectively reviewed other corrective action program reports generated since the previous inspection that dealt with the radioactive material or radwaste shipping program, and interviewed staff and reviewed 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; and
- Implementation/consideration of risk significant operational experience feedback.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

2PS3 Radiological Environment Monitoring Program (REMP) and Radioactive Material Control Program (71122.03, 71007)

.1 Temporary Storage of Unit 1 Reactor Vessel Head

a. Inspection Scope

The inspectors reviewed the radiological controls and the survey data for the temporary storage location (the Mausoleum Building outside the main radiologically controlled area) being used for the Unit 1 reactor vessel head, which was moved from the Containment Building into the Mausoleum in October 2006. The inspectors walked down the mausoleum building to determine if the radiological controls including the contamination controls, the radiological postings and the barricades were adequate and satisfied the requirements of 10 CFR Part 20. The inspectors discussed with the licensee its plans for the characterization and waste classification of the old head incident to its disposal at a low level waste burial site planned for 2007.

This review represented one inspection sample under inspection procedure 71007, "Reactor Vessel Head Replacement."

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification (71151)

a. Inspection Scope

The inspectors sampled the licensee's submittals for the following performance indicators for the periods listed below. The inspectors used performance indicator definitions and guidance contained in Revision 4 of Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," to verify the accuracy of the performance indicator data. The following performance indicators were reviewed:

Cornerstone: Barrier Integrity

- RCS Specific Activity

The inspectors reviewed Unit 1 and Unit 2 Chemistry Department records, including isotopic analyses for selected dates in 2005 through September 2006, to determine if the greatest dose equivalent iodine (DEI) values determined during steady state operations corresponded to the values reported to the NRC. The inspectors also reviewed selected DEI calculations including the application of dose conversion factors as specified in plant TSs. Additionally, the inspectors accompanied two chemistry technicians and observed the collection and preparation of RCS samples to evaluate compliance with the licensee's sampling procedure. Further, sample analyses and calculation methods were discussed with chemistry staff to determine their adequacy relative to TSs, licensee procedures and industry guidelines.

Cornerstone: Occupational Radiation Safety

- Occupational Exposure Control Effectiveness

The inspectors reviewed licensee monthly occupational exposure control related data packages for September 2005 through September 2006. For the time period reviewed, no reportable occurrences were identified by the licensee. To assess the adequacy of the licensee's performance indicator data collection and analyses, the inspectors discussed with RP staff the scope and breadth of its data review and the results of those reviews. The inspectors independently reviewed electronic dosimetry dose rate and accumulated dose alarm reports, the dose assignments for any intakes that occurred and the licensee's AR database along with individual ARs generated during the period reviewed to verify there were no other potentially unrecognized occurrences. Additionally, as discussed in Section 2OS1, the inspectors walked down the boundaries of selected LHRAs to verify the adequacy of postings and physical barriers.

Cornerstone: Public Radiation Safety

- Radiological Effluent Technical Specification/Offsite Dose Calculation Manual Radiological Effluent Occurrence

The inspectors reviewed the licensee's corrective action program database and selected action requests generated in 2005 through September 2006, to identify any potential occurrences such as unmonitored, uncontrolled or improperly calculated effluent releases that may have impacted offsite dose. The inspectors discussed with the licensee its methods for determining effluent dose, and reviewed gaseous and liquid effluent monthly summary data and the results of selected offsite dose calculations for 2005 through the third quarter of 2006 to determine if indicator results were accurately reported.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (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 are not discussed in this report.

b. Findings

No findings of significance were identified.

.2 Semi-annual Trend Review

a. Inspection Scope

The inspectors completed one inspection sample regarding the semi-annual review of trends. 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 trend and implemented in a timely manner commensurate with the significance.

b. Findings and Observations

In general, the inspectors concluded that the trending program was effective at identifying, monitoring and correcting adverse trends. However, the inspectors identified during baseline inspections of maintenance effectiveness several deficiencies regarding Maintenance Rule implementation, which recurred throughout 2006. Specifically, the inspectors identified examples of failing to perform Maintenance Rule evaluations when required for equipment failures and, a lack of rigor and lack of attention to detail in preparation and review of action plans.

This trend was subsequently captured in AR 00803309, "Maintenance Rule Program Implementation Weaknesses," in September 2006, after the inspectors debriefed licensee personnel following inspection activities in the third quarter of 2006. Consequently, the issue was entered into the licensee's corrective action program to be evaluated and monitored as a trend. Also, two Unresolved Items (URIs) were opened regarding Maintenance Rule evaluations, URI 05000315/316/2006006-01 for the nuclear instrumentation system and URI 05000315/316/2006007-03 for the ice condenser system, to further assess the specific plant equipment problems with respect to the Maintenance Rule. Therefore, the inspectors concluded that the Maintenance Rule program implementation adverse trend was of minor significance.

4OA3 <u>Followup of Events and Notices of Enforcement Discretion</u> (71153)

.1 <u>(Closed) LER 05000315/2006-001-00:</u> "Plant Shutdown Required by Technical Specification (TS) Action 3.6.5.B.1." On the evening of July 29, 2006, operators identified that Unit 1 was exceeding the 120°F lower containment air temperature limit of TS 3.6.5, Condition 'A.' The licensee subsequently performed a reactor shutdown on July 30th to comply with TS action requirement 3.6.5, Condition 'B.' Very warm Lake Michigan water temperatures (over 80°F for many days) and warm ambient temperatures (upper 80's and low 90's) created a problem for cooling the Containment Building, which utilizes non-essential service water to provide cooling to air coolers in the Containment Building. The inspectors reviewed the licensee's actions to address the elevated lower containment temperatures and observed portions of the reactor shutdown from the Control Room during the last inspection period and identified no findings of significance. The inspectors subsequently reviewed the licensee's root cause evaluation for the event during this inspection period. The licensee reported this event as a completion of a plant shutdown required by the plant's TS in accordance with 10 CFR 50.73(a)(2)(i)(A). This event did not constitute a violation of NRC requirements. This LER is closed.

4OA5 <u>Other</u>

.1 <u>Pressurized Water Reactor Containment Sump Blockage</u> (TI 2515/166)

a. <u>Inspection Scope - Partial Completion of the TI for Unit 1</u>

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 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 1 during the Fall 2006 refueling outage. In addition, the inspectors performed a detailed review of the Unit 1 main recirculation sump 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 1 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 1 that were committed to be accomplished during the Fall 2006 refueling outage. This did not, however, complete all of the activities necessary for Unit 1 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 1.

The licensee's actions to address the issues identified in GL 2004-02 include numerous plant modifications, which are to be completed in two phases. The licensee had originally planned to complete all of the modifications on Unit 1 during the Fall 2006 refueling outage; however, the design and implementation of some of the modifications was more challenging than originally anticipated. The licensee therefore requested and received an extension of the completion date for final resolution of recirculation sump related issues in Unit 1 until the Spring 2008 refueling outage, currently scheduled to begin in March 2008. This extension allowed installation of remote strainer(s) and waterway(s) in Unit 1 to be deferred until then. The creation of additional openings in the overflow wall, modification of associated radiation shields, and installation of annulus and overflow wall debris interceptors, were also deferred until the Spring 2008 refueling outage, since they would serve no function without the remote strainer(s) and waterway(s). All other actions needed to achieve compliance with the requirements in the Applicable Requirements section of GL 2004-02 in Unit 1, including the refined evaluations and programmatic changes, are to be completed by December 31, 2007.

During the Unit 1 Fall 2006 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 inch round openings; whereas, the previous strainer consisted of nominal 1/4 inch square openings in a vertical screen and grating arrangement. The reduction in opening size represents a 300 percent improvement in filtration capability.

The following additional plant modifications were completed during the Unit 1 Fall 2006 refueling outage:

- 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. This removed 100 percent of the calcium silicate insulation assumed removed in the baseline analysis. No removal of fiberglass insulation was done because the Unit 1 containment is essentially fiberglass free.
- Removal of qualified and unqualified labels within potential loss-of-coolant accident zones of influence inside containment, and removal of a significant number of the unqualified labels inside containment. The licensee estimated that 200 ft^2 of qualified and unqualified labels were removed.
- Extension of the front recirculation sump vents 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 inch opening. The openings are now smaller than the 1/12 inch 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.
- Installation of redundant, safety-related level instruments 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 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.

- 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.
- 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.
- Capping of the existing 8-inch diameter crossover pipe between the recirculation sump and the lower containment sump. 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 inch opening.

The licensee updated numerous plant procedures to reflect the above modifications. These procedures included surveillance test procedures, alarm response procedures, normal operating procedures, and instrument calibration procedures. An update was also made to one of the emergency operating procedures, 1-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.

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

Yes. The licensee updated its licensing bases to reflect the corrective action taken thus far for Unit 1 in response to GL 2004-02. This did not, however, complete all of the licensing bases updates that will accompany the remaining modifications for Unit 1 necessary to achieve full compliance with the requirements in the Applicable Requirements section of GL 2004-02.

The inspectors reviewed the changes identified by the licensee to the UFSAR and the associated 10 CFR 50.59 screenings/evaluations and found no significant issues. No changes to the plant were identified by the licensee that would require NRC approval prior to implementing as stated in 10 CFR 50.59.

c. Findings

No findings of significance were identified.

.2 Reactor Vessel Closure Head (RVCH) 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 1 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 one spare thermocouple penetration;
- the new RVCH design eliminates seven part length CRDM penetrations;
- new CRDM mechanical assemblies;
- new TECSAs replace core exit thermocouple columns;
- the new RVCH design has a dedicated RVLIS penetration nozzle;
- the new RVCH design has a dedicated reactor RVHV penetration nozzle; 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 September 5 through September 8, 2006, from September 25 through September 29, 2006, and from October 10 through October 13, 2006, the inspectors reviewed the licensee's design changes associated with the Unit 1 RVCH, CRDM, and TECSA replacement effort.

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 the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB (1995 Edition including addenda through 1996 Addenda). 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 the ASME Code Section III (Appendix XXIII of Section III Appendices). The inspector also confirmed that the replacement RVCH was provided as a Code NPT stamped component.

b. Findings

No findings of significance were identified.

.3 Reactor Vessel Head Replacement Inspection - Enhanced Service Structure (ESS) (71007)

During the Unit 1 Fall 2006 refueling outage, the licensee installed an enhancement 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 RVHV cables;
- replacement RVHV resistance temperature detector cables;
- revised RVLIS and RVHV piping and valve layout;
- new handrail assembly on existing CRDM platform; and
- additional fall protection attachment points.

a. Inspection Scope

From September 5 through September 8, 2006, from September 25 through September 29, 2006, and from October 10 through October 13, 2006, the inspectors reviewed the licensee's design changes associated with the installation of the ESS. 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 Reactor Vessel Head Replacement Inspection - Control of Heavy Loads (71007)

a. Inspection Scope

From September 5 through September 8, 2006, from September 25 through September 29, 2006, and from October 10 through October 13, 2006, the inspectors reviewed rigging and load path calculations associated with lifting and moving the old RVCH from inside containment and lifting and moving the new RVCH through the auxiliary building to inside containment. In addition, the inspectors reviewed RVCH drop calculations and licensee procedures that control removal and replacement of the RVCH during refueling operations.

b. Findings

.1 Licensing Basis Requirements of Reactor Vessel Head Drop Analysis

Introduction: The inspectors identified an URI concerning the current licensing basis requirements for control of heavy loads related to a postulated reactor vessel head drop onto the reactor vessel flange when the RVCH is being removed or replaced during refueling operations. The licensee's refueling procedures and actual RVCH lift practices were not bounded by drop scenarios analyzed in WCAP-9198, "Reactor Vessel Head Drop Analyses," Revision 0.

Description: The inspectors reviewed licensee documentation related to the safe load path associated with removing the old RVCH from the reactor vessel and replacing the new RVCH onto the reactor vessel during refueling operations. Plant engineering specification ES-MECH-0908-QCN, "Reactor Vessel Closure Head Installation - Licensing Considerations," limited the maximum combined weight of the RVCH and ESS assembly to 340,000 pounds. ES-MECH-0908-QCN indicated documents MED-REE-3803 and WCAP-9198 shall be re-evaluated if the new RVCH with fully assembled ESS exceeded the 340,000 weight limit. Licensee documentation demonstrated that the new RVCH with fully assembled ESS would not exceed 340,000 pounds.

The inspectors also compared WCAP-9198 analyzed reactor vessel head drop scenarios against the licensee's refueling procedures associated with removing the old RVCH from the reactor vessel and replacing the new RVCH onto the reactor vessel during refueling operations. Reactor vessel head drop scenarios analyzed in WCAP-9198 include Case I: "Head Assembly Falls Approximately 14 Feet through Air While Engaged on Guide Studs and Impacts the Vessel Flange," and Case II: "Head Assembly Falls 4 Feet through Air, 24.5 Feet through Water, and Impacts the Vessel Flange." The inspectors identified that licensee had performed RVCH lifts that were outside these WCAP-9198 analyzed cases. Specifically, the licensee practice was to lift the RVCH in excess of 38 feet through air above the reactor vessel flange without flooding the reactor vessel cavity.

Subsequently, the licensee performed additional analysis, SD-060607-001, "Sensitivity Evaluation of Reactor Head Drop Analyses Presented in WCAP-9198, Revision 0," representative of the minimum lift height required to remove the existing RVCH from the reactor vessel. The analysis evaluated a 340,000 pound load drop of 34 feet (10 feet through air and 24 feet through water). After review of the evaluation by the inspectors, in consultation with NRC headquarters technical and licensing specialists, the licensee removed the existing RVCH in accordance with these analyzed parameters.

The licensee performed additional analysis, SD-060926-001, "Analysis of Postulated Reactor Head Load Drop onto the Reactor Vessel Flange," prior to installing the new RVCH onto the reactor vessel flange. The analysis evaluated 340,000 pound load drops of 33 feet 1.5" (9 feet 1.5" through air and 24 feet through water) and 15 feet through air. After review of the evaluation by the inspectors, in consultation with NRC headquarters technical and licensing specialists, the licensee installed the replacement RVCH in accordance with analyzed parameters.

The inspectors identified that in a letter dated August 27, 1982, Indiana & Michigan Electric Company, the former licensee of D. C. Cook Nuclear Plant, responded to a letter from the NRC staff dated December 22, 1980, regarding control of heavy loads. On page 9 and Table 2.3.4-c of the attachment, the licensee referenced WCAP-9198 which evaluated the consequences of dropping a reactor vessel head during refueling operations. At the time of the inspection, the licensee's position was that the reactor vessel head drop analyses evaluated in WCAP-9198 were not considered a part of the D. C. Cook licensing basis.

To address the licensing basis requirements related to WCAP-9198, the licensee has entered the concern into their corrective action program: A/R No. 00803123, "Clarify Licensing Basis Regarding Head Drop Analysis" and A/R No. 00802936, "Place Procedures on Administrative Hold."

This item is being held as an URI pending final resolution of the reactor head drop analysis licensing basis by NRC headquarters staff. (URI 05000315/2006007-05).

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

a. Inspection Scope

On September 23, 2006, the inspectors observed portions of lifting and moving the old reactor vessel closure head from the reactor vessel to the storage stand inside containment. The inspectors verified that the commitments specified in D. C. Cook Letter AEP:NRC:6514, dated September 22, 2006, regarding the head lift were adhered to. The commitments included a limitation on lift height, filling the reactor cavity with water during the lift, availability of specific safety systems and other various administrative controls.

The inspectors also observed portions of lifting and moving the old reactor closure head 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 1-MOD-55003, "Install Unit 1 Replacement Reactor Vessel Closure Head and Modify the Existing Unit 1 Service Structure." The inspectors also verified that operations personnel conducted pre-job briefings that highlighted contingency plans and response procedures for plant systems, such as spent fuel pool cooling, that could be impacted if the vessel head were dropped.

On October 27, 2006, the inspectors observed portions of lifting and installing the new reactor vessel closure head on the reactor vessel. The inspectors verified that the evolution was performed in accordance with plant procedures and that the precautions and limitations specified in procedure 12-OHP-5040-FHP-034, "Reactor Vessel Head Installation With Fuel In the Vessel," were adhered to.

For RVCH post-installation inspections, the inspectors toured the reactor cavity on November 7, 2006, with the plant at normal operating temperature and pressure to look for evidence of leakage from the reactor vessel head penetrations. On November 8, 2006, the inspectors observed rod drop testing and reviewed test data to verify that test acceptance criteria specified in procedure 01-EHP-4030-102-386, "Multiple Rod Drop Measurements," was satisfied. The inspectors also reviewed various RCS leak rate calculations following reactor vessel head installation to verify that leak rates were within TS limits.

b. Findings

No findings of significance were identified.

.6 (Closed) URI 05000315/2005004-03, "Failure to Complete Code Repair or Flaw Evaluation for Pressurizer Safe End-to-Elbow Weld Crack Prior to Returning Component to Service"

a. Inspection Scope

The inspectors reviewed actions taken to address an unresolved item (URI) identified May 3, 2005, while performing the NRC baseline Inspection Procedure 71111.08 conducted during the Unit 1 Cycle 20 refueling outage. The issue had been held as an unresolved item pending sufficient additional information to determine if violations existed (EA-06-134).

b. Findings

Failure to Take Corrective Measures for Pressurizer Weld Flaw

Introduction: The inspectors identified a minor violation of 10 CFR 50.55(a)(g)(4) related to the acceptance of a component with flaws for continued service without being dispositioned in accordance with ASME Code.

Description: On April 28, 2005, the inspectors identified that the licensee had failed to properly disposition an unacceptable weld indication found on April 21, 2005, in the downstream fusion zone of pressurizer weld 1-RC-9-01F, a stainless steel safe end-to-elbow weld, prior to changing modes and restarting Unit 1 on April 22, 2005. This action is contrary to the ASME Code. Ultrasonic testing results showed the indication to be axial, located approximately 0.09" from the pipe inner diameter and contained in an area 0.29" in length by an area of 0.30" in width. This resulted in an aspect ratio of 20 percent which exceeded the ASME flaw acceptance criteria contained in Table IWB-3514-2 of the 1989 edition of ASME Section XI.

Weld 1-RC-9-01F, located downstream of pressurizer nozzle-to-safe-end weld 1-PRZ-23, received a full structural weld overlay as part of an overlay of 1-PZR-23. The weld overlay becomes the structural weld which does not take credit for any remaining original weld or base material under the weld overlay. The licensee believed that they could take credit for the 1-PZR-23 weld overlay for weld 1-RC-9-01F and defray additional analysis or operability concerns. However, the licensee had failed to submit a required relief request (subsequently requested from the NRC and approved) in order to take credit for the overlay on weld 1-RC-9-01F and as a result was not in compliance with the ASME Code requirements.

Analysis: The inspectors determined that the failure of the licensee to properly disposition the weld indication prior to start-up of Unit 1 as required by the ASME Code, warranted a significance determination. The licensee submitted an evaluation of the 1-RC-0-01F weld flaw (TAC NO. MC7287), and while the NRC staff disagreed with the characterization and size of the initial flaw assumption in the licensee’s flaw evaluation, the staff concluded that the structural integrity of weld 1-RC-9-01F was acceptable because the structural weld overlay of 1-PZR-23, which extended over IRC-9-01F, was designed and inspected in accordance with the NRC staff’s approved Relief Request ISIR-17. The inspectors determined that due to the existence of the structural weld overlay, none of the “minor” questions in IMC-0612 Section 3 could be answered in the affirmative. After considering these, the inspectors determined that the finding was minor.

Enforcement: Title10 CFR 50.55(a)(g)(4), requires pressurized water-cooled nuclear power facility components classified as ASME Class 1, Class 2, and Class 3 to meet the requirements set forth in ASME Code Section XI.

ASME Code Section XI, IWB-3132, requires that components which do not meet the acceptance standards of Table IWB-3410-1 shall be corrected in accordance with the provisions of IWB-3132.2 (acceptance by repair), IWB-3132.3 (acceptance by replacement), or IWB-3132.4 (acceptance by analytical evaluation).

Contrary to the above, the licensee failed to accept the flaw in the pressurizer safe end-to-pipe elbow stainless steel weld 1-RC-9-01F, by repair, replacement, or analytical evaluation prior to returning Unit 1 to service on April 22, 2005. However, because the NRC staff concluded that the structural integrity of weld 1-RC-9-01F was acceptable due to the existence of the 1-PZR-23 structural weld overlay which extended over weld 1-RC-9-01F and since the overlay was designed and inspected in accordance with Relief Request ISIR-17 the inspectors concluded that this failure to comply with 10 CFR 50.55a(g)(4)ii constitutes a violation of minor significance and is not subject to enforcement action in accordance with Section IV of the NRC’s enforcement policy. Unresolved Item 0500315/2005004-03 is closed.

4OA6 Meetings

.1 Resident Inspectors' Exit Meeting

The inspectors presented the inspection results to Mr. J. Jensen and other members of the licensee's staff at the conclusion of the inspection on January 9, 2007. 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

Interim exits were conducted for:

- Occupational Radiation Safety Radiological Access Control and ALARA inspection with Mr. M. Peifer and other licensee staff on October 6, 2006.
- Inservice Inspection, with a focus on steam generators, with Mr. J. Gebbie and other members of licensee management on October 12, 2006, and on October 26, 2006, with regard to the balance of the baseline inspection. The inspectors returned proprietary information reviewed during the inspection and the licensee confirmed that none of the potential report input discussed was considered proprietary.
- Reactor vessel closure head replacement inspection with Mr. M. Peifer, and other members of the licensee management on October 13, 2006. The licensee confirmed that their contractor drawings, calculations, and design reports have been

classified as proprietary. All copies of these proprietary documents were returned to licensee staff.
- Public Radiation Safety Radioactive Waste Processing and Transportation Program inspection with Ms. S. Simpson and other licensee staff on November 8, 2006.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements, which meets the criteria of Section VI of the NRC Enforcement Manual, NUREG-1600, for being dispositioned as a Non-Cited Violation.

Cornerstone: Initiating Events Cornerstone

- In May 2002, SG tube 69/45 in SG No. 14 had been identified with ET indications indicative of manufacturing burnish marks and was designated for repair by installation of a tube plug. On October 6, 2006, the licensee identified that a plug had not been installed on the hot leg side of tube 69/45 and instead had been installed at an incorrect tube end location. 10 CFR Part 50, Appendix B, Criterion V required in part that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions. Procedure 1275284A-005 "Field Procedure for Remote Rolled Plugging Utilizing the LAN SAP Box," Revision 5, Step V.1.4.1 required verification and Step V.1.4.3 required marking of the tubes designated for repair. SG 14 tube 69/45 was designated for repair on plugging list memorandum dated May 19, 2002, and in Enclosure 1 of Procedure 1275284A-005. Contrary to these requirements, on May 19, 2002, the licensee failed to perform verification and marking of tube location 69/45 in the hot leg of SG-14 in accordance with procedure Step V.1.4.1 and V.1.4.3 of procedure 1275284A-005 prior to installation of tube plug intended to remove this tube from service. Consequently, tube 69/45 was not plugged and instead, was returned to service for three cycles of operation prior to identification in October of 2006.

  The licensee determined that past operation of SG 14 with the unplugged hot leg side of tube 69/45 did not affect the tube structural and leakage integrity based on a fabrication vendor letter which indicated that the ET signals identified were caused by thermal aging induced changes in material property and were benign. Therefore, this finding is of very low safety significance because there was no actual degradation of safety-related equipment. The licensee plugged tube 69/45 and entered this issue into the corrective action program (AR 0628009) and verified the correct location of each plugged tube in the Unit 1 and Unit 2 steam generators.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

J. Beer, Staff Health Physicist
T. Brown, Radiation Protection Manager
L. Bush, Site Senior License Holder
J. Carlson, Environmental Manager
P. Carteaux, Emergency Preparedness Manager
E. Crane, RVCH Project
R. Crane, Regulatory Affairs Specialist
T. Craven, System Engineering
M. Dixon, System Engineering
P. Donavin, Engineering Programs ISI Engineer
J. Eaton, Maintenance Rule Program Engineer
H. Etheridge, Regulatory Affairs Specialist
D. Fadel, Design Engineering Director
J. Gebbie, Plant Engineering Director
C. Graffenius, Emergency Preparedness Coordinator
D. Hafer, RVCH Project
J. Jensen, Support Services Vice President
J. Kingseed, RVCH Project
C. Lane, Engineering Programs Manager
Q. Lies, Operations Manager
J. Long, Senior Nuclear Specialist
R. Meister, Regulatory Affairs Specialist
P. Monk, Engineering Programs SG Engineer
M. Peifer, Site Vice President
S. Simpson, Regulatory Affairs Manager
W. Wah, System Engineering
D. Walton, ALARA Supervisor
L. Weber, Plant Manager
V. Woods, Performance Assurance Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000315/2006007-01	NCV	Failure to Perform 10 CFR 50.59 Evaluation to Remove Design Basis Requirement from UFSAR (Section 1R02.1)
05000315/2006007-02	NCV	Failure to Accept for Continued Service, by Correction, Evaluation or Test, a Degraded CVCS Support (Section 1R08.5)
05000315/2006007-03 05000316/2006007-03	URI	Review of Maintenance Rule Evaluations for Unit 1 and Unit 2 Ice Condensers (Section 1R12.1)
05000315/2006007-04	URI	Review of Compliance with Unit 1 RCS Boundary Leakage TS Surveillance Requirements During Plant Startup (Section 1R20.1)
05000315/2006007-05	URI	Licensing Basis Requirements of Reactor Vessel Head Drop Analysis (Section 4OA5.4)

Closed

05000315/2006007-01	NCV	Failure to Perform 10 CFR 50.59 Evaluation to Remove Design Basis Requirement from UFSAR (Section 1R02.1)
05000315/2006007-02	NCV	Failure to Accept for Continued Service, by Correction, Evaluation or Test, a Degraded CVCS Support (Section 1R08.5)
05000315/2006001-00	LER	Plant Shutdown Required by Technical Specification (TS) Action 3.6.5.B.1 (Section 4OA3.1)
05000315/2005004-03	URI	Failure to Complete Code Repair or Flaw Evaluation for Pressurizer Safe End-to-Elbow Weld Crack Prior to Returning Component to Service (Section 4OA5.6)

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 list of licensee documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document in 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 4
PMP-2291-SCH-002, Work Control Seasonal Readiness Process, Revision 2
PMI-5055, Winterization/Summerization, Revision 1
PMP-5055-001-001, Winterization/Summerization, Revision 2
12-IHP-5040-EMP-004, Plant Winterization and De-Winterization, Revision 6
AR 00804073, "Severe Winter Weather Walkdown Checklist Steps 3.1.4 and 3.2.4 Are Obsolete"
AR 06256030, "Loose / Torn Insulation on South Side of Unit 2 RWST"
AR 05017042, "Both of the West Main Steam Stop Enclosures Are Susceptible for Freezing If the Unit Is Taken Off-line"

1R02 Evaluation of Changes, Tests, or Experiments

Design Change 1-MOD-55003, Install Unit 1 Replacement Reactor Vessel Closure Head (RVCH) and Modify the Existing Unit 1 Service Structure (1-OME-1), Revision 0
Design Change 1-MOD-55520, Replace Unit 1 Reactor Vessel Closure Head (1-OME-1), Revision 0
10 CFR 50.59 Screen No. 2005 -0548-00, Document No. ES-MECH-0908-QCN, Reactor Vessel Closure Head - Licensing Considerations, Revision 0
10 CFR 50.59 Evaluation No. 2005 -0548-00, Document No. ES-MECH-0908-QCN, Reactor Vessel Closure Head - Licensing Considerations, Revision 0
10 CFR 50.59 Screen No.2006-0041-00, Document No. 1-MOD-55520, Replace Unit 1 Reactor Vessel Closure Head (1-OME-1), Revision 0
10 CFR 50.59 Screen No. 2006-0042-00, Document No. 1-MOD-55003, D. C. Cook Unit - 1 Enhanced Service Structure (ESS), Revision 0

1R04 Equipment Alignment

2-OHP-4021-008-002, Line Up Sheet 1, "Placing SI System In Standby Readiness (Manual Valves Outside Containment), Revision 17
2-HOP-4021-008-002, Line Up Sheet 3, "Placing SI System In Standby Readiness (Remote Operated Valves, Control Room), Revision 17
OP-2-5142, Flow Diagram, "Emergency Core Cooling System (SIS), Revision 50
12-OHP-4022-018-001, "Loss of Spent Fuel Pit Cooling," Revision 8
OP-12-5136, Flow Diagram, "Spent Fuel Pit Cooling and Cleanup Unit 1 and 2," Revision 21

2-OHP-4021-009-001, "Placing the Containment Spray System in Standby Readiness," Revision 10
OP-2-5144-53, "Flow Diagram Containment Spray Unit No. 2," Revision 53

1R05 Fire Protection

D. C. Cook Fire Hazards Analysis, Units 1 and 2, Revision 12
D. C. Cook Fire Pre-Plan, Units 1 and 2, Revision 2
FB-DR-007, Fire Drill Pre-Plan, November 30, 2006
AR 06335087, "Fire Drill Critique Items for November 30, 2006 Drill"
AR 06336006, "Fire Drill Critique Items for December 2, 2006 Fire Drill Debrief"
AR 06342097, "Documentation of Fire Drill and Critique"

1R08 Inservice Inspection (ISI) Activities

12-QHP-5050-NDE-001, Liquid Penetrant Examination, Revision 4
12-QHP-5050-NDE-001, Liquid Penetrant Examination, Revision 5
12-QHP-5070-NDE-005, Visual Examinations, Metallic Containment Pressure retaining components and their Integral Attachments, Revision 2
12-QHP-5070-NDE-006, Visual Examinations, Concrete Containment, Revision 2
12-QHP-5050-NDE-027, Visual Examination for Boric Acid and Condition of Component Surface, Revision 2
12-QHP-5070-NDE-001, Visual VT-2 Examination, RCS System Leakage Test, Revision 2b
12-QHP-5070-NDE-002, Visual VT-2 Examinations, Inservice and Repair/Replacements, Revision 3a
PMP-5030-001-001, Boric Acid Corrosion of Ferritic Steel Components and Material, Revision 10
ES-PIPE-1002-QCN, Operability Screening Guideline for Pipe Support Conditions and Discrepancies Found by In-Service Inspections, Revision 0
Code Case –491-1, Alternative Rules for Examination of Class 1, 2, 3, and MC Component Supports of Light-Water Cooled Power Plants Section XI, Division 1, April 30, 1993
AREVA Procedure 1246524A, "Instructions for Plug Inspection", Revision 6
AREVA Procedure 1275284A-005, "Field Procedure for Remote Rolled Plugging Utilizing the LAN SAP Box", Revision 5
ISI-UT-208, Ultrasonic Procedure for Manual Examination of Vessel Welds, Revision 0
12-EHP-5037-SGP-004, Secondary Side Visual Inspections, Revision 1
1-EHP-5037-SGP-003, SG Primary Side Inspections, Revision 6
12-EHP-5037-SGP-008, SG Management Program - In-Situ Testing, Revision 0
Site Specific Eddy Current Data Analysis Guidelines, Revision 2
Certification of Near-Distance Vision Test Charts Q.C.-2, VT-1, Q.C.-3, VT-2, Q.C.-4, VT-1, October 4, 2001
Drawing Number 1-GBD-S573, Blowdown, Revision 4
Drawing Number 1-GBD-S574, Blowdown, Revision 5
Code Interpretation XI-1-86-30R, Section XI, IWF-2430, IWF-2420(b), and IWF-3000, Visual Examination of Supports and Corrective Measures (1980 Edition and Later Editions through the 1993 Addenda), March 8, 1995
AEP:NRC:1100C, Donald C. Cook Nuclear Plant Units 1 and 2 Piping System Modifications, March 20, 1995
Examination Summary 009400, 1-PZR-15, June 6, 2005

Examination Summary 011500, 1-PZR-26, April 23, 2005
AREVA ETSS BOB001 R1 MIZ 80, October 5, 2006
AREVA ETSS RPC002 RI MIZ 80, October 5, 2006
AREVA ETSS RPC003 R1 MIZ80, October 5, 2006
Work Order O55256312 01, 2-AFW-C-4034 Correct Hanger Bolting, June 26, 2006
Job Order O05016003-38, Install Suction Piping, March 2, 2005

Corrective Action Documents

AR 00107226, 1-HE-13, Evidence of Leakage at the Inlet Piping Flange, March 29, 2005
AR 00107974, Rejectable Indication Found in Pressurizer Nozzle-to-Safe End Weld, April 9, 2005
AR 00108067, A NDE Technician was Performing a Calibration in Preparation for an Examination, April 8, 2005
AR 00108504, NRC Inspector Questioned the ability of ISI Examiner to Meet UT Procedure, April 19, 2005
AR 00112298, AEP Failed to Identify the Need for Relief from ASME Section II, Appendix VIII, Supplement 11, July 8, 2005
AR 00109921, Develop Strategy for Inspection/Mitigation of Alloy 600 and SS Welds in the ISI Program, May 18, 2005
AR 00124500, AREVA NDE Personnel were Unfamiliar with the Requirements of an Liquid Penetrant Examination Procedure, March 31, 2006
AR 00124541, CNP did not Document Section XI Evaluations Within a Number of CR's, March 31, 2006
AR 00124689, AREVA Contractor Personnel Violated Procedural Requirements, April 5, 2006
AR 00125388, NRC Inspector Felt that Unnecessary Dose was received During Preparation for an ISI Examination, April 19, 2006
AR 00108883, Indication Identified in the fusion Zone of Weld 1-RC-9-01F, April 21, 2005
AR 06102054, 2-AFW-C-4034, Loose Nut on Double Bolt Pipe Clamp, April 12, 2006
AR 06092032, NRC Observation During ISI Inspection regarding Construction Defects in ASME Code Class Pipe Supports, March 31, 2006
AR 06016013, 2-ACS-R-913, One of the Two Anchor Bolt Nuts is not Tight Against the Base Plate, January 1, 2006
eSAT 06272025, 12-QHP-5050-NDE-001, Lack of Procedural Guidance, September 25, 2006
eSAT 06270049, 1-NRV-103 Potential Weld Leak, September 28, 2006
AR 0628009, SG Tube Plugging Error During the U1C18 Outage, October 6, 2006
AR 02207014, Enhancement of SG Program, July 26, 2002
AR 05005014, EPRI Supplemental Guidance, January 5, 2005
AR 05032021, EPRI Revised Structural Performance Criteria, February 1, 2005
AR 05039056, SG Bowl Drain Welds, February 8, 2005
AR 05242029, NEI 97-06 Revision, May 30, 2005
AR 05267001, Unit 2 Sulfate Spike, September 23, 2005
AR 05284018, EPRI SG Flaw Handbook Error, October 11, 2005
AR 06130035, EPR PWR Secondary Water Guidelines Action Level 2, May 10, 2006

Corrective Action Documents As A Result of NRC Inspection

ESAT 06285023, 1-ECCS-Boron Injection Tank 1-TK-11, October 11, 2006

AR 0083738, Apparent Historical Failure to Perform the Required Extend of Condition Evaluations, October 9, 2006

Other Documents

Drawing 90D0079, 3" Plate Ultrasonic Calibration Block, Revision 0
SDG-DA-01-C21, SG Degradation Assessment - Unit 1 Cycle 21, Revision 0
Babcock and Wilcox Canada Letter DC-03-01, DC Cook Refueling Outage (U1C18) ISI Inspection Results, March 13, 2003
EPRI Letter, Review D. C. Cook Unit 1 Indications, August 28, 2002
AREVA 51-5040658-00, D. C. Cook Unit 1 - U1C19 Condition Monitoring/Operational Assessment Report, March 3, 2004
AREVA 51-5004764-06, D. C. Cook Units 1 and 2 Appendix H Eddy Current Technique Review, September 20, 2006
AREVA 51-5001545-00, M/ULC Type H Combo Probe Equivalency, August 29, 2000
AREVA 51-5001223-00, Appendix H Equivalency Cable Lengths, February 13, 1998
AREVA 51-5014354-00, Eddy Current Probe Extension Cable Comparison, September 10, 2001
AREVA 51-5001301-00, Appendix H Equivalency PWSCC Sizing at Higher Examination Speeds, May 20, 1998
AREVA 51-5002881-00, MRPC Examination Probe Extensions, Cable Lengths and Motor Unit Length, January 6, 1999

1R12 Maintenance Effectiveness

Maintenance Rule Evaluation Desktop Guide, Revision 1
ASME Code OM, Subsection ISTC, "Inservice Testing of Valves in Light-Water Reactor Nuclear Power Plants," 2001
AR 00801648, "Creation of Action Request Not Specified for IST Failure of 1-WCR-929 Described in Esat 06350681"
Control Room Logs, July 23-24, 2006
Work Order 55280395-01, "1-WCR-929, Inspect, Clean and Lube Valve Stem," July 23, 2006
AEP DIT B-02345-00, Lower Inlet Door Failure Effect on LBLOCA Analysis
Commitment 5975, Units 1 and 2 Ice Condenser Bed Surveillance Requirement
12-EHP-4030-010-262, Revision 5, Ice Condenser Surveillance and Operability Evaluation, Data Sheets 1 and 2, April 5, 2006
Ice Condenser Maintenance Rule Scoping Document, Revision 1
Ice Condenser Maintenance Rule Scoping Document, Revision 2
Maintenance Rule Evaluation for AR 00803741
Maintenance Rule Evaluation for AR 00803747
System Health and Status Report for Unit 1 Ice Condenser, 2nd Quarter 2006
System Health and Status Report for Unit 2 Ice Condenser, 2nd Quarter 2006
AR 00803314, Ice Condenser Bay #6 Left Side Lower Inlet Door
AR 06032058, Commitment #5975 to NRC states that a sample of the Intermediate Deck Doors will be opened on a weekly basis to ensure they are not frozen shut. There is no record of this being performed for the past several years.
AR 06027045, Change management issue leads to possible missed Unit One/Two Surveillance on Intermediate Deck Doors in Ice Condensers. This is NRC commitment 5975.

AR 06072005, Two Unit 2 Intermediate Deck Doors did not pass the As-Found Opening Force Test
AR 06361030, This eSAT is written to have a Maintenance Rule Evaluation
AR 06086046, While obtaining As-Found Ice Condenser flow blockage data, Bay 4 had flow Channel Blockage Exceeding the Technical Specification limit of 15%
AR 06112035, During Performance of 12-MHP-4030-010-003, As-Left Lower Inlet Door pull tests, Bay 5 right door failed the initial test. Door passed on the retest.
AR 00803747, 2 Lower Inlet Doors failed TS surveillance with high opening force
AR 00803741, 2 Lower Inlet Doors failed TS surveillance with high opening force

1R13 Maintenance Risk Assessments and Emergent Work Control

PMP-2291-OLR-001, "On-Line Risk Management," Data Sheet 1, "Work Schedule Review and Approval Form," Cycle 60, Week 4, October 15 through 21, 2006
PMP-2291-OLR-001, "On-Line Risk Management," Data Sheet 1, "Work Schedule Review and Approval Form," Cycle 60, Week 5, October 22 through 28, 2006
PMP-2291-WAR-001, "Work Activity Risk Management Process," Data Sheet 2, "Site High Level Risk Integrated Review," Risk Management Plan for Repair of 1-CRID-1-INV, December 19, 2006
PMP-4010-ODM-001, " Operational Decision Making," Data Sheet 1, "Operational Decision Making Checklist," Activity: CRID 1 Inverter Frequency Spiking, December 17, 2006
Unit 1 and 2 Part 1 Configuration Risk Assessment, October 24 and 25, November 13 through 17, November 27 through December 1, and December 14, 2006
Unit 1 Shutdown Risk Status, October 30 - 31, 2006
Unit 2 Control Room Logs, October 24 and 25, 2006
In addition, the inspectors requested copies of the following CRs be mailed to the Region III office: CR 04265022, CR04265057, CR 04265042, CR 04266002, CR 04266001, CR 04274105, CR 04274106, CR 04277043, CR04277044, CR 04279048, CR 04279057, CR 04280020, CR 04298015, CR04351012, CR 05066002, CR 05073053, CR05073064, CR 05088069, CR 05089094, CR 05096045, CR 05105046, CR 05128004, CR 05168024.

1R15 Operability Evaluations

D. C. Cook Units 1 and 2 Technical Specifications and Bases
D. C. Cook Updated Final Safety Analysis Report, Revision 20
AR 06259019, "Overboration of RCS"
AR 06271003, "Investigate Apparent Fatigue Failure Exhaust Manifold Bolt"
AR 06275019, "1-OHP-4021-082-001 Allows Backfeed in Modes 5 and 6"
AR 00802716, "Valves 1-SI-158-L1 and 1-SI-158-L4 Had a Combined Leak Rate of 21.029 GPM"
AR 06276085, "Missing Bolts on Divider Barrier Seal"
AR 06284020, "The Minimum Wall Thickness on Main Steam Line Is Higher Than Pipe Initial Thickness"
AR 06285/023, "Boron Injection Tank Telltale Drains"
AR 00802749, "Flood-up Tube #9 at 1-CEP-218 Has a Thru-wall Hole as a Result of an Arc Strike"
AR 06352129, "1-ICM-311 Is Over-thrusting Open and Closed"
AR 00801035, "Accumulator Water Temperatures Exceeding LBLOCA [Large Break Loss-of-Coolant Accident] Analysis Assumption"

Calculation D. C.-D-12-MSC-39, "Seismic Qualification of Modified EDG Exhaust Manifold Support," July 26, 1994
"Unit 1 EDG 1AB Exhaust Manifold Support - Upper Flange Bolts Evaluation," October 4, 1006
Westinghouse Technical Bulletin NSD-TB-81-7, "Boron Injection Tank Leaks," June 18, 1981
ASME Boiler and Pressure Vessel Code, Section III, Subsection C, "Requirements for Class C Vessels," Article 21, "General Requirements," 1968
ASME Boiler and Pressure Vessel Code, Section VIII, "Division I Pressure Vessels," Paragraphs UG-25, "Corrosion," and UCL-25, "Corrosion of Cladding or Lining Material," 1968
Drawing 12-AEP-W113E275-0, "900 Gallon Tank Boron Injection," Revision 0

1R17 Permanent Plant Modifications

1-MOD-65754, "Containment Sump Debris Accumulation Modifications," Revision 0
12-MHP-4030-031-001, "Inspection of Lower Containment and Recirculation Sumps," Revision 6
Design Information Transmittal (DIT)-S-00408-04, "Unit 1 and Unit 2 Containment Recirculation Sump and Lower Containment Sump Inspection Requirements," Revision 4

1R19 Post Maintenance Testing

AR 06291059, "High Peak Inrush Currents on 1-ICM-311"
AR 06292071, "Mid-Loop Boss to Pipe Weld Doesn't Meet Configuration"
Work Order 55284086-04, "Perform Diagnostic Testing of 1-ICM-311," October 17, 2006
12-IHP-5030-EMP-014, "MOV [Motor Operated Valve] Diagnostic Testing Using VIPER Test System," Revision 4
Valve Test Reports S30, S31, S33, and S34 for Stroke Testing 1-ICM-311, October 17, 2006
1-EHP-4030-103-208, "Unit 1 ECCS Flow Balance - Boron Injection System," Revision 12
1-OHP-4030-108-008R, "ECCS Check Valve Test," Attachment 6, "Charging System Check Valve Test - Preferred And Comprehensive Pump Test," Revision 6
1-MOD-40053, "RTD Bypass Elimination Modification," Revision 1
PCI Energy Services Nonconformance Report No. 7, Project No. 900684, "RTD Bypass Field Machining and Welding," October 19, 2006
1-OHP-4030-114-049, Attachment 14, "SG PORV Operability Test," December 8, 2006
Work Order 55286738-01, "SG #12 PORV Lifting," December 8, 2006
Work Order 55247905-01, "1-OME-150-CD Perform Leak Inspection / Final Run PMT," October 18, 2006
Work Order 55242232, 01-EHP-4030-102-386, "Multiple Rod Drop Measurements," November 8, 2006
1-EHP-4030-150-255, "Verification of Main Turbine Trips," Revision 007
Work Order 55231271-01, "1-XRV-222 Replace Valve/Actuator," October 3, 2006
Work Order 55284371-04, "Replace Voltage Regulator," October 19, 2006
12-IHP-6030-032-001, "Emergency Diesel Generator Automatic Voltage Regulator Tuning and Adjustment," October 19, 2006
1-OHP-4030-132-027CD, "CD Diesel Generator Operability Test (Train A)," October 19, 2006
Work Order 55247905-33, "Replace 2RB Fuel Injector on 1-OME-150-CD-EN," October 18, 2006
1-OHP-4030-132-027AB, "AB Diesel Generator Operability Test (Train B)," October 4, 2006

1R20 Refueling and Outage Activities

PMP-4100-SDR-001, "Plant Shutdown Safety and Risk Management," Revision 13
1-OHP-4021-002-005, "RCS Draining," Revision 33
1-OHP-4030-127-041, "Refueling Integrity," Revision 10
1-OHP-4021-001-002, "Reactor Start-Up," Revision 35
AR 00802582, "Aggregate Ice Basket Evaluation"
AR 00802733, "Exceeded Heat Up Limits on the Pressurizer"
AR 00802749, "Flood-up Tube #9 at 1-CEP-218 has a Thru Wall Arc Strike"
AR00802871, "Diaphragm Air Leak Preventing Valve Operation"
AR 00802944, "1-MRV-223 #2 SG PORV on 50% Open With 100% Demand"
AR 00803600, "Could Not Complete all Sections of 1W CCW Surveillance"
AR 00803614, "Low Flow to CTS Heat Exchanger During Essential Service Water Flow Verification Test"
AR 00804326 PMP-2080-EPP-101 Revision Required Prior to Mode 4
AR 00804658, "Transformer 101CD LTC Trouble Due to Loss of Voltage"
AR 00804713, "Evaluate Nonintrusive Test Data for 1-SI-152N/S"

1R22 Surveillance Testing

D. C. Cook Unit 1 Technical Specifications and Bases
D. C. Cook Updated Final Safety Analysis Report, Revision 20
12-MHP-4030-010-003, "Ice Condenser Lower Inlet Door Surveillance," October 5, 2006
1-EHP-4030-134-001, "Unit 1 Primary Containment Leak Rate Running Total," Revision 1
1-EHP-4030-134-203, "Unit 1 LLRT [Local Leak Rate]," Revision 4
12-EHP-5030-001-008, "Recirculation Loop Total Leak Rate," Revision 7
Design Information Transmittal DIT-B-02434-00, "Emergency Core Cooling System Recirculation Loop Total Leak Rate," Revision 0
Letter AEP: NRC:00334B from J. Dolan, Indiana & Michigan Electric Company, to H. Denton, USNRC, "NUREG 0578 Requested Action Items to Complete Post-Implementation," 03/10/80
Letter from J. Stang, USNRC, to A. Bakken III, IMPC, "D. C. Cook Nuclear Plant, Units 1 and 2 - Issuance of Amendments (TAC Nos. MB5318 and MB5319)," 11/14/02
AR 00803500, "LLRT Failure of 1-CCW-244-72"
2-OHP-4030-216-020W, West Component Cooling Water Loop Surveillance Test
AR 06291060, "Could Not Complete Valve Stroke of 2-CRV-445"
AR 06291019, "CCW Surveillance Revision Change Request"
1-OHP-4030-132-217A, "DG1CD Load Sequencing & ESF Testing," November 2, 2006
AR 06321044, "TS Basis Inconsistent on DG "Ready to Load" Definition"
AR 06321039, "NRC Raised Concern Meeting SR 3.8.1.17"
AR 00803064, "1-SI-101 Has Minor Leak for Trending"
AR 06279012, "2 Lower Inlet Door Failed TS Surveillance With High Opening Force"

1EP6 Drill Evaluation

EMD-32a, Nuclear Plant Event Notification, November 29 and December 12, 2006
RQ-E-3106A, Period 3106 As-Found Simulator Evaluation A, Revision 1
RQ-E-3106B, Period 3106 As-Found Simulator Evaluation B, Revision 1

2OS1 Access Control to Radiologically Significant Areas

12-THP-6010-RPP-006, Radiation Work Permit (RWP) Processing, Revision 24
12-THP-6010-RPP-104, Issue and Control of Special Dosimetry, Revision 07
12-THP-6010-RPP-206, Internal Dose Assessment and Calculation, Revision 05
12-THP-6010-RPP-211, Operation of the Canberra Fastscan Whole Body Counter, Revision 06
12-THP-6010-RPP-703, Monitor Alarm Response and Personnel Decontamination, Revision 23
AR 05298035, Self-Assessment Report - Radiation Protection Program - Prevention Review Board, September 05, 2006
AR 06013003, Quick Hit Assessment of Radiation Worker Practices, January 13, 2006
AR 06013005, Quick Hit Assessment of Non-Consequential Procedural Errors, 01/13/06
AR 06002014, RP As-Found During Routine LHRA Initial Entry Survey, 01/02/06
AR 06016059, PA Audit Tour of the Building, January 16, 2006
AR 06016059, Rad Worker Received 59 mrem with a 50 mrem ED Dose Setpoint, 10/01/06
AR 06112042, A Worker Received an Uptake while Removing Strippable Paint, 04/22/06
AR 06260062, U1C21 Forced Oxidation Cobalt Peak Results, September 17, 2006
AR 06274001, Wrong RWP Assigned to Scaffold Activity, October 01, 2006
PMP-6010-RPP-003, High, Locked High, and Very High Radiation Area Access, Rev.17
PMP-6010-RPP-006, Radiation Work Permit Program, Revision 09
PMP-6010-RPP-200, Internal Radiation Dose Monitoring, Revision 06
Replacement of Reactor Vessel Closure Heads at Cook Nuclear Plant, Project Plan, RVCH-2005-002-REP, Revision 0
RWP 06-1049, CVCS Work Activities, Revision 01
RWP 06-1107, Unit-1 RVCH Replacement Package, Revision 00/02
RWP 06-1147, U1C21 - S/G Manway and Diaphragm Activities, Revision 00
RWP 06-1148, U1C21 - SG Platform Activities, Revision 01
RWP 06-1177, U1C21 - 1-Mod-40053 - RTD Bypass Removal, Revision 02
RWP 06-1188, U1C21 - Debris Removal Activities, Revision 00
RWP 06-1190, Alloy 600 Weld Overlay Repairs on Pressurizer, Revision 00
Technical Specification 5.7 High Radiation Area, Amendment No. 287
THG-026, Locked High radiation Area and Very High Radiation Area Weekly Verification Process, Revision 07

2OS2 As-Low-As-Reasonably-Achievable Planning and Controls

12-THP-6010 RPP-014, TEDE Calculation Sheets, Revision 7, Various Jobs
12-THP-6010-RPP-405, Analysis of Airborne Radioactivity, Revision 10
ALARA In-Progress Review, U1C21 Reactor Vessel Head Replacement, September 26, 2006
AR 00801884, Chain Configuration on Door Corrected, August 23, 2006
AR 06061048, Quick Hit Assessment of the U2C16 Dose Estimation Process, March 2, 2006
AR 06260036, More Outage Dose Received Than Was Necessary, September 17, 2006
AR 06264047, Expanded Use of Radiation Source Term Technology, September 21, 2006
PMP-6010-ALA-001, ALARA Program - Review of Plant Work Activities, Revision 15
Radiological Area Status Sheets, Various Areas
Radiography Shot Plan, Elevation 587' G.D. Valve Gallery, October 2, 2006
RWP Totals Reports, October 3, 2006 and October 5, 2006
U1C21 Outage Daily ALARA Report, October 1, 2006
U1C21 Outage Job Checklist, Various 2006
U2C16 Post Outage Draft Report, April 2006

2PS2 Radioactive Material Processing and Transportation

Annual Radioactive Effluent Release Reports for 2004 and 2005, Tables Summarizing Solid Waste and Irradiated Fuel Shipments, March 28, 2005 and March 29, 2006
PMP-6010-PCP-900, Radioactive Waste Process Control Program, Revision 7
12-THP-6010-RPP-900, Preparation of Radioactive Shipments, Revision 16
12-THP-6010-RPP-905, Solid Waste Handling and Packaging, Revision 10
12-THP-6010-RPP-903, Activity Determination and Waste Classification, Revision 5
12-THP-6010-RPP-902, Dewatering of High Integrity Containers, Revision 4
12-THP-6010-RPP-908, Surveys, Inventory and Inspection of Stored Radioactive Waste, Revision 5
2005 Scaling Factor Determination Report and Scaling Factor Data Analyses for Primary Filters, Radwaste Demineralizer Resin, Primary System Resin and Dry Active Waste, October 24 - 30, 2005
General Engineering Laboratory, LLC, Report of 10 CFR Part 61 Analyses and Quality Control Data, September 14, 2005
12-THP-6010-RPP-913, Scaling Factor Determination, Revision 1
AR 05278030, Vendor Lab Analyses for 10 CFR 61 do not Correspond Well with Onsite Results, October 5, 2006
Shipment Manifest, Radiological Surveys and Associated Documentation for Shipment RMC-04-014, High Activity DAW, June 10, 2004
Shipment Manifest, Radiological Surveys and Associated Documentation for Shipment RMC-04-068, Primary System Resins, December 8, 2004
Shipment Manifest, Radiological Surveys and Associated Documentation for Shipment RMC-05-083, Primary System Resins, November 4, 2005
Shipment Manifest, Radiological Surveys and Associated Documentation for Shipment RMC-05-086, SG Blowdown Resins, November 18, 2005
Shipment Manifest, Radiological Surveys and Associated Documentation for Shipment RMC-06-056, Primary Filters, June 21, 2006
Shipment Manifest, Radiological Surveys and Associated Documentation for Shipment RMC-06-095, Reactor Coolant Pump Impeller, October 9, 2006
Environmental Technician Training Lesson Plan, Radioactive Waste Course No. EA-C-RW01, Revision 3
Environmental Technician Training Lesson Plan, Transporting DOT Hazardous Materials, Course No. EA-C-HZ01, Revision 2
AR 06293026, Radioactive Shipping Package Content Surveys Suspect, October 20, 2006
AR 04029036, 05062113 & 06307031, Abandoned Radwaste Equipment, January 29, 2004, May 24, 2004 and November 3, 2006
AR 05227032, Improperly Characterized SCO Shipment, August 15, 2005
AR 04180053, Container Integrity Evaluation, June 24, 2004
Performance Observation Cards No. 48572, 49286 and 56994, Observations of Shipping Evolutions, dated between November 8, 2005 and August 3, 2006
Report of Performance Assurance Audit PA-05-01, Radiation Protection, February 2005
Report of Self-Assessment No. SA-2004-REA-001-F, Packaging and Shipping of Radioactive Material and Radioactive Waste, October 10, 2004
Report of Self-Assessment of Radioactive Shipping Program, March 24, 2006
Report of Performance Assurance Audit PA-06-01, Radiation Protection, March 6, 2006

Performance Assurance Field Observation No. FO-06-J-004 and FO-05-I-002, Review of Shipment Paperwork, August 26, 2005 and October 9, 2006

2PS3 Radiological Environment Monitoring Program (REMP) and Radioactive Material Control

Radiological Survey Data for the Mausoleum Building, various dates in October 2006

4OA1 Performance Indicator (PI) Verification

Monthly Summary Data of Dose Calculations and Projections Due to Liquid and Gaseous Effluents, January 2005 - September 2006
1-THP-6020-Chem-121 RCS Sampling and Chemical Addition, Revision 00
12-THP-6020-Chem-101 RCS, Dose Equivalent Iodine Determination, Various Revisions
12-THP-6020-Chem-109 Chemical and Volume Control System, Revision 14
AR 06051053, Quick Hit Assessment of ANI Dosimetry Records Retrieval, February 20, 2006
Chemistry Isotopic Analyses, Various Counts and Dates
ED Dose/Rate Alarm Transaction Reports, June 2005 - September 2006
Internal Dose Assessment Calculations and Whole Body Count Data, January - September 2006
PMP-7110-PIP-001, Regulatory Oversight Program Performance Indicators, Revision 5

4OA2 Problem Identification and Resolution

2006 Trend Closure Status
Second Quarter 2006 Event Code Report
AR 00121468, "The Surveillance Data Base Tracking System is Not Robust"
AR 00128038, "CARB #350 Held on June 16, 2006 Did Not Concur With The IDACE"
AR 00117568, "Learning Organization Department Self Assessment"
AR 06066065, "February Event Code Data Shows an Adverse Trend in Event Code C2a"

4OA3 Followup of Events and Notices of Discretion

LER 05000315/2006-001-00, "Plant Shutdown Required by TS Action 3.6.5.B.1," September 27, 2006
Root Cause Evaluation (AR 00801033), "Lower Containment High Temperature Shutdown July 2006"

4OA5 Other

Reactor Vessel Head Replacement

AR 6276023, RVCH Replacement Nozzle Penetration J-groove PT Examinations, October 3, 2006
AREVA NCR 2006-4147, PT Inspection J-Groove Welds Under RVCH, September 26, 2006
2NDE-1065-00, Liquid Penetrant Examination Reports J-Groove Welds Nozzles 1-70, September 23, 2006 through September 24, 2006
AREVA 54-PT-200-06, Color Contrast Solvent Removable Liquid Penetrant Examination of Components, March 16, 2006

TI 2515/166

NRC Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors, September 13, 2004
Letter AEP:NRC:5054-11 from J. Jensen, Indiana Michigan Power, to USNRC, "D. 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 USNRC, "D. 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, to USNRC, "D. 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, to USNRC, "D. 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 from P. Tam, USNRC, to M. Nazar, Indiana Michigan Power, "D. C. Cook Nuclear Plant, Unit 1 (D. C.CNP-1) - Extension of Completion Date for Actions in Response to GL 2004-02 (TAC NO. MC4679)," July 28, 2006

Replacement Reactor Vessel Closure Head

Design Change 1-MOD-55520, Replace Unit 1 Reactor Vessel Head (1-OME-1), Revision 0
Document No. 06 NIAVP 02, Analytical Review, NCR ILH No. June 6, 2006, D. C. Cook Unit 1, May 31, 2006
Document No. 6 GA 20260, ASME Code Reconciliation: Modified L-106A CRDM Appurtenances, D. C. Cook Units 1 & 2, Revision B
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 4
Document No. 08-5070166, Design Specification: Thermocouple Column Sealing Assembly, D. C. Cook Units 1 & 2, Revision 3
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 2
Document No. 32-9007359, Seismic and LOCA Analysis, D. C. Cook - CRDM, Revision 1
Document No. 32-9012285, Calculation: D. C. Cook Units 1 & 2 Thermocouple Sealing Assembly (TECSA) Analysis, Revision 3
Document No. 32-9001581, Calculation: D. C. Cook Units 1 & 2, CRDM Nozzle to Adapter Weld and Taper, Revision 1
Document No. 32-9001581, Calculation: D. C. Cook Units 1 & 2, CRDM Nozzle to Adapter Weld and Taper, Revision 2
Document No. 32-9003111, Calculation: D. C. Cook Units 1 and 2, RVCH Lifting Lug and Support Lug Qualification, Revision 1

Document No. 32-9003111, Calculation: D. C. Cook Units 1 and 2, RVCH Lifting Lug and Support Lug Qualification, Revision 2
Document No. 32-9003249, Calculation: D. C. Cook Units 1 and 2, RVCH Sizing Calculation, Revision 1
Document No. 32-9003249, Calculation: D. C. Cook Units 1 and 2, RVCH Sizing Calculation, Revision 2
Document No. 32-9003276, Calculation: D. C. Cook Units 1 and 2, Reactor Vessel Head Vent/RVLIS Nozzle Connection, Revision 1
Document No. 32-9003276, Calculation: D. C. Cook Units 1 and 2, Reactor Vessel Head Vent/RVLIS Nozzle Connection, Revision 2
Document No. 32-9004298, Calculation: D. C. Cook, Replacement RVCH CRDM/TC Nozzle Connection, Revision 1
Document No. 32-9004298, Calculation: D. C. Cook, Replacement RVCH CRDM/TC Nozzle Connection, Revision 2
Document No. 32-9005921, Calculation: D. C. Cook Unit 1, Closure Analysis with Replacement Head, Revision 1
Document No. 32-9006361, Calculation: D. C. Cook Unit 1, Appendix G RV Closure Head Limits, Revision 1
Document No. 32-9006361, Calculation: D. C. Cook Unit 1, Appendix G RV Closure Head Limits, Revision 2
Document No. 32-9006790, Calculation: D. C. Cook Unit 1, Design Basis RV Closure Head Limits, Revision 1
Document No. 32-9006790, Calculation: D. C. Cook Unit 1, Design Basis RV Closure Head Limits, Revision 2
Document No. 33-9009036, ASME Design Report: D. C. Cook - Unit 1, Replacement Reactor Vessel Closure Head, Revision 0
Document No. 33-9013947, Design Report: Control Rod Drive Mechanism Pressure Housing Assembly, Appurtenances ASME III Class 1, D. C. Cook Nuclear Plant Units 1 & 2, Revision 3
Document No. 33-9023774, ASME Design Report: Thermocouple Column Sealing Assembly (TECSA), D. C. Cook - Units 1 & 2, Revision 1
Document No. 51-9002069, Engineering Information Record: D. C. Cook Unit 1, Reactor Vessel Closure Head Code Reconciliation, Revision 0
Document No. 51-9025725, Engineering Information Record: TECSA Section XI Code Reconciliation, Revision 1
Document No. 87-5072390, Contract Variation Approval Request (CVAR), Revision 0
Document No. 87-9013375, Contract Variation Approval Request (CVAR), Revision 1
Document No. 87-9016326, Contract Variation Approval Request (CVAR), Revision 0
Document No. 87-9016841, Contract Variation Approval Request (CVAR), Revision 0
Document No. 87-9016938, Contract Variation Approval Request (CVAR), Revision 0
Document No. 87-9022719, Contract Variation Approval Request (CVAR), Revision 0
Document No. 87-9022844, Contract Variation Approval Request (CVAR), Revision 0
Document No. JQA-04-005, Certified Material Test Report: Japan Steel Works, ASME SA-508, Gr. 3, Cl. 1, Heat No. 03W104-1-1, D. C. Cook Unit 1, March 25, 2004
Drawing No. 1GA20011, C.R.D.M., Design Drawing, I.L.H. Assembly Details, D. C. Cook Unit 1, Revision G
Drawing No. 1GA20013, L106A, Comparison between Original Design and Improved Design, D. C. Cook Unit 1, Revision A
Drawing No. 1GA20014, C.R.D.M., Pressure Housing Assembly, D. C. Cook Unit 1, Revision E
Drawing No. 2GA20081, C.R.D.M., L106A and L106B, Revision B

Drawing No. 2GA20134, Sheet 1, Thermocouple Column Sealing Assembly, Revision F
Drawing No. 4GA20152, Thermocouple Column Tie Rod, Revision A
Drawing No. 5054718E, Specification Drawing, D. C. Cook Unit 1, Reactor Vessel Closure Head, J-Groove Weld Details, Revision 5
ES-MECH-0902-QCN, Engineering Specification: Certified Design Specification for Reactor Vessel Replacement Head, Revision 0
ES-MECH-0903-QCN, Engineering Specification: Reactor Vessel Replacement Head, Revision 0
ES-MECH-0904-QCN, Engineering Specification: Replacement Control Rod Drive Mechanisms, Revision 0
Reactor Vessel Closure Head, Serial No.: CC/DK11, ASME NPT Certification, AREVA NP, Year Built: 2006
12-OHP-4050-FHP-034, "Reactor Vessel Head Installation With Fuel in the Vessel," Revision 8
12-OHP-4050-FHP-023, "Reactor Vessel Head Removal With Fuel in the Vessel," Revision 5

Replacement Enhanced Service Structure

Design Change 1-MOD-55003, Install Unit 1 Replacement Reactor Vessel Closure Head (RVCH) and Modify the Existing Unit 1 Service Structure (1-OME-1), Revision 0
Document No. 08-5059507, Design Specification: Enhanced Service Structure for D. C. Cook Units 1 & 2, Revision 3
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 2
Document No. 32-5068857, Design Report: D. C. Cook Unit 1, Enhanced Service Structure (ESS), Revision 2
Document No. 32-9002275, Calculation: Core Exit Thermocouple Cable Tray Support Details - Units 1 & 2, D. C. Cook ESS, Revision 1
Document No. 32-9003174, Calculation: D. C. Cook ESS Tie Rod Embedment Qualification, Revision 1
Document No. 51-9003721, Engineering Information Record: LOCA Loadings on the ESS, Revision 2
Document No. 51-9006451, Engineering Information Record: D. C. Cook Unit 1 RCS Evaluation, Revision 1
Document No. 87-9021957, Contract Variation Approval Request (CVAR), Revision 0
Document No. 2006-890, Non-Conformances, Problems, and Concerns Tracking Form
Document No. 2006-2186, Non-Conformances, Problems, and Concerns Tracking Form
Drawing No. 02-9005587E, Cable Tray Support Details, D. C. Cook Units 1 & 2, Enhanced Service Structure (ESS) Project, Revision 2
ES-CIVIL-0427-QCN, Engineering Specification: Field Installation of Hilti Kwik Bolt II Expansion Anchors, Revision 0
ES-MECH-0905-QCN, Engineering Specification: Replacement Integrated Head Assembly (IHA) and Service Structure Upgrades (SSU), Revision 0
SDS-88, Structural Design Standard: Recommended Expansion Type Anchors, Revision 12

Supporting Modifications - Replacement Reactor Vessel Closure Head

Design Change 12-MOD-55633, Replace Auxiliary Building Roll-Up Door 12-DR-AUX381, Revision 0

SD-050503-001, Qualification of Concrete and Structural Steel Components for Installation of New Roll-Up Door 12-DR-AUX381 Per 12MOD-55633, Revision 0

Control of Heavy Loads - Replacement Reactor Vessel Closure Head

Calculation No. 06Q3576-01, Auxiliary Building East Crane Seismic Adequacy for the Reactor Vessel Head Replacement Lift with a 0.05g and 0.07g Earthquake Using OBE Acceptance Criteria, Revision 1
Calculation No. DC-D-3195-375-SC, Development of In-Structure Seismic Response Spectra within the D. C. Cook Nuclear Plant Auxiliary Building at the Crane Support Level, Revision 0
Calculation No. DC-D-3195-376-SC, Evaluation of Seismic Response of Auxiliary Building Crane Using Revised Vertical Response Spectra wit Full Load and SSE, Revision 0
AR 04323067, OE: Westinghouse Nuclear Advisory Letter NSAL-04-6, "Reactor Vessel Head Drop Analysis", November 18, 2004
AR 05090002, Unit 1 Reactor Lift Rig Damaged, March 30, 2005
AR 05307053, OE: Review of Regulatory Issue Summary 2005-25, Clarification of NRC Guidelines for Control of Heavy Loads, by the Heavy Loads Program Owner, November 3, 2005
Document No. 02-9019066B, Sketch: Unit 1 Containment Runway, Plan View, RVCH Replacement Project, D. C. Cook Units 1 & 2, Revision 1
Document No. 23-9005081, QA Data Package: RVCH Tripod Sling Block Assembly and Load Test, Revision 1
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
Document No. 32-9018873, Rigging Calculation: Runway, Revision 2
Document No. 32-9019055, Lifting New/Old Heads Horizontal Rigging System, Revision 6
Document No. 32-9026717, Rigging Calculation: Lifting New Head w/ Shipping Container, Revision 2
Document No. 51-5068632, Engineering Information Record: D. C. Cook TRD - RVCH Replacement Heavy Rigging & Lifting - Bigge, Revision 2
Document No. 51-5068632, Engineering Information Record: D. C. Cook TRD - RVCH Replacement Heavy Rigging & Lifting - Bigge, Revision 3
Document No. 51-9005936, Engineering Information Record: D. C. Cook Unit 1, RVCH Weight Evaluation, Revision 0
Document No. 51-9006064, Engineering Information Record: D. C. Cook Unit 1, RVCH Lift Weight Evaluation, Revision 1
Document No. 51-9006092, Engineering Information Record: D. C. Cook Unit 1, RV Head Drop Evaluation, Revision 1
Drawing No. 113E326, Block Sling Assembly, Head Lifting Rig, Revision 3
Drawing No. 113E529, Head Lifting Rig - General Assembly, Revision 7
ES-MECH-0908-QCN, Engineering Specification: Reactor Vessel Closure Head Installation - Licensing Considerations, Revision 0
Letter AEP:NRC:00514A, R. S. Hunter (I&MEC) to H. R. Denton (NRC), Subject: Control of Heavy Loads - Phase II.a, August 27, 1982
Letter from J. F. Stang (NRC) to R. P. Powers (IMPC), Subject: Issuance of Amendments - D. C. Cook Nuclear Plant, Units 1 and 2, Re: Movement of SG Sections in the Auxiliary Building for SG Replacement Project, December 7, 1999
MED-REE-3803, Evaluation of the Permanent Reactor Vessel Head Radiation Shield, Revision 1

NSAL-04-6, Westinghouse Nuclear Safety Advisory Letter: Reactor Vessel Head Drop Analysis, November 15, 2004
Procedure No. PMP-4050-CHL-001, Maintenance Procedure: Control of Heavy Loads, Revision 6
Procedure No. 12-OHP-4050-FHP-023, Reactor Vessel Head Removal with Fuel in the Vessel, Revision 2
Procedure No. 12-OHP-4050-FHP-028, Reactor Vessel Head Installation with No Fuel in the Vessel, Revision 3
Procedure No. 12-OHP-4050-FHP-029, Reactor Vessel Head Removal with No Fuel in the Vessel, Revision 2
Procedure No. 12-OHP-4050-FHP-034, Reactor Vessel Head Installation with Fuel in the Vessel, Revision 6
SD-060907-001, Sensitivity Evaluation of Reactor Vessel Head Drop Analysis Presented in WCAP-9198, Revision 0, Revision 2
SD-060907-001, Sensitivity Evaluation of Reactor Vessel Head Drop Analysis Presented in WCAP-9198, Revision 0, Revision 3
SD-060907-001, Sensitivity Evaluation of Reactor Vessel Head Drop Analysis Presented in WCAP-9198, Revision 0, Revision 4
SD-060926-001, Analysis of Postulated Head Drop onto the Reactor Vessel Flange, Revision 0
WCAP-9198, Reactor Vessel Head Drop Analysis, Revision 0
WCAP-9198, Reactor Vessel Head Drop Analysis, Revision 1
WCAP-10230, Evaluation of the Acceptability of the Reactor Vessel Head Lift Rig, Reactor Vessel Internals Lift Rig, Load Cell, and Load Cell Linkage to the Requirements of NUREG-0612, Revision 1

Action Requests Initiated as a Result of NRC Inspection

AR 00802936, "Place Procedures on Administrative Hold"
AR 00803123, "Clarify Licensing Basis Regarding Head Drop Analysis"
AR 00803398, "NRC Question on Safe Shutdown Earthquake Categorization"
AR 00803828, "UFSAR Change Not Adequately Addressed in ESS 50.59 Review"
AR 00804059, "Methodology Errors in AREVA Calculation 32-9002775-001 for RVCH Project"
AR 06286030, "Improper Temporary Storage of Rails During RVCH Move on 650"

LIST OF ACRONYMS USED

ADAMS	Agency-wide Documents Access and Management System
ALARA	As-Low-As-Reasonably-Achievable
AR	Action Request
ASME	American Society of Mechanical Engineers
BACC	Boric Acid Corrosion Control
CEDE	Committed Effective Dose Equivalent
CFR	Code of Federal Regulations
CRDM	Control Rod Drive Housing Mechanism
CVCS	Chemical and Volume Control System
DAW	Dry-Active Waste
DBE	Design Basis Earthquake
DEI	Dose Equivalent Iodine
°F	Degrees Fahrenheit
U1C21	D. C. Cook Unit 1 Cycle 21 Refueling Outage
DOT	Department of Transportation
ECCS	Emergency Core Cooling System
ED	Electronic Dosimeter
EDG	Emergency Diesel Generator
EPRI	Electric Power Research Institute
ESS	Enhanced Service Structure
ET	Eddy Current Testing
GL	Generic Letter
IST	Inservice Testing
LER	Licensee Event Report
LHRA	Locked High Radiation Area
LLRT	Local Leak Rate Test
LSA	Low Specific Activity
MRE	Maintenance Rule Evaluation
NCV	Non-Cited Violation
NDE	Nondestructive Examination
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records
PI	Performance Indicator
Radwaste	Radioactive Waste
RCS	Reactor Coolant System
RIS	Regulatory Issue Summary
ROP	Reactor Oversight Process
RP	Radiation Protection
RVCH	Reactor Vessel Closure Head
RVLIS	Reactor Vessel Level Instrumentation System
RWP	Radiation Work Permit
SDP	Significance Determination Process
SG	Stream Generator
SSC	Structures, Systems, and Components

TECSA	Thermocouple Column Sealing Assemblies
TI	Temporary Instruction
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
UT	Ultrasonic Examination
VHRA	Very High Radiation Area