UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
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LISLE, IL 60532-4352

October 30, 2012

Mr. Larry Weber
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 05000315/2012004; 05000316/2012004; 07200072/2012001; 07200072/2012003; AND 07200072/2012004

Dear Mr. Weber:

On September 30, 2012, 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 results of this inspection, which were discussed on October 9, 2012, with Mr. J. Gebbie, and other members of your staff.

The 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.

One NRC identified finding of very low safety significance (Green) was identified during this inspection. This finding was determined to involve a violation of NRC requirements. Additionally, the NRC has determined that a traditional enforcement Severity Level IV violation occurred. The NRC is treating these violations as a non-cited Violations (NCVs) consistent with Section 2.3.2 of the Enforcement Policy.

If you contest the subject or severity these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at D.C. Cook.

If you disagree with the cross-cutting aspect assignment in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III; and the NRC Resident Inspector at D.C. Cook, Units 1 and 2.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's Agencywide Document Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

John B. Giessner, Chief
Branch 4
Division of Reactor Projects

Docket Nos. 50-315; 50-316; and 072-00072
License Nos. DPR-58 and DPR-74

Enclosure: Inspection Report 05000315/2012004; 05000316/2012004; 07200072/2012001; 07200072/2012003; and 07200072/2012004
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	05000315; 05000316; and 072-00072
License Nos:	DPR-58; DPR-74
Report No:	05000315/2012004; 05000316/2012004; 07200072/2012001; 07200072/2012003; 07200072/2012004
Licensee:	Indiana Michigan Power Company
Facility:	D.C. Cook Nuclear Power Plant, Units 1 and 2
Location:	Bridgman, MI
Dates:	July 1 through September 30, 2012
Inspectors:	J. Ellegood, Senior Resident Inspector P. LaFlamme, Resident Inspector Rhex Edwards, Reactor Inspector Matthew Learn, Reactor Engineer J. Lennartz, Project Engineer V. Meghani, Reactor Inspector C. Morell, Safety Inspector M. Mitchell, Health Physicist James Neurauter, Senior Reactor Inspector J. Parrott, Safety Inspector R. Temps, Senior Safety Inspector
Approved by:	John B. Giessner, Chief Branch 4 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

Inspection Report (IR) 05000315/2012004, 05000316/2012004, 07200072/2012001, 07200072/2012003, 07200072/2012004; 07/01/2012 – 09/30/2012; D.C. Cook Nuclear Power Plant, Units 1 & 2; Flooding; Other Activities

This report covers a 3-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. One Green finding was identified by the inspectors. The finding was considered a non-cited violation (NCV) of NRC regulations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Cross-cutting aspects were determined using IMC 0310, "Components Within the Cross-Cutting Areas." Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

- Severity Level IV: A Severity Level IV NCV of very low safety significance of Title 10 of the Code of Federal Regulations (CFR) Part 72.150, "Instructions, Procedures, and Drawings," was identified by the inspectors for the failure of the licensee to have procedures in place to ensure that the design basis peak fuel cladding temperature limit would not be exceeded during dry cask canister processing operations. The licensee took appropriate actions prior to conducting evolutions that may have challenged these limits. This has been documented in the licensee's corrective action program as Action Request (AR) 2012-9676.

  Consistent with the guidance in Section 2.2 of the NRC Enforcement Manual, Independent Spent Fuel Storage Installation (ISFSIs) are not subject to the Reactor Oversight Process enforcement and, thus, traditional enforcement will be used for these facilities. Therefore the violation was dispositioned using the traditional enforcement process using Section 2.3 of the Enforcement Policy. The violation was determined to be of more than minor significance using IMC 0612, "Power Reactor Inspection Reports," Appendix E, "Examples of Minor Issues," Example 3i, since the bounding conditions for the analyzed thermal condition was not reflected in the procedures to perform the port cap repair. Specifically, the licensee's lack of evaluation did not ensure spent fuel cladding temperatures during canister processing operations would remain less than Spent Fuel Storage and Transportation Interim Staff Guidance-11, "Cladding Considerations for the Transportation and Storage of Spent Fuel," safety limits. The inspectors determined that that the violation could be evaluated using Section 6.5.d.2 of the NRC Enforcement Policy, as a Severity Level IV violation, in that the licensee failed to establish, maintain, or implement adequate controls to ensure that the replacement of the port cap was performed under conditions bounded by a thermal analysis that ensured the integrity of the fuel would be maintained during the repair. Because the finding is associated only with traditional enforcement, there is not an associated cross-cutting aspect. (Section 4OA5)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance with an associated NCV of Technical Specification (TS) 5.4.1 for the failure to implement procedures to perform preventative maintenance in vaults containing safety related cabling subject to water intrusion. Specifically, licensee personnel failed to ensure the cables were not wetted as required by PMI-5053, “Cable Management Program.” Cables were on the ground in the vaults exposing the cables to periodic wetting, which will degrade the cable insulation. On August 27, 2012, the inspectors noted that cables in one vault were on the ground and the vault showed evidence of periodic wetting of the cables. For corrective action, the licensee is performing an apparent cause evaluation; inspecting all cable vaults that have had safety-related cabling elevated since February 2010; and raising and re-securing cabling in vaults subject to water intrusion. This issue was entered into the licensee’s corrective action program as AR 2012-10680.

  This finding affected the Mitigating Events Cornerstone and was more than minor because the issue could become a more significant safety concern if left uncorrected. Specifically, failure to properly perform preventative maintenance in vaults containing safety-related cables subjected to water intrusion resulted in periodic wetting of cables. Wetting of cables has led to degradation of cable insulation at nuclear facilities. The inspectors used IMC 0609, “Significance Determination Process,” Attachment 4, “Initial Characterization of Findings,” which directed the inspectors to Exhibit 2, “Mitigating Systems Screening Questions,” of Appendix A, to determine significance. This finding was of very low safety significance (Green) because the finding constituted a design or qualification deficiency but did not result in a loss of system safety function. This finding is associated with a cross-cutting aspect in the work control component of the human performance cross-cutting area. Specifically, engineering did not appropriately plan for maintenance personnel to assist with lifting the floor grating to ensure visual inspections were adequately performed (H.3 (a)). (1R06)

B. Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at or near full power until July 19 when the licensee commenced a Technical Specification (TS) shutdown due to an inoperable Engineered Safety Feature Actuation System (ESFAS). The licensee stopped the down power at 50 percent after the NRC granted a Notice of Enforcement Discretion. After restoring the ESFAS to operable, the licensee raised power to 100 percent. Unit 1 operated at or near 100 percent for the remainder of the inspection period.

Unit 2 operated at or near full power for the entire inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors performed a detailed review of the licensee's procedures and preparations for operating the facility during the week of July 2, 2012 - July 6, 2012, when ambient outside temperature was high and the ultimate heat sink was experiencing elevated temperatures. The inspectors focused on plant specific design features and implementation of the procedures for responding to or mitigating the effects of these conditions on the operation of the facility's containment cooling system, component cooling water system and essential service water system. Inspection activities included a review of the licensee's adverse weather procedures, daily monitoring of the off-normal environmental conditions, and that operator actions specified by plant specific procedures were appropriate to ensure operability of the facility's normal and emergency cooling systems.

This inspection constituted one readiness for impending adverse weather condition sample as defined in Inspection Procedure (IP) 71111.01-05.

b. Findings

No findings were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

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

- Unit 1AB emergency diesel generator;
- Unit 1 turbine driven auxiliary feed water system; and
- Unit 2 east essential service water system.

The inspectors selected these systems based on their risk significance relative to the Reactor Safety Cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could impact the function of the system and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, Updated Final Safety Analysis Report (UFSAR), TS requirements, outstanding work orders (WOs), condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions 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 and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program (CAP) with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

These activities constituted three partial system walkdown samples as defined in IP 71111.04-05.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors conducted fire protection walkdowns which were focused on availability, accessibility, and the condition of firefighting equipment in the following risk-significant plant areas:

- Fire Zone 1A and 1B, Unit 1 east and west containment spray pump rooms;
- Fire Zone 6N, Unit 1 north auxiliary building;
- Fire Zone 17F and 17G, Unit 2 turbine driven and east motor driven auxiliary feed pump rooms; and
- Fire Zone 55, Unit 2 CD battery room.

The inspectors reviewed areas to assess if the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capability, maintained passive fire protection features in good material condition, and implemented adequate compensatory measures for out-of-service, degraded or inoperable fire protection equipment, systems, or features in accordance with the licensee's fire plan.
The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant's Individual Plant Examination of External Events with later additional insights, their potential to impact equipment which could initiate or mitigate a plant transient, or their impact on the plant's ability to respond to a security event. Using the documents listed in the Attachment to this report, the inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed; that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor

issues identified during the inspection were entered into the licensee's CAP. Documents reviewed are listed in the Attachment to this report.

These activities constituted four quarterly fire protection inspection samples as defined in IP 71111.05-05.

b. Findings

No findings were identified.

1R06 Flooding (71111.06)

.1 Internal Flooding

a. Inspection Scope

The inspectors reviewed selected risk important plant design features and licensee procedures intended to protect the plant and its safety-related equipment from internal flooding events. The inspectors reviewed flood analyses and design documents, including the UFSAR, engineering calculations, and abnormal operating procedures to identify licensee commitments. The specific documents reviewed are listed in the Attachment to this report. In addition, the inspectors reviewed licensee drawings to identify areas and equipment that may be affected by internal flooding caused by the failure or misalignment of nearby sources of water, such as the fire suppression or the circulating water systems. The inspectors also reviewed the licensee's corrective action documents with respect to past flood related items identified in the corrective action program to verify the adequacy of the corrective actions. The inspectors performed a walkdown of the following plant area to assess the adequacy of watertight doors and verify drains and sumps were clear of debris and were operable, and that the licensee complied with its commitments:

- Unit 1 and Unit 2 safety related cable vaults.

Specific documents reviewed during this inspection are listed in the Attachment to this report. This inspection constituted one internal flooding sample as defined in IP 71111.06-05.

b. Findings

Introduction

The inspectors identified a finding of very low safety significance (Green) with an associated NCV of TS 5.4.1 for the failure to implement procedures as recommended by Regulatory guide 1.33. Specifically, licensee personnel failed to properly inspect vaults containing safety-related cabling to ensure cables were not wetted.

Description

On August 27, 2012, while performing a flooding walk down inspection, the inspectors identified that several vaults containing safety-related cabling in the auxiliary building had cables either sagging towards the floor or laying on the floor. In addition, the inspectors noted some water on the vault floors. The NRC previously issued non-cited violation (NCV) 05000315/2010002-02 to address a similar condition. In response to the 05000315/2010002-02 NCV, the licensee installed concrete blocks in cable vaults to limit cable wetting, which can degrade cables. However, the inspectors noted that the licensee actions had not been effective at protecting the cables from periodic wetting. The inspectors communicated their observations to the licensee and the licensee entered the condition into the corrective action program as Action Request (AR) 2012-10680. The inspectors noted that there has been a long history of water intrusion into cable vaults throughout the plant. In Issue Report (IR) 05000315/2010002, the inspectors documented that the licensee had ARs dating back to 2000 that reported water intrusion into cable vaults. In addition to modifying the design to support cables off the ground in the vaults, the licensee also required periodic visual inspections of each vault to ensure the cables remained dry. The inspectors reviewed PMI-5053, "Cable Management Program Description," the procedure for performing visual inspections, and concluded that the licensee failed to identify wetted cables per step 4.2. Discussions with engineering revealed that visual inspections did not include lifting the floor grating and instead only utilized the 3/4 inch view holes in the grating. Based on additional interviews with engineering personnel and independent visual inspections, the inspectors concluded that the inspection performed by the licensee was not adequate to satisfy the procedural requirement because the method used did not provide sufficient visibility to detect that the cables were subjected to periodic wetting.

Analysis

The inspectors determined that failure to implement procedure PMI-5053 was a performance deficiency that warranted a significance evaluation in accordance with the Significance Determination Process (SDP). The inspectors reviewed the samples of minor issues in Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix E, "Examples of Minor Issues," issued on August 11, 2009, and determined that there were no examples related to this issue. Consistent with the guidance in IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," issued on September 7, 2012, the inspectors determined that this issue could become a more significant safety concern if left uncorrected and was therefore more than minor. Specifically, failure to implement the procedure for performing visual inspections of vaults containing safety-related cables subject to water intrusion allowed cables to remain on the ground where the cables were subjected to periodic wetting, which could result in subsequent degradation.

Because this issue involved safety-related cabling which could affect systems required for safe shutdown, the inspectors concluded that this finding was associated with the Mitigating System Cornerstone. The inspectors performed an SDP review using the guidance provided in IMC 0609, Attachment 4, "Initial Characterization of Findings," issued on June 19, 2012, which directed the inspectors to Appendix A, "The SDP for Findings At-Power," issued on September 19, 2012. Using Appendix A, Exhibit 2, "Mitigating System Screening Questions," the inspectors determined that this finding screened as Green, very low safety significance, because the finding constitutes a design or qualification deficiency but did not result in a loss of operability of functionality.

The inspectors concluded that this finding was associated with a cross-cutting aspect in the work control component of the human performance cross-cutting area. Specifically, engineering did not appropriately plan for maintenance personnel to assist with lifting the floor grating to ensure visual inspections were adequately performed. (H.3 (a)).

Enforcement

Technical Specification 5.4.1.a requires, in part, that written procedures be established, implemented, and maintained for the activities recommended in Regulatory Guide 1.33, Revision 2, Appendix A. Regulatory Guide 1.33, Appendix A, Section 9, Procedures for Performing Maintenance, states, in part, that maintenance that can affect the performance of safety related equipment should be properly pre-planned and performed in accordance with written procedures. Contrary to the above, between February 8, 2010, and August 27, 2012, licensee personnel failed to implement maintenance procedure PMI-5053, "Cable Management Program Description," Revision 1, a procedure required by Regulatory Guide 1.33, to visually inspect cable vaults containing safety-related cabling that were subject to water intrusion. Specifically, the visual inspections as performed did not include lifting the vault covers to ensure safety-related cabling was not wetted. Consequently, cabling exposed to a wetted environment remained undetected. Continued exposure of cabling to a wetted environment could result in subsequent cable degradation. For corrective action, the licensee began inspections of additional vaults and entered the condition into the CAP. Because this violation was of very low safety significance, was not repetitive or willful, and was entered into the licensee's CAP as AR 2012-10680, this violation is being treated as an NCV consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000315/2012004-01, 05000316/2012-004-01, Failure to Properly Preplan and Perform Maintenance on Safety-related Equipment).

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review of Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

On August 28, 2012, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification training to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of abnormal and emergency procedures;
- control board manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications.

The crew's performance in these areas was compared to pre-established operator action expectations and successful critical task completion requirements. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly licensed operator requalification program simulator sample as defined in IP 71111.11.

b. Findings

No findings were identified.

.2 Resident Inspector Quarterly Observation of Heightened Activity or Risk (71111.11Q)

a. Inspection Scope

On September 18, 2012, the inspectors observed a failed nuclear power range instrument power supply replacement in Unit 1. This was an activity that required heightened awareness or was related to increased risk. The inspectors evaluated the following areas:

- licensed operator performance;
- maintenance worker performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of procedures;
- control board manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions.

The performance in these areas was compared to pre-established operator action expectations, procedural compliance and task completion requirements. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly licensed operator heightened activity/risk sample as defined in IP 71111.11.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following risk-significant systems:

- Units 1 and Unit 2 post accident hydrogen monitoring systems.

The inspectors reviewed events such as where ineffective equipment maintenance had resulted in valid or invalid automatic actuations of engineered safeguards systems and independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- implementing appropriate work practices;
- identifying and addressing common cause failures;
- scoping of systems in accordance with 10 CFR 50.65(b) of the maintenance rule;

- characterizing system reliability issues for performance;
- charging unavailability for performance;
- trending key parameters for condition monitoring;
- ensuring 10 CFR 50.65(a)(1) or (a)(2) classification or re-classification; and
- verifying appropriate performance criteria for structures, systems, and components/functions classified as (a)(2), or appropriate and adequate goals and corrective actions for systems classified as (a)(1).

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly maintenance effectiveness sample as defined in IP 71111.12-05.

b. Findings

No findings were identified.

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

a. Inspection Scope

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

- emergent supplemental diesel outage; and
- emergent essential service water pipe repair.

These activities were selected based on their potential risk significance relative to the Reactor Safety Cornerstones. As applicable for each activity, the inspectors verified that risk assessments were performed as required by 10 CFR 50.65(a)(4) and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's probabilistic risk analyst or shift technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed TS requirements and walked down portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met.

Specific documents reviewed during this inspection are listed in the Attachment to this report. These maintenance risk assessments and emergent work control activities constituted two samples as defined in IP 71111.13-05.

b. Findings

No findings were identified.

1R15 Operability Determinations and Functional Assessments (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- past operability of power operated relief valves due to improperly sized pinions;
- large break loss of coolant accident thermal conductivity degradation evaluation;
- Part 21 emergency diesel generator 20 pound air regulator evaluation;
- Unit 2 west essential service water through-wall pipe leak; and
- steam generator power operated relief valves due to pneumatic design.

The inspectors selected these potential operability issues based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that TS operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the TS and UFSAR to the licensee's evaluations to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. Additionally, the inspectors reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the Attachment to this report.

This operability inspection constituted five samples as defined in IP 71111.15-05.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

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

- Unit 1 CD emergency diesel generator testing following maintenance overhaul;
- Unit 2 west motor driven auxiliary feedwater motor operated valve and preventative maintenance overhaul;
- Unit 1 east residual heat removal system maintenance overhaul;
- Unit 1, IMO-911, refueling water storage tank suction to charging valve repair;
- Unit 2 west essential service water pipe through-wall leak weld repair; and
- Unit 1 steam generator stop valve dump valve testing following fuse replacement.

These activities were selected based upon the structure, system, or component's ability to impact risk. The inspectors evaluated these activities for the following (as applicable):

the effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed; acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate; tests were performed as written in accordance with properly reviewed and approved procedures; equipment was returned to its operational status following testing (temporary modifications or jumpers required for test performance were properly removed after test completion); and test documentation was properly evaluated. The inspectors evaluated the activities against TSs, the UFSAR, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with post-maintenance tests to determine whether the licensee was identifying problems and entering them in the CAP and that the problems were being corrected commensurate with their importance to safety. Documents reviewed are listed in the Attachment to this report.

This inspection constituted six post-maintenance testing samples as defined in IP 71111.19-05.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

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

- Unit 1 4KV loss of voltage and degraded voltage surveillance (routine);
- Unit 2 east motor driven auxiliary feed water pump test (inservice);
- Unit 1 and Unit 2 alternate oxygen monitor surveillance test (routine); and
- Unit 2 nuclear instrumentation power range channel operation (routine).

The inspectors observed in-plant activities and reviewed procedures and associated records to determine the following:

- did preconditioning occur;
- were the effects of the testing adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- were acceptance criteria clearly stated, demonstrated operational readiness, and consistent with the system design basis;
- plant equipment calibration was correct, accurate, and properly documented;
- as-left setpoints were within required ranges; and the calibration frequency was in accordance with TSs, the USAR, procedures, and applicable commitments;
- measuring and test equipment calibration was current;
- test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied;

- test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used;
- test data and results were accurate, complete, within limits, and valid;
- test equipment was removed after testing;
- where applicable for inservice testing activities, testing was performed in accordance with the applicable version of Section XI, American Society of Mechanical Engineers (ASME) code, and reference values were consistent with the system design basis;
- where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable;
- where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure;
- where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished;
- prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test;
- equipment was returned to a position or status required to support the performance of its safety functions; and
- all problems identified during the testing were appropriately documented and dispositioned in the CAP.

Documents reviewed are listed in the Attachment to this report.

This inspection constituted three routine surveillance testing samples and one inservice testing sample as defined in IP 71111.22, Sections -02 and -05.

b. Findings

No findings were identified.

2. RADIATION SAFETY

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

The inspection activities supplement those documented in IR 05000315/2012002; 05000316/2012002 and constitute one complete sample as defined in IP 71124.01-05.

.1 Radiological Hazard Assessment (02.02)

a. Inspection Scope

The inspectors observed work in potential airborne areas and evaluated whether the air samples were representative of the breathing air zone. The inspectors evaluated whether continuous air monitors were located in areas with low background to minimize false alarms and were representative of actual work areas. The inspectors evaluated the licensee's program for monitoring levels of loose surface contamination in areas of the plant with the potential for the contamination to become airborne.

b. Findings

No findings were identified.

.2 Instructions to Workers (02.03)

a. Inspection Scope

The inspectors reviewed selected occurrences where a worker's electronic personal dosimeter noticeably malfunctioned or alarmed. The inspectors evaluated whether workers responded appropriately to the off-normal condition. The inspectors assessed whether the issue was included in the CAP and dose evaluations were conducted as appropriate.

b. Findings

No findings were identified.

.3 Contamination and Radioactive Material Control (02.04)

a. Inspection Scope

The inspectors selected several sealed sources from the licensee's inventory records and assessed whether the sources were accounted for and verified to be intact.

The inspectors evaluated whether any transactions, since the last inspection, involving nationally tracked sources were reported in accordance with 10 CFR 20.2207.

b. Findings

No findings were identified.

.4 Radiological Hazards Control and Work Coverage (02.05)

a. Inspection Scope

The inspectors examined the licensee's physical and programmatic controls for highly activated or contaminated materials (nonfuel) stored within spent fuel and other storage pools. The inspectors assessed whether appropriate controls (i.e., administrative and physical controls) were in place to preclude inadvertent removal of these materials from the pool.

The inspectors examined the posting and physical controls for selected high radiation areas and very high radiation areas to verify conformance with the occupational performance indicator.

b. Findings

No findings were identified.

.5 Risk-Significant High Radiation Area and Very High Radiation Area Controls (02.06)

a. Inspection Scope

The inspectors discussed the controls in place for special areas that have the potential to become very high radiation areas during certain plant operations with first-line health physics supervisors, (or equivalent positions having backshift health physics oversight authority). The inspectors assessed whether these plant operations require communication beforehand with the health physics group, so as to allow corresponding timely actions to properly post, control, and monitor the radiation hazards including re-access authorization.

b. Findings

No findings were identified.

.6 Radiation Worker Performance (02.07)

a. Inspection Scope

The inspectors reviewed radiological problem reports since the last inspection that found the cause of the event to be human performance errors. The inspectors evaluated whether there was an observable pattern traceable to a similar cause. The inspectors assessed whether this perspective matched the corrective action approach taken by the licensee to resolve the reported problems. The inspectors discussed with the Radiation Protection Manager any problems with the corrective actions planned or taken.

b. Findings

No findings were identified.

.7 Radiation Protection Technician Proficiency (02.08)

a. Inspection Scope

The inspectors reviewed radiological problem reports since the last inspection that found the cause of the event to be radiation protection technician error. The inspectors evaluated whether there was an observable pattern traceable to a similar cause. The inspectors assessed whether this perspective matched the corrective action approach taken by the licensee to resolve the reported problems.

b. Findings

No findings were identified.

.8 Problem Identification and Resolution (02.09)

a. Inspection Scope

The inspectors evaluated whether problems associated with radiation monitoring and exposure control were being identified by the licensee at an appropriate threshold and were properly addressed for resolution in the licensee's CAP. The inspectors assessed the appropriateness of the corrective actions for a selected sample of problems documented by the licensee that involve radiation monitoring and exposure controls. The inspectors assessed the licensee's process for applying operating experience to their plant.

b. Findings

No findings were identified.

2RS2 Occupational As-Low-As-Is-Reasonably-Achievable Planning and Controls (71124.02)

The inspection activities supplement those documented in IR 05000315/2012002; 05000316/2012002 and constitute a partial sample as defined in IP 71124.02-05.

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed pertinent information regarding plant collective exposure history, current exposure trends, and ongoing or planned activities in order to assess current performance and exposure challenges. The inspectors reviewed the plant's 3-year rolling average collective exposure.

The inspectors reviewed the site-specific trends in collective exposures (using NUREG-0713, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities," and plant historical data) and source term (average contact dose rate with reactor coolant piping) measurements (using Electric Power Research Institute TR-108737, "BWR Iron Control Monitoring Interim Report," issued December 1998, and/or plant historical data, when available).

b. Findings

No findings were identified.

.2 Problem Identification and Resolution (02.06)

a. Inspection Scope

The inspectors evaluated whether problems associated with as-low-as-is-reasonably-achievable (ALARA) planning and controls are being identified by the licensee at an appropriate threshold and were properly addressed for resolution in the licensee's CAP.

b. Findings

No findings were identified.

3. OTHER ACTIVITIES

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

4OA1 Performance Indicator Verification (71151)

.1 Reactor Coolant System Specific Activity

a. Inspection Scope

The inspectors sampled licensee submittals for the Reactor Coolant System (RCS) Specific Activity performance indicator (PI) for D.C. Cook Nuclear Power Plant, Units 1 and 2, for the period from the second quarter 2011 through the second quarter 2012. The inspectors used PI definitions and guidance contained in the Nuclear Energy Institute (NEI) Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, dated October 2009, to determine the accuracy of the PI data reported during those periods. The inspectors reviewed the licensee's RCS chemistry samples, TS requirements, issue reports, event reports, and NRC Integrated Inspection Reports to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. In addition to record reviews, the inspectors observed a chemistry technician obtain and analyze a RCS sample. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two RCS Specific Activity samples as defined in IP 71151-05.

b. Findings

No findings were identified.

.2 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors sampled licensee submittals for the Occupational Radiological Occurrences PI for the period from the second quarter 2011 through the second quarter 2012. The inspectors used PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, dated October 2009, to determine the accuracy of the PI data reported during those periods. The inspectors reviewed the licensee's assessment of the PI for occupational radiation safety to determine if indicator related data was adequately assessed and reported. In assessing the adequacy of the licensee's PI data collection and analyses, the inspectors discussed with radiation protection staff, the scope, and breadth of its data review and the results of those reviews. The inspectors independently reviewed electronic personal dosimetry dose rate, accumulated dose alarms, dose reports, and the dose assignments for any intakes that occurred during the time period reviewed to determine if there were potentially unrecognized occurrences. The inspectors also conducted walkdowns of numerous locked high and very high radiation areas entrances to determine the adequacy of the controls in place for these areas. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one Occupational Exposure Control Effectiveness sample as defined in IP 71151-05.

b. Findings

No findings were identified.

.3 Radiological Effluent Technical Specification/Offsite Dose Calculation Manual Radiological Effluent Occurrences

a. Inspection Scope

The inspectors sampled licensee submittals for the Radiological Effluent Technical Specification (RETS)/Offsite Dose Calculation Manual (ODCM) Radiological Effluent Occurrences PI for the period from the second quarter 2011 through the second quarter 2012. The inspectors used PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, dated October 2009, to determine the accuracy of the PI data reported during those periods. The inspectors reviewed the licensee's issue report database and selected individual reports generated since this indicator was last reviewed to identify any potential occurrences such as unmonitored, uncontrolled, or improperly calculated effluent releases that may have impacted offsite dose. The inspectors reviewed gaseous effluent summary data and the results of associated offsite dose calculations for selected dates to determine if indicator results were accurately reported. The inspectors also reviewed the licensee's methods for quantifying gaseous and liquid effluents and determining effluent dose. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one RETS/ODCM Radiological Effluent Occurrences sample as defined in IP 71151 05.

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems (71152)

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Public Radiation Safety, Occupational Radiation Safety, and Physical Protection

.1 Routine Review of Items Entered into the Corrective Action Program

a. Inspection Scope

As part of the various baseline inspection procedures 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 CAP at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Attributes reviewed included: identification of the problem was complete and accurate; timeliness was commensurate with the safety significance; evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent-of-condition reviews, and previous occurrences reviews were proper and adequate; and that the classification, prioritization, focus, and timeliness of corrective actions were commensurate with safety and sufficient to prevent recurrence of the issue. Minor issues entered into the licensee's CAP as a result of the inspectors' observations are included in the Attachment to this report.

These routine reviews for the identification and resolution of problems did not constitute any additional inspection samples. Instead, by procedure they were considered an integral part of the inspections performed during the quarter and documented in Section 1 of this report.

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's CAP. This review was accomplished through inspection of the station's daily condition report packages.

These daily reviews were performed by procedure as part of the inspectors' daily plant status monitoring activities and, as such, did not constitute any separate inspection samples.

b. Findings

No findings were identified.

.3 Selected Issue Follow-Up Inspection: Feedwater Reliability Apparent Cause Evaluation

a. Inspection Scope

The inspectors selected the following equipment apparent cause evaluation for an in-depth review:

- AR 2011-15022-01, "Feedwater Reliability Apparent Cause Evaluation"

The inspectors discussed the evaluation and associated corrective actions with licensee personnel and verified the following attributes while reviewing the apparent cause evaluation:

- Complete and accurate problem identification in a timely manner commensurate with its safety significance and ease of discovery; extent of condition, generic implications, common cause and previous occurrences were considered;
- problem resolution was classified and prioritized commensurate with safety significance;

- apparent and contributing causes were identified; and
- corrective actions were appropriately focused.

This review constituted one in-depth problem identification and resolution sample as defined in IP 71152-05.

b. Findings

No findings were identified.

.4 Selected Issue Follow-up Inspection: Station Response to Wetted Cables in the Unit 1AB Diesel Generator Motor Control Center Pit Root Cause Evaluation.

a. Inspection Scope

The inspectors selected the following equipment root cause evaluation for an in-depth review:

- AR 2010-2558, “Station Response to Wetted Cables in the Unit 1AB Diesel Generator Motor Control Center Pit Root Cause Evaluation.”

The inspectors discussed the evaluation and associated corrective actions with licensee personnel and verified the following attributes while reviewing the root cause evaluation:

- complete and accurate problem identification in a timely manner commensurate with its safety significance and ease of discovery;
- extent of condition, generic implications, common cause and previous occurrences were considered;
- problem resolution was classified and prioritized commensurate with safety significance;
- root and contributing causes were identified; and
- corrective actions were appropriately focused.

This review constituted one in-depth problem identification and resolution sample as defined in IP 71152-05.

b. Findings

No findings were identified.

4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153)

.1 Unresolved Item: Follow-up Inspection of Actions from Notice Of Enforcement Discretion 12-3-002

a. On July 19, 2012, at 7:34 a.m. Eastern Daylight Time (EDT), the fuses for two Unit 1 steam generator stop valves (specifically the associated dump valves) blew. Without power, the valves would not open to cause a closure of the associated steam generator stop valves. The failure affected the Train B portion; Train A remained operable. The blown fuses rendered one train of ESFAS inoperable and the licensee entered TS 3.3.2 Condition C which requires restoration to operable within 6 hours. If the completion time cannot be met, TS 3.3.2 condition I required the licensee to enter Mode 3 within 6 hours and Mode 4 within 12 hours. Since the licensee could not complete repairs within 6 hours, the licensee began a down power and requested that the NRC exercise discretion and extend the required action time to 30 hours to enter Mode 3 and 36 hours to enter Mode 4. The licensee verbally requested the discretion via telephone and the NRC granted discretion, effective at 7:34 p.m. EDT (Notice of Enforcement Discretion (NOED) 12-3-002, ML12207A516). During the telephone call, the licensee informed the NRC that repairs would be completed within 24 hours and the risk associated with discretion was low. The licensee also proposed a number of compensatory measures to mitigate the risk associated with operating during the period of discretion. The licensee completed repairs and exited the limiting condition for operation at 8:30 p.m. EDT the same day.

b. Inspection Scope

The inspectors responded to the control room after being informed of the inoperable steam generator stop valves. Initially, the licensee entered TS 3.0.3 due to an erroneous conclusion that the failure of the solenoids rendered two steam generator stop valves inoperable. Subsequently, the licensee determined that a TS 3.0.3 entry was not required and that the correct TS was TS 3.3.2, ESFAS Instrumentation. The licensee then entered the correct TS condition based on the time when the fuse blew. The inspectors reviewed the licensee's actions and concluded that the delay in recognizing the correct TS did not result in the licensee failing to perform a required action within the specified completion time. The inspectors observed the licensee's actions in the control room to understand the condition and evaluate the cause. Over the course of the day, the inspectors monitored the licensee's troubleshooting activities and efforts to correct the condition without requiring enforcement discretion. At 3:39 p.m. EDT, the licensee commenced a down power to support compliance with TS 3.3.2 condition I. The licensee identified that a short occurred between the associated power distribution panel and a solenoid. Because repairs could not be completed before the TS 3.3.2 completion time expired, the licensee requested, and the NRC granted, an NOED. The licensee stopped the shutdown at 6:09 p.m. EDT after the NRC granted the NOED. During the period of discretion, the inspectors verified through plant walk downs, observation, document reviews, and discussions with operations and engineering personnel, that the licensee appropriately implemented the compensatory actions as approved. At 8:30 p.m. EDT, the licensee restored the system to operable and exited the NOED.

The inspectors reviewed the licensee's written NOED to validate that the information was consistent with information provided by the licensee during the NOED telephone call. The licensee's cause evaluation was not complete at the end of the inspection period. Therefore, the inspectors will review the cause analysis and corrective actions after the evaluation is completed. This issue will be an unresolved item (URI) until licensee completion and NRC review of causal analysis and follow-up actions for the issue (URI 05000315/2012004-02, Follow-up Inspection of Actions from NOED 12-3-002).

Documents reviewed in this inspection are listed in the Attachment to this report.

This event follow-up review constituted one sample as defined in IP 71153-05.

c. Findings

No findings were identified.

4OA5 Other Activities

.1 Preoperational Testing of an Independent Spent Fuel Storage Facility Installation at Operating Plants (60854.1)

a. Inspection Scope

(1) Training

The inspectors reviewed the licensee's Independent ISFSI training program, which consisted of classroom and on-the-job training to ensure involved staff was adequately trained for the job they were responsible to perform. The inspectors also reviewed training records and qualifications of individuals performing work activities associated with the ISFSI. The inspectors interviewed licensee personnel in various departments to verify that they were knowledgeable in the scope of work that was being performed.

(2) Quality Assurance

The inspectors reviewed the licensee's Quality Assurance Program, as it applied to the ISFSI. In a letter from Indiana Michigan Power to the NRC on May 21, 2010, the D.C. Cook Nuclear Plant communicated their intent to incorporate the ISFSI Quality Assurance Program into their established Title 10 of the CFR Part 50 Quality Assurance Program as allowed by 10 CFR 72.140(d), "Quality Assurance."

The inspectors reviewed procedures pertaining to the receipt inspection of MPC and HI-STORM storage casks. The inspectors observed the licensee implement their Materials and Test Equipment program into ISFSI activities. The inspectors observed that gauges were within their calibration date, and that the use of 99.995 percent pure helium was used during backfilling. The inspectors reviewed the calibration dates of various components used for ISFSI operations.

(3) Emergency Preparedness and Fire Protection

The inspectors reviewed the licensee's Emergency Preparedness Plan required by 10 CFR 50.47, "Emergency Plans," for conformance with 10 CFR 72.32(c), "Emergency Plans." The inspectors verified that the licensee incorporated Emergency Action Levels into the Emergency Plan to address the emergency scenarios, their classification, and recovery actions associated with the ISFSI. The inspectors reviewed the licensee's procedure that addressed contingency actions, including a fire at the ISFSI.

(4) Fuel Selection

The inspectors reviewed the licensee's calculations associated with fuel characterization and selection for storage. The inspectors reviewed the initial campaign cask fuel selection packages to verify that the licensee planned to load fuel in accordance with the Certificate of Compliance approved contents.

The inspectors reviewed the licensee's evaluations that characterize fuel as fuel debris, damaged, or intact. The licensee did not plan to load any damaged fuel assemblies, fuel assemblies with pinhole leaks or hair line cracks, or fuel debris during the initial campaign.

The licensee is planning to load array/class 15x15B Unit 1 fuel and 17x17A Unit 2 fuel. The inspectors reviewed the licensee's selection of fuel spacers required for these particular fuel assemblies.

The inspectors reviewed the selected assemblies to identify the maximum burnup and noted that the maximum burnup of any assembly planned for loading during the initial campaign was less than 45,000 megawatt days/metric tons of uranium. Fuel assemblies with burnup greater than 45,000 megawatt days/metric tons of uranium limit are considered high burnup assemblies, and additional requirements exist. As such the licensee planned to only load moderate burnup assemblies during the 2012 campaign.

The inspectors reviewed the licensee's use of NRC Regulatory Guide 3.54, "Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation," that was used to calculate decay heat loads for each assembly. The licensee assessed heat loads for both the total canister as well as individual canister assembly locations. The inspectors verified that the licensee incorporated appropriate cooling times, initial uranium enrichments, and burnup limiting acceptance criteria into their calculations to ensure the radiation dose limits from the ISFSI were in compliance with 10 CFR 72.104, "Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI or MRS."

(5) Radiation Protection

The inspectors evaluated the licensee's Radiation Protection Program pertaining to the operation of the ISFSI and maintaining exposures ALARA. The inspectors reviewed the licensee's procedures describing the methods and techniques used when performing dose rate and surface contamination surveys and verified that they ensured dose rate limits and surveillance requirements of the TSs were met. The inspectors verified that the licensee's Radiation Protection staff considered lessons learned from other utilities' spent fuel loading campaigns during development of the radiological controls for the loading, storage, and unloading operations. The inspectors interviewed licensee personnel to verify their knowledge regarding the scope of the work and the radiological hazards associated with the transfer and storage of spent fuel. The inspectors reviewed licensee dose rate calculations to verify that the licensee's ISFSI was in compliance with 10 CFR 72.104, "Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI or MRS." The inspectors verified that the licensee has a radiation monitoring program in place to ensure compliance with 10 CFR 20.1301 "Dose Limits for Individual Members of the Public" and interviewed staff on the implementation of this program in regards to ISFSI storage operations.

(6) Control of Heavy Loads

The inspectors reviewed the licensee's implementation of the crane and control of heavy loads program for ISFSI operations. The inspectors reviewed inspection, testing, and maintenance documentation associated with the Auxiliary Building crane, HI-TRAC (transfer cask) lifting trunnions, lift yoke, and Vertical Cask Transporter to ensure compliance with industry standards, station procedures, and design specifications. The inspectors observed the licensee perform heavy loads movements inside and outside of the Auxiliary Building.

(7) Dry Run Activities

The licensee performed pre-operational dry run activities in order to fulfill the requirements of the Certificate of Compliance (CoC). The NRC inspectors were onsite to observe dry run activities: April 2 through April 5, 2012; April 23 through April 27, 2012; April 30 through May 5, 2012; July 10 through July 13, 2012; and July 25 through July 28, 2012. These activities included MPC welding and processing; heavy loads operations inside and outside of the Auxiliary Building; reviews of the licensee's 10 CFR 72.212, "Condition of General License Issued under 72.210," report; and other documentation reviews.

The inspectors observed the licensee place the HI-TRAC containing the MPC into the spent fuel pool (SFP) and the subsequent loading and unloading of dummy fuel assemblies into the MPC basket. The licensee demonstrated removal of a dummy fuel assembly from the SFP storage rack, placement of the assembly into the MPC, and retrieval of the fuel assembly from the MPC to the SFP rack. The inspectors observed the licensee remove a HI-TRAC containing a MPC from the SFP and subsequent placement of the HI-TRAC in the processing lay down area.

The inspectors observed the licensee demonstrate all closure welds and non-destructive testing required to prepare the MPC for storage. Welding procedures, procedure qualification reports, and welder qualifications were reviewed to ensure compliance with the ASMEs Boiler and Pressure Vessel code Section IX requirements. The non-destructive testing procedures and the qualifications of the technician performing the testing were reviewed by the inspectors to ensure compliance with ASMEs Boiler and Pressure Vessel Section V requirements.

The inspectors observed the licensee demonstrate MPC processing activities including MPC hydrostatic testing, blow-down, forced helium dehydration, and helium backfilling. The inspectors also observed the licensee demonstrate MPC unloading dry run activities.

The inspectors observed transfer of the MPC from the HI-TRAC transfer cask to the HI-STORM storage cask in a restrained support structure in the Auxiliary Building and the subsequent movement of the HI-STORM outside of the Auxiliary Building on a Goldhofer transport vehicle. The inspectors observed transfer of the HI-STORM overpack from the Auxiliary Building to the ISFSI pad via the Goldhofer where a Vertical Cask Transporter lifted the HI-STORM off the Goldhofer load deck and placed the HI-STORM in its proper location on the ISFSI pad.

The inspectors observed communication between dry cask personnel, operations, radiation protection, and security staff. The inspectors verified adequate communication and coordination between departments and adherence to procedures.

The inspectors attended licensee briefings during dry run operations including: infrequently performed test or evolution briefings, pre-job briefs, post-job briefs, ALARA radiation dose briefs, and in-field briefs.

The inspectors reviewed loading and unloading procedures to ensure that they contained commitments and requirements specified in the CoC, TS, UFSAR, and 10 CFR Part 72.

b. Findings

Unresolved Item: Design Basis of Seismic Category I Structures and Equipment

Introduction

A URI was identified by the inspectors regarding the licensee's design and licensing basis for structural steel allowable design stresses for seismic Category 1 structures.

Description

The plant UFSAR states that the auxiliary building seismic Category 1 structure was designed to the "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings"; adopted in 1963 by the American Institute of Steel Construction (AISC). In various sections of the UFSAR, certain structures have allowable stresses that are specified for both the Operating Basis Earthquake (OBE) and the Safe Shutdown Earthquake (SSE). Where specified, these allowable stresses are not increased above normal allowable stresses for the OBE load condition. However, in discussion of the auxiliary building requirements, the UFSAR simply states "for the OBE, all stresses in the steel superstructure are within allowable as specified by the 1963 code… (AISC)." This AISC specification defined the structural steel allowable design stresses for normal load cases and, under a separate subsection, allowed a one third increase over normal stresses for seismic and wind loads. The AISC specification is used as a design code in the building industry where structures are not designed to OBE or SSE loads, but to earthquake levels specified in the building codes. The licensee maintained that the one third increase over normal stresses was acceptable for use in design under the OBE load conditions for the auxiliary building even though the one third increase was not used in the OBE analysis in other sections of the UFSAR. The NRC Safety Evaluation Report, dated September 10, 1973, stated that the acceptance criteria were consistent with the other plants licensed at the time. The inspectors cannot determine if the NRC ever accepted the use of the additional margin, for the OBE load condition, at D.C. Cook. The inspectors are not aware of any other plants that have a Safety Evaluation Report documenting approval to utilize the one third increase. D.C. Cook is a pre-standard review plan (NUREG-0800) plant; however, the inspectors have noted for context, that NUREG-0800 as well as the AISC "Specification for Safety-related Steel Structures for Nuclear Facilities," N690, has always specified use of normal allowable stresses without the one third increase.

The licensee applied the one third increase to new evaluations of the auxiliary building crane and the auxiliary building structure to support a crane up-rate from 60 tons single failure proof to 145 tons single failure proof. This modification was performed according to Title 10 of the CFR Part 50.59, "Changes, Tests, and Experiments," and evaluated per Engineering Change 0000049518, "Auxiliary Building East Crane Uprate and Upgrade," Revision 0. The auxiliary building crane and the auxiliary building structure are both seismic Category 1 per their UFSAR. The licensee maintained that use of the one third increase for the OBE load case was part of their original licensing basis; however, the one third increase was not used in the original evaluations.

For the SSE load condition, the inspectors noted that the calculated stresses in both the crane and the auxiliary building structure were less than the allowable stresses and the SSE load condition was consistent with the licensing basis per the UFSAR. Since the SSE load condition was satisfied, for lesser seismic events, such as the OBE event, the inspectors concluded that the crane would retain its ability to safely hold and lower the load. The use of higher allowable stresses for the OBE load condition is a design margin concern used to justify continuing plant operations following seismic events less severe than the OBE. Therefore, the inspectors concluded that there was no immediate safety concern with the conduct of dry cask operations.

Whether the one third increase for the OBE load condition was permitted by D.C. Cook's licensing basis, and if the change was correctly applied using 10 CFR 50.59, will be categorized as a URI (URI 05000315/2012004-03; 05000316/2012004-03; 07200072/2012001-01, Design Basis of Seismic Category I Structures and Equipment). Further review of the site's design and licensing basis was needed with assistance from the Office of Nuclear Reactor Regulation through a Task Interface Agreement.

.2 Initial Loading Campaign - Operation of an Independent Spent Fuel Storage Installation at Operating Plants (60855.1)

a. Inspection Scope

The inspectors observed and evaluated the licensee's performance during loading the first canister of the initial spent fuel storage campaign to verify compliance with the Certificate of Compliance, TS, 10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste," and associated procedures.

The inspectors observed heavy loads movements inside the Auxiliary Building including: lifting the transfer cask (HI-TRAC) and placing it into the spent fuel pool; lifting the HI-TRAC from the spent fuel pool and placing it in the decontamination area; lifting the HI-TRAC from the decontamination area and placing it atop a storage cask (HI-STORM); and transfer of the MPC from the HI-TRAC to the HI-STORM while the casks were stacked on one another in a restrained configuration. The inspectors observed loading of spent fuel assemblies from the spent fuel pool into the MPC. The inspectors observed MPC processing operations including: decontamination and surveying, MPC welding, non-destructive weld examinations, hydrostatic testing, MPC draining, forced helium dehydration, and helium backfilling. The inspectors also observed heavy loads operations outside the Auxiliary Building including: movement of the HI-STORM outside the Auxiliary Building on a Goldhofer transport vehicle; transfer of the HI-STORM overpack from the Auxiliary Building to the ISFSI pad via the Goldhofer; lifting the HI-STORM off the Goldhofer utilizing a Vertical Cask Transporter; and placement of the HI-STORM in its proper location on the ISFSI pad.

During performance of the activities, the inspectors evaluated: the familiarity of the licensee's staff with procedures, supervisory oversight, and communication and coordination between the groups involved. The inspectors reviewed loading and monitoring procedures and evaluated the licensee's adherence to these procedures.

The inspectors verified that contamination and radiation levels from the HI-TRAC and HI-STORM were below the regulatory, TS, and administrative limits. The inspectors performed walkdowns of the licensee's ISFSI pad to assess the material condition of the pad and HI-STORMs.

The inspectors attended licensee briefings during dry run operations including: infrequently performed test or evolution briefings, pre-job briefs, post-job briefs, ALARA radiation dose briefs, and in-field briefs to assess the licensee's ability to identify critical steps of the evolution, potential failure scenarios, and tools to prevent errors.

The inspectors reviewed ARs and the associated follow-up actions that were generated during the loading campaign. The Inspectors also reviewed the licensee's 10 CFR 72.48, "Changes, Tests, and Experiments," screenings.

In addition the inspectors observed the licensee implement contingency procedures, following helium backfill, when a port cap would not turn in preparation for final closure of the MPC. The inspectors reviewed repair work orders, procedures, and regulatory reviews in support of replacing the stuck port cap and were present throughout the actual repair.

b. Findings

Inadequate Procedures for Implementation of Annulus Cooling to Remain in an Analyzed Thermal Condition

Introduction

The inspectors identified a Severity Level IV NCV of very low safety significance of 10 CFR 72.150, "Instructions, Procedures, and Drawings," for the failure of the licensee to have procedures in place that ensured the design basis peak fuel cladding temperature limit would not be exceeded during canister processing operations.

Description

Title 10 CFR 72.150 requires that activities affecting quality be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall require that these instruction, procedures, and drawings be followed.

After completing helium backfill, the licensee attempted to isolate the MPC by shutting the port caps, a component that isolates the MPC to permit final closure welding. When the vent port cap was turned in the shut direction, resistance was encountered by the operator. The operators in the field ceased operations and ensured the MPC was in a safe condition, then sought additional assistance. At this point, the annulus region between the MPC and HI-TRAC was not filled with water because it was previously drained in support of Forced Helium Dehydration.

The licensee generated WO 55394496-15, "Remove and Restore Vent Port Cap on MPC-141", dated August 6, 2012, to perform actions to restore the vent port cap to a functional condition to facilitate completion of canister processing activities. WO 55394496-15, Step 4.3 directed the licensee to depressurize the MPC using Steps 4.19.10.a through 4.19.10.j of 12-OHP-4051-DCO-400, "MPC Welding, Blowdown, Drying and Backfill," from approximately 5 atmospheres to 1 atmosphere.

The depressurization of the canister was needed to facilitate removal and restoration of the vent port cap. The procedure as written did not contain provisions to fill the annulus region between the MPC and the HI-TRAC with water.

The Holtec Final Safety Analysis Report Chapter 4, "Thermal Evaluation," contains a thermal evaluation of spent fuel peak cladding temperatures inside an MPC pressurized with helium to 5 atmospheres and an empty annulus region; however, it does not contain a thermal evaluation of spent fuel peak cladding temperatures inside an MPC pressurized with helium to 1 atmosphere with an empty annulus region.

The licensee provided the inspectors a specific analysis not contained within the Final Safety Analysis Report, HI-2114871, "Analysis Supporting Response to NRC Vacuum Drying Violation Notice", Revision 0, which analyzed an MPC filled with 1 atmosphere of helium and an annulus filled with un-circulated water. Following discussions with the inspectors, the licensee recognized that the WO as written did not reflect the Final Safety Analysis Report analysis or the HI-2114871 analysis. The licensee made preparations and filled the annulus prior to depressurizing the MPC to conform to the HI-2114871 analyzed condition.

Analysis

The inspectors determined that the licensee's failure to have adequate procedures to ensure the annulus was filled with water prior to depressurization to 1 atmosphere was a violation of 10 CFR 72.150 that warranted a significance evaluation. Consistent with the guidance in Section 2.2 of the NRC Enforcement Manual, ISFSIs are not subject to the SDP and, thus, traditional enforcement will be used for these facilities. Therefore the violation was dispositioned using the traditional enforcement process using Section 2.3 of the Enforcement Policy.

The violation was determined to be of more than minor significance using IMC 0612, "Power Reactor Inspection Reports," Appendix E, "Examples of Minor Issues," issued on August 11, 2009, Example 3i, since the bounding conditions for the analyzed thermal condition were not reflected in the procedures to perform the port cap repair. Specifically the licensee's lack of evaluation did not ensure spent fuel cladding temperatures during canister processing operations would remain less than the HI-2114871 acceptance criteria specified as the Spent Fuel Storage and Transportation Interim Staff Guidance-11, "Cladding Considerations for the Transportation and Storage of Spent Fuel," safety limits.

Consistent with the guidance in Section 2.6.D of the NRC Enforcement Manual, if a violation does not fit an example in the Enforcement Policy Violation Examples, it should be assigned a severity level: (1) Commensurate with its safety significance; and (2) informed by similar violations addressed in the Violation Examples. The inspectors determined that that the violation could be evaluated using section 6.5.d.2 of the NRC Enforcement Policy, as a Severity Level IV violation, in that the licensee failed to establish, maintain, or implement adequate controls to ensure that the replacement of the port cap was performed under conditions bounded by a thermal analysis that ensured the integrity of the fuel would be maintained during the repair.

Cross-cutting aspects are not assigned to traditional enforcement violations. Since this violation was dispositioned using traditional enforcement, a cross-cutting aspect is not applicable.

Enforcement

Title 10 CFR 72.150, "Instructions, Procedures, and Drawings," requires, in part, that the licensee prescribe activities affecting quality by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall require that these instructions, procedures, and drawings be followed. The instructions, procedures, and drawings must include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

Contrary to the above, on August 6, 2012, WO 55394496-15, "Remove and Restore Vent Port Cap on MPC-141," failed to prescribe activities that affect quality in documented instructions or procedures, with appropriate acceptance criteria, that ensured the design basis peak fuel cladding temperature limit would not be exceeded during canister processing operations.

This is a violation of 10 CFR 72.150, "Instructions, Procedures, and Drawings." There were no actual safety consequences since the licensee took steps to ensure the MPC remained in analyzed condition by filling the annulus with water prior to depressurizing the MPC. This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy because it was of very low safety significance (Severity Level IV) and has been documented in the licensee's CAP (AR 2012-9676) (NCV 05000315/2012004-04; 05000316/2012004-04; 07200072/2012003-01, Inadequate Procedures for Implementation of Annulus Cooling to Remain in an Analyzed Thermal Condition).

.3 Review of 10 CFR 72.212(b) Evaluations at Operating Plants (60856.1)

a. Inspection Scope

(1) Review of Independent Spent Fuel Storage Installation Pad Evaluations

The inspectors reviewed the licensee's ISFSI pad evaluations for compliance with the requirements in 10 CFR 72.212 (b)(5)(ii) during ISFSI inspections documented in NRC Inspection Report Nos.: 07200072/2011001, 05000315/2011010, and 05000316/2011010.

(2) Review of Site Characteristics Against SAR and SER

The inspectors evaluated the licensee's compliance with the requirements of 10 CFR 72.212 and 10 CFR 72.48, "Changes, Tests, and Experiments." The inspection consisted of interviews with cognizant personnel and a review of applicable documentation. The licensee is required, as specified in 10 CFR 72.212(b)(1), to notify the NRC of the intent to store spent fuel at the D.C. Cook Nuclear Plant, Units 1 and 2 ISFSI facility at least 90 days prior to the first storage of spent fuel. The licensee notified the NRC on November 17, 2011, of their intent to store spent fuel using the Holtec HI-STORM 100 Cask System according to CoC No. 72-1014, Amendment 5.

A written evaluation is required per 10 CFR 72.212(b)(6), prior to use, to establish that the conditions of the CoC have been met. "D.C. Cook Nuclear Plant Independent Spent Fuel Storage Installation (ISFSI) 10 CFR 72.212 Evaluation Report," Revision 0, dated July 29, 2012, documented the evaluations performed by the licensee prior to use of the 10 CFR Part 72 general license.

The inspectors reviewed and assessed the licensee's 10 CFR 72.212 evaluation report. The inspectors verified that applicable reactor site parameters, such as possible: fires, explosions, tornadoes, wind-generated missile impacts, seismic qualifications, lightning strikes, flooding, and temperature extremes were evaluated for acceptability with bounding values specified in the Holtec HI-STORM 100 FSAR.

(3) Review of Independent Spent Fuel Storage Installation Activities for Determination of No Adverse Impact on Site Operation or Technical Specifications

The inspectors reviewed documentation associated with increasing the Single Failure Proof capacity of the Auxiliary Building crane, crane support structure, and cask lay down areas. The review included structural evaluations associated with the seismic design of the trolley, hoist/reeving equipment, miscellaneous components, crane bridge girders, supporting structural steel, and modifications affecting the operating plant. The inspectors also reviewed various cask staging configurations and the associated evaluations demonstrating structural adequacy of the floors and other building structural components for the imposed loads and floor loading in cask lay down areas. The licensee installed, and the inspectors reviewed, seismic restraints that were used during placement of the HI-TRAC on top of the HI-STORM during MPC transfer operations.

The existing auxiliary building east crane has a Design Rated Capacity of 150 tons. The crane was upgraded to single failure proof in accordance with the requirements of NUREG 0554, "Single Failure Proof Cranes for Nuclear Power Plants," and NUREG 0612, "Control of Heavy Loads at Nuclear Power Plants," through Amendment 100 to the facility operating license in 1988. This amendment established a Maximum Critical Load of 60 tons. For the ISFSI campaign the licensee further upgraded the crane to a Maximum Critical Load of 145 tons. As part of the upgrade, the licensee performed a new seismic analysis of the auxiliary building and generated a set of new seismic in-structure response spectra to replace the existing design basis spectra. The inspectors reviewed the licensee analyses and the 10 CFR 50.59 evaluation associated with the crane upgrade modification. The inspectors noted that the new seismic analysis for the auxiliary building used methodologies consistent with the those described in the Standard Review Plan (NUREG 0800) Section 3.7.1, "Seismic Design Parameters," Revision 3; Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants, Revision 1; and Regulatory Guide 1.61, "Damping values for Seismic Design of Nuclear Power Plants," Revision 1. The inspectors reviewed structural evaluations and the associated documents to verify that the licensee adequately evaluated the crane and the supporting auxiliary building structure for the increased lifted loads due to dry cask storage operations, concurrent with a postulated OBE or a Safe Shutdown Earthquake (SSE), consistent with the plant design bases.

b. Findings

No findings were identified.

.4 (Discussed) NRC Temporary Instruction 2515/187, Inspection of Near-Term Task Force Recommendation 2.3 Flooding Walkdowns, and NRC Temporary Instruction 2515/188, Inspection of Near-Term Task Force Recommendation 2.3 Seismic Walkdowns

a. Inspection Scope

Inspectors accompanied the licensee on a sampling basis, during their flooding and seismic walkdowns, to verify that the licensee's walkdown activities were conducted using the methodology endorsed by the NRC. These walkdowns are being performed at all sites in response to a letter from the NRC to licensees, entitled "Request for Information Pursuant to Title 10 of the *Code of Federal Regulations* 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," dated March 12, 2012 (ADAMS Accession No. ML12053A340).

Enclosure 3 of the March 12, 2012, letter requested licensees to perform seismic walkdowns using an NRC-endorsed walkdown methodology. Electric Power Research Institute document 1025286 titled, "Seismic Walkdown Guidance," (ADAMS Accession No. ML12188A031) provided the NRC-endorsed methodology for performing seismic walkdowns to verify that plant features, credited in the current licensing basis (CLB) for seismic events, are available, functional, and properly maintained.

Enclosure 4 of the letter requested licensees to perform external flooding walkdowns using an NRC-endorsed walkdown methodology (ADAMS Accession No. ML12056A050). NEI document 12-07 titled, "Guidelines for Performing Verification Walkdowns of Plant Protection Features," (ADAMS Accession No. ML12173A215) provided the NRC-endorsed methodology for assessing external flood protection and mitigation capabilities to verify that plant features, credited in the CLB for protection and mitigation from external flood events, are available, functional, and properly maintained.

b. Findings

Findings or violations associated with the flooding and seismic walkdowns, if any, will be documented in the 4th quarter integrated inspection reports.

4OA6 Management Meetings

.1 Exit Meeting Summary

On October 9, 2012, the inspectors presented the inspection results to Mr. J. Gebbie and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

.2 Interim Exit Meetings

Interim exits were conducted for:

- The inspection results for the areas of Radiological Hazard Assessment and Exposure Controls; Occupational ALARA Planning and Controls; RCS Specific Activity; Occupational Exposure Control Effectiveness; and RETS/ODCM

Radiological Effluent Occurrences PI Verification with Mr. S. Lies and other members of the licensee staff on August 17, 2012.

- The results of the ISFSI dry run readiness inspections were presented on July 30, 2012, to Mr. M. Carlson and other members of the licensee staff.

- The results of the ISFSI initial loading operational inspection were presented on September 12, 2012, to Mr. L. Weber and other members of the licensee's staff.

Licensee personnel acknowledged the information presented and the inspectors confirmed that none of the potential report input discussed was considered proprietary. Proprietary material received during the inspection was returned to the licensee.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee and Contractor Employees

*L. Weber, Chief Nuclear Officer
** Terry Brown – Director, Nuclear Projects
* ** M. Carlson, Vice President of Site Support Services
*P. Schoepf, Director of Projects
*M. Scarpello, Regulatory Affairs Manager
*G. Weber, Project Manager for Dry Cask Storage Project
*H. Etheridge, Licensing Manager
P. Carteaux, Manager, Dry Cask Operations
**J. Pfabe, Licensing Lead, Dry Cask Storage
S. Bakhtiari, Dry Cask Storage Project Engineer
D. Wagemaker ,Supervisor, Radiation Protection
R. Hite, Radiation Protection Manager
J. W. Flaherty, Project Manager, Dry Cask Storage
**G. A. Weber, Program Manager, Dry Cask Storage

*Licensee and Contractor Employees in Attendance during the September 12, 2012, ISFSI Initial Loading Campaign (Operational) Interim Exit Meeting

** Employees in Attendance during the July 30, 2012, ISFSI Interim Exit Meeting

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000315/2012004-01 05000316/2012004-01	NCV	Failure to Properly Preplan and Perform Maintenance on Safety-related Equipment (1R06)
05000315/2012004-02	URI	Follow-up Inspection of Actions from NOED (4OA3)
05000315/2012004-03 05000316/2012004-03 07200072/2012001-01	URI	Design Basis of Seismic Category I Structures and Equipment (4OA5)
05000315/2012004-04 05000316/2012004-04 07200072/2012003-01	NCV	Inadequate Procedures for Implementation of Annulus Cooling to Remain in a Analyzed Thermal Condition (4OA5)

Closed

05000315/2012004-01 05000316/2012004-01	NCV	Failure to Properly Preplan and Perform Maintenance on Safety-related Equipment (1R06)
05000315/2012004-04 05000316/2012004-04 07200072/2012003-01	NCV	Inadequate Procedures for Implementation of Annulus Cooling to Remain in a Analyzed Thermal Condition (4OA5)

Discussed

NONE

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspector reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

1R01 Adverse Weather Protection

- 12-EHP-5025-TMP-002, Temperature Monitoring Program, Revision 004
- 12-OHP-4022-001-010, Severe Weather, Revision 009
- AR 2012-8261, Switchgear Room Temperature High Alarm Received
- AR 2012-8263, Unexpected Alarm 1-TSI Cabinet Temperature
- AR 2012-8267, Unexpected Alarm DCS 1-FP-1 A/C Trouble
- AR 2012-8296, Safety Related Ventilation Fan Found Tripped
- AR 2012-9841, South CRAC Humidifier Not Maintaining Control Room Humidity
- PMP-4030-001-001, Impact of Safety Related Ventilation on the Operability of Technical Specification Equipment, Revision 11
- PMP-5055-001-001, Check List for Elevated Lake Temperature, Revision 17
- PMP-5055-SWM-001, Sever Weather Guidelines, Revision 004

1R04 Equipment Alignment

- 1-OHP-4021=-56-001, Filling and Venting Auxiliary Feedwater System, Revision 30
- 1-OHP-4021-032-008AB, Operating DG1AB Subsystems, Revision 14
- 2-OHP-4030-219-022E, East Essential Service Water System Test, Revision 24
- AR 2011-2995, EDG Governor Oil Level Must be Revised in Ops Procedures
- AR 2011-3547, 2-QT-534-CD1 Coalescent Filter has an Air Leak
- AR 2012-10626, Wall Supports Lacks Tracibility Documentation
- AR 2012-10845, 1-MR-37 Containment Pressure Display Periodically Fails
- AR 2012-10919, 2-LLI-215 (2CD EDG Lube Oil Sump Tank Level Indicator)
- AR 2012-11070, System Walk downs Not Being performed Per Procedure
- AR 2012-11544, 2-QRV-251 Active Boric Acid Leak
- AR 2012-8332, U2 Received Alarm for 21A Ground
- AR 2012-8997, Oil Leaking from 1CD EDG Injectors
- AR 2012-9931, Air/Gas Void found at 2-RH-152 High Point Vent
- AR-2012-7485, U1 RCP Seal Water Return Filter D/P Reads Less than Zero
- DB-12-AFWS, Design Basis Document for the Auxiliary Feedwater System, Revision 4
- OP-2-5113, Essential Service Water Flow Diagram, Revision 82
- Technical Data Book, 1-Figure 19.9, Diesel Generator Pot Settings, Revision 35

1R05 Fire Protection

- AR 2012-10856, Fire Proofing Material Damaged
- AR-2012-100030, East Diesel Fire Pump Auto Start
- AR-2012-6358, Fire Seal F6982 Found Inoperable
- AR-2012-9791, Two of the Fire Response Carts Were Found With a Flat Tire

- DCC-CEST-180-QCF, Fireproofing (Cementitious Inorganic)-Purchase, Storage, Installation, Testing and Quality Control, Revision 2
- FHA Fire Hazards Analysis, Revision 15

1R06 Flood Protection

- AR 2012-10660, Cable Not Elevated from the Pit Floor
- AR 2012-10750, Water in manholes Contacting Cables
- AR-2012-6666, Water in Manholes, Touching Cables
- AR 2012-10680, Deficiencies Noted on Several MCC's During Walkdown
- AR 2012-12048, Work Request to Elevate Cables in MCC 2-ABD-C and 2-ABD-D
- AR 2012-12039, Work Request to Elevate Cables in MCC 1-AB-C and 1-AB-D
- AR 2012-8790, Items Identified During Routine NRC Tour
- EHI-5040-ICC, Non-EQ Insulated Cables and Connections Program, Revision 0
- PMI-5053, Cable Management Program Description, Revision 1

1R11 Licensed Operator Requalification Program

- 1-IHP-4030-113-131R, Nuclear Instrumentation Power Range Channel and Calibration Including Peripherals, Revision 6
- AR 2012-11089, Received 3 U1 Simultaneous Alarms – Suspect Power Failure
- AR 2012-11568, Failure of Power Range NI
- EMD-32A, Michigan State Police, Nuclear Plant Event Notification, Drill, August 28, 2012
- RQ-E-3704-U2-A, Cycle 3704 As-Found Simulator Evaluation – Primary, Revision 0
- WO-55411122-02, Failure of Power Range NI, September 18, 2012

1R12 Maintenance Effectiveness

- 12-EHP-4030-040-01, Leak Test of Unit 1 and Unit 2 Post Accident Containment Hydrogen Monitoring System, Revision 3
- 12-EHP-5035-MRO-001, Maintenance Rule Program Administration, Revision 20
- 2-OHP-4030-214-011, Containment Isolation and 1st Valve Operability Test, Revision 14
- AR 2010-12395, 2-TRN-B PACHMS Inoperable 2-QC-596-1
- AR 2010-8458, Pin Limit Switch Actuation Arm on Blue Hat AOVs
- AR 2010-9759, Pin Limit Switch Actuation Arm on Blue Hat AOVs
- AR 2011-1199, U2 B PACHMS as Found Data Failed
- AR 2011-14518, 2-NS-345 Has a Small Packing Leak
- AR 2011-6780, 2-CP-A-REC Recorder Pen Does Not Write
- AR 2011-7209, Low Flow Through 1-EFI-82 Renders U-1 B PACHMS Inoperable
- AR 2011-9755, 1A PACHMS Hot Box Temperature>400 degrees F.
- AR 2012-2658, 2-ECR-13 Has a Erratic Stroke But is Still Operable
- AR 2012-5347, U-2 Train B PACHMS Failed Operability Test
- AR 2012-5923, 1 'S' PACHMS Failed As Found
- AR 2012-6273, U2 Tr B PACHMS Failed As found During Scheduled Surveillance
- AR 2012-9165, 2B PACHMS 2-QC-596-1 Failed as Found Data
- Maintenance Rule A(1) PACHMS Action Plan for Calibration Failure, Revision 0
- Maintenance Rule A(1) PACHMS Action Plan for Misaligned Limit Switches, Revision 0
- PACHMS System Health Reports, 2010 – 2012
- Post Accident Containment Hydrogen Monitoring System (PACHMS) Maintenance Rule Scoping Document, Revision 2
- WO 55257968-06, MTM 2-ECR-19: Perform Leak Inspection (PMT), September 15, 2012

- WO 55357672-05, Chem: 2-QC-596-2: Investigate and Repair, April 27, 2010
- WO 55363208-02, 2_ECR-11, PMT Leak Check, March 5, 2011
- WO 55373340-16, MTO, 2-QC-596-1, Replace Amplifier Board Assemble on 2B-PACH, May 28, 2011
- WO 55398490-02, MTM 1-ECR-23-ACT Perform PMT Leak Inspection, May 12, 2012
- WO 55402504-02, MIT, 2-CA-7040, PMT Leak Testing, August 4, 2012
- WO 55403177-02, PDMT, 1-SP-A, Perform "As-Found" LLRT Testing, July 6, 2012
- WO 55410886-03, U1 Train B PACHMS Backup Air Trip Mechanism Failed, September 9, 2012

1R13 Maintenance Risk Assessments and Emergent Work Control

- AR-2012-9264, SDG #1 Tripped on Under Frequency During Testing
- PMP-7030-OPR-001, Operability Determination, Revision 20
- Unit 2 West ESW Header Part 2 Risk Assessment Sheet and Recommended Risk Management Actions, September 15, 2012

1R15 Operability Determinations and Functionality Assessments

- 01-OHP-4025-LS-3, Steam Generator 2/3 Level Control, Revision 3
- 01-OHP-4025-LS-4, Steam Generator ¼ Level Control, Revision 3
- 12-IHP-6030-IMP-031, Air Operated Valve (AOV) Diagnostic Testing and Calibration, Revision 19
- 12-OHP-4051-DCO-100, Transport Operations, Revision 2
- 1-EHP-4030-102-001, Steam Generator Primary Side Surveillance, Revision 9
- 2-EHP-4030-202-00, Steam Generator Primary Side Surveillance, Revision 11
- AR 2012 11561, Immediate Operability Determination for 2012-11462
- AR 2012-10004, 1-QFR-30 Flange Bolt Loose
- AR 2012-10262, Unexpected RCP 24 Lower BRG CLG Water Flow Low
- AR 2012-10845, 1-MR-37 Containment Pressure Display Periodically Fails
- AR 2012-10880, PMID 100762 to 100765 Not Updated EDG Monthly Fuel Rack Lube
- AR 2012-10893, 10 CFR Part 21 Notification ITT Conoflow GFH25 Regulator
- AR 2012-11174, Large Gasket Leak on 1-QT-502-AB, Needs to Be Replaced
- AR 2012-11462, Pinhole Leak from ESW Outlet Piping on U-2 West CCW Hx.
- AR 2012-11654, CD Battery Ground Alarm Ann 220 Drop 8 is coming in and out
- AR 2012-9956, Unexpected Alarm in U-2 Control Room
- AR-2011-14431, LBLODA Fuel Pellet Thermal Conductivity Degradation
- AR-2011-14431-17, LBLOCA Fuel Pellet Thermal Conductivity Degradation
- AR-2012-5450, Wrong Pinion gear was Installed on 2-NMO-151/152/153
- AR-2012-9233, SGTR DBA Analysis May Not be Met with Some Unavailable Equipment
- Case N-513-3, Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 or 3 Piping Section XI, Division 1, January 26, 2009
- Case N-661-2, Alternative Requirements for Wall Thickness Restoration of Class 2 and 3 Carbon Steel Piping for Raw Water Service Section WI, Division 1, March 22, 2007
- DIT-B-03506-00, Minimum Required Pipe Wall Thickness for Axial Loading for the ESW Piping, September 14, 2012
- MDS-609, Steam Generator Tube Plugging, Revision 7
- OP-1-5151D-67, Flow Diagram Emergency Diesel Generator "CD" Unit No. 1, November 16, 2011

- OP-2-5120Y-11, Flow Diagram 100# Control Air System Hdr. Diesel Generators 2AB & 2CD Unit #2
- VTD-ROBT-0032, Robertshaw Controls Pilot Temperature Controller, Revision 0

1R19 Post Maintenance Testing

- 12-IHP-5021-EMP-004, LIMITTORQUE SMB-100 Valve Operator Maintenance September 10, 2012
- 12-IHP-5030-EMP-014, MOV Diagnostic Testing Using Viper Test System, September 11, 2012
- 12-IHP-6030-RLY-001, General Electric Single Contact Type IAC Relays Without Instantaneous Overcurrent Device Calibration and Maintenance, Revision 008
- 12-QHP-5050-NDE-008, Ultrasonic Examination for Thickness Measurements, September 19, 2012
- 1-OHP-4021-032-001CD, DG1CD Operation, Revision 27
- 1-OHP-4030-108-053B, ECCS Valve Operability Test – Train B, Revision 20
- 1-OHP-4030-108-053V, ECCS Valve Position Verification Modes 1 – 4, Revision 3
- 1-OHP-4030-132-027CD, DG1CD Slow Speed Start, Revision 21
- 1-OHP-4030-151-018, Steam Generator Stop Valve Dump Valve Surveillance Test, Revision 4
- AR 2012-10204, Issues Identified During MOV Diagnostic Testing
- AR-2012-9158, Installed Strainers on EDG System Might Be QL 4 Instead of 1
- AR-2012-9620, Unit 2 Pressurizer Level Control with the West CCP Inservice
- AR-2012-9912, Replace 1-DCR-320-ACT
- AR-2012-8834, 1-MRV-222, #2 SG Stop Valve Dump Valve Failure
- ES-CIV-000306-QCN, Containment Isolation System Licensing/Design Basis Requirements, Revision 0
- WO 55263720-04, 2-FMO-212: Perform 'As Left' Diagnostics Testing, September 11, 2012
- WO 55390346, 1-IMO-911 PM to Perform Diagnostic Testing, July 3, 2012
- WO 55407679, EDDS Valve Lineup Modes 1-4, August 14, 2012
- WO 55410990-07, NQQS, 1-WMO-737, UT inspection of Piping, September 15, 2012
- WO 55410990-18, UT of Piping Downstream of 1-WMO-737 & 1-WMO-733, September 19, 2012
- WO-55277649-01, 2-FRV-257, 'As Found', 'As Left' Diagnostics and Valve Set Up, September 5, 2012
- WO-55389843-03, Perform PMT Leak Check on 2-FRV-257, September 5, 2102

1R22 Surveillance Testing

- 12-IHP-5030-013-001, Westinghouse Nuclear Instrumentation System Detector DC Current Characteristic Testing, Revision 3
- 12-THP-6020-INS-524, Alternate Oxygen Monitor, Revision 12
- 1-IHP-6030-IMP-309, 4Kv Bus Loss of Voltage and 4Kv Bus Degraded Voltage Relay Calibration, Revision 7
- 1-OHP-4030-108-053A, ECCS Valve Operability Test – Train A, Revision 18
- 2-IHP-4030-213-231Q, Nuclear Instrumentation Power Range Channel Operational Test and Calibration with New flux Data Equivalent Voltages, Revision 008
- 2-IHP-4030-213-231Q, Nuclear Instrumentation Power Range Channel Operational Test and Calibration with New flux Data Equivalent Voltages, Revision 8
- 2-OHP-4030-256-017E, East Motor Driven Auxiliary Feedwater System Test, September 5, 2012
- AR 2012-10885, Critical parameter Found OOS; 2-MPP-221

- AR 2012-11314, Steam Generator Regulating Valve Procedure Change
- AR-2012-10133, Step in Test Procedure Not Performed Correctly
- AR-2012-9904, CT in Breaker Cube Failed
- AR-2012-9909, 1-DCR-320 Failed Drop Test
- Figure 2-15.1, Safety Related Pump Inservice Test Hydraulic Reference, Revision 109
- Figure 2-19.1, Power Operated Valve Stroke Time Limits, Revision 102
- Technical Data Book, 1-Figure 19.1, Power Operated Valve Stroke Time Limits, Revision 112
- Technical Data Book, Figure 19.8, Safety Related Throttle Valves, Revision 28
- WO 55401427 03, Place Drawer 2-NRI-44B Out Of and Into Service per Applicable Portions of Procedure 2-IHP-4030-213-231Q for Testing
- WO-55409258, STP017E L1 E MD Aux Feed PP Flowpath, September 6, 2012

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

- 12-THP-6010-RPP-011, Radioactive Source Control, Revision 21
- 12-THP-6010-RPP-401, Performance of Radiation and Contamination Surveys, Revision 33
- 12-THP-6010-RPP-403, Portable Air Sampling, Revision 17
- 12-THP-6010-RPP-405, Analysis of Airborne Radioactivity, Revision 15
- AR 00829317, Unit 1 Vestibule Door Lock is Sticking and Not Functioning
- AR 00831020, Cannot Lock Door With B9 Key, Lock Needs Repair
- AR 00854738, Un-posted High Radiation Areas Discovered in Unit 2 Vestibule
- AR 00856787, High Radiation Area Door Found Open (unlocked)
- AR 2010-11511, Unit 2 Vestibule Door Latch Screw Broken
- AR 2011-00921, High Radiation Area Door Found Unlocked
- AR 2012-03321, Vestibule Door 2-DR-AUX370 is Significant Safety Hazard
- AR 2012-04586, MTI Personnel Locked in Unit 2 Vestibule During Job Activity
- AR 2012-05063, Two Individuals Entered a High Radiation Area Without an Alarming Vibrating Electronic Dosimeter
- AR 2012-05088, High Radiation Area Door Closed and Posted But Not Locked
- AR 2012-07360, Unit 1 Vestibule Does Not Send/Receive Messages
- AR 2012-08091, Replace Locking Bar on Unit 1 Vestibule With Hasp
- AR 2012-10103, Cook Policy for Locking High Radiation Areas
- CNP-1202-0223, Survey of Unit 2 Residual Heat Removal Vestibule, February 26, 2012
- CNP-1203-0530, Survey of Unit 2 Residual Heat Removal Vestibule, March 30, 2012
- CNP-1204-0317, Survey of Unit 2 Residual Heat Removal Vestibule, April 10, 2012
- CNP-1204-0430, Survey of Unit 2 Residual Heat Removal Vestibule, April 14, 2012
- CNP-1204-0444, Survey of Unit 2 Residual Heat Removal Vestibule, April 15, 2012
- CNP-1204-0448, Survey of Unit 2 Residual Heat Removal Vestibule, April 17, 2012
- CNP-1204-0548, Survey of Unit 2 Residual Heat Removal Vestibule, April 19, 2012
- CNP-1205-0243, Survey of Unit 2 Residual Heat Removal Vestibule, May 27, 2012
- GT 00859148, Install Hasp on Door Unit 2 Vestibule, October 16, 2010
- GT 2012-0071, Procedure Change Request, Enhance 12-OHP-4050-FHP-0005 Core Unload/Reload, August 16, 2012
- PMP-6010-RPP-003, High, Locked High, and Very High Radiation Area Access, Revision 22

2RS2 Occupational ALARA Planning and Controls (71124.02)

- AR 2012-02705, Work Order 55353648 for U2C20 was Poorly Scheduled
- AR 2012-03886, Extra Dose Due To Insulation Rework
- AR 2012-04311, Poor Work Order Task Locations and Man-Hour Breakdown
- AR 2012-04521, Outage Work Added To On-Line With No Regard To Dose Impact

- AR 2012-04699, Jacking Device Installed In Wrong Location
- AR 2012-05312, Wasted Dose
- AR 2012-06415, Radiation Protection Work Went Over Dose Estimate
- AR 2012-07561, Project Dose Budget Exceeded
- AR 2012-08966, Construction Failed to Contact ALARA Planning to Perform a Required Micro-ALARA Plan
- AR 2012-3782, Personnel Safety and Station ALARA Need Greater Focus
- AR 2012-4214, Unnecessary Dose Accrued Setting Upper Internals
- AR 2012-4749, Organization Failure to Identify Reactor Head Lift Prerequisites
- AR 2012-4879, More Work Added to On-line With No Regard to Dose Impact
- AR 2012-5324, Inadequate Planning Caused Additional Outage Dose
- AR 2012-08966, Valve Maintenance Team Failed to Contact ALARA Planning to Perform a Required Micro-ALARA Plan

4OA1 Performance Indicator Verification

- 12-THP-6020-CHM-109, Chemical and Volume Control System, Revision 18
- 1-THP-6020- CHM 121, Unit 1 Reactor Coolant System Sampling, Revision 8
- 2-THP-6020- CHM 121, Unit 2 Reactor Coolant System Sampling, Revision 9
- AR 2012-04745, Incorrect Reporting of Highest Dose Equivalent Iodine 131 for March 2012
- PMP-7110-PIP-001, Reactor Oversight Program Performance Indicators and Monthly Operating Report Data, Occupational Exposure Control Effectiveness, Revision 13
- PMP-7110-PIP-001, Reactor Oversight Program Performance Indicators and Monthly Operating Report Data, RETS/ODCM Radiological Effluent Occurrences, Revision 13
- PMP-7110-PIP-001, Reactor Oversight Program Performance Indicators and Monthly Operating Report Data, Reactor Coolant System Specific Activity, Revision 13

4OA2 Problem Identification and Resolution

- AEPDCC017-PR-02, D. C. Cook Nuclear Plant Units 1 and 2 System Vulnerability Review Report for Main Feedwater System, February 14, 2012, Revision 0
- PMP-7030-CAP-005, Conduct of Casual Evaluations, Revision 2
- PMP-7030-CAP-004, Conduct of Effectiveness Reviews, Revision 2
- PMP-7030-CAP-002, Condition Action and Closure, Revision 23
- AR 2012-8765, 2 AB EDG Room Door Degraded
- AR 2012-8444, Ladder Staged for Emergency Boration NOT Properly Secured
- AR 2012-2558, Root Cause on Station Response to Wetted Cables
- AR 2012-11783, Inadequate Walk down Performed on Job Order # 5529610801
- AR 2012-10655, Evaluate Why a PMCR Was Not Written to Inspect MCC Pits

4OA3 Followup of Events and Notices of Enforcement Discretion

- AEP-NRC-2012-61, Ltr J. P. Gebbie to U.S. NEC, Donald C. Cook Nuclear Plant Unit !, Enforcement Discretion Regarding Engineered Safety Feature Acutation System Steam Line Isolation Automatic Acution Logic and Actuation relays for Steam Generator Stop Valve Dump Valve, July 23, 2012
- AR 2012-8833, 1-MRV-212, #1 SG Stop Valve Dump Valve Lost Power
- AR 2012-8834, 1-MRV-222, #2 SG Stop Valve Dump Valve Failure
- AR 2012-8958, Potential Knowledge Deficiency Leading to Unneeded LCO 3.0.3
- AR 2012-9024, U-1 Down Power due to Entering LCO 3.2.2 on 7/19/12

- EN 48120, Technical Specification Required Shutdown due to Inability to restore Main Steam Isolation, July 19, 2012

4OA5 Other Activities

- 10CFR50.59 Evaluation 2010-0324-00, Auxiliary Building East Crane Uprate and Upgrade Modification, November 4, 2011
- 12-EHP-4051-DCO-300, Fuel Selection and Sequence Development for Dry Cask Storage, Revision 0
- 12-MHP-4030-048-001, Auxiliary Building Cranes Interlock Verification, Revision 8
- 12-OHP-4051-DCO-100, Transport Operations, Revision 0
- 12-OHP-4051-DCO-101, Goldhofer Operations, Revision 0
- 12-OHP-4051-DCO-200, MPC Preparation for Loading, Revision 1
- 12-OHP-4051-DCO-300, MPC Loading Operations, Revision 1
- 12-OHP-4051-DCO-400, MPC Welding, Blowdown, Drying and Backfill, Revision 1
- 12-OHP-4051-DCO-400, MPC Welding, Blowdown, Drying, and Backfill, April 30, 2012
- 12-OHP-4051-DCO-500, Transfer Operations, Revision 0
- 12-OHP-4051-DCO-500, Transfer Operations, Revision 1
- 12-OHP-4051-DCO-600, Dry Cask Operations Response to Abnormal Conditions, Revision 0
- 12-OHP-4051-DCO-700, MPC Unloading, Revision 0
- 12-OHP-4051-DCO-700, MPC Unloading, Revision 1
- 12-OHP-4051-DCO-701, MPC Gas Sampling, April 30, 2012
- 12-OHP-4051-DCO-805, Dry Cask Storage Lifting Device Inspections, Revision 0
- 12-QHP-5050-NDE-001, Liquid Penetrant Examination, Revision 8
- 12-THP-6010-RPP-401, Performance of Radiation and Contamination Surveys, Revision 33
- 1-OHP-4030-114-030, Independent Spent Fuel Storage Cask Heat Removal System Operability Checks, Revision 25
- 50.59 Evaluation 2010-0324-00, Auxiliary Building East Crane Uprate and Upgrade Modification, November 4, 2011
- 50.59 Evaluation 2012-0009-00, Plant Design and Licensing Bases Changes for ISFSI, Revision 0
- 72.48 Screen 2012-0111-00, Dry Cask Storage Lifting Device Inspections
- AR 2012-0747, East Aux Building Crane Main Hoist Wire Rope
- AR 2012-0876, East Aux Crane Operating Temperature
- AR 2012-3061, Possible Non-Conservative Vendor Calc – Dry Cask Lift Yoke
- AR 2012-5011, Holtec UFSAR Conflicting Requirements
- AR 2012-5826, Dry Cask Storage Run #2 RVOA Condition
- AR 2012-7725, Document Resolution of NRC Questions re: East Aux Bldg Crane
- AR 2012-9498, Dry Cask Multi-Purpose Pump Damaged During Hydro Test
- AR 2012-9556, Dry Cask MPC-141 Vent Port Cap Did Not Turn to Close
- AR 2012-9676, Dry Cask Operations During Off Normal Condition
- Calculation No. 07Q3702-01, Development Of In-Structure Seismic Response Spectra Within The D.C. Cook Nuclear Plant Auxiliary Building Using USNRC Regulatory Guide 1.60 Seismic Input, Revision 1
- Calculation No. 07Q3702-02, Auxiliary Building East crane Seismic Adequacy for a 145 Ton Lift, Revision 3
- Calculation No. 07Q3702-03, Auxiliary building and Crane rail Seismic adequacy assessment for an Operational Basis Earthquake (OBE) or Safe Shutdown Earthquake (SSE) Developed from a Regulatory Guide 1.60 Ground Input, Revision 2
- Calculation No. 07Q3702-04, Auxiliary Building Floor Evaluation for Dry Cask Storage Operation Loading, Revision 0

- Calculation No. 07Q3702-05, Auxiliary Building Truss Qualification for Seismic Loads from the HITrac/HISTORM Seismic Bracing System Connection, Revision 1
- Calculation No. 13090401-R-M-009, Evaluation of Fire and Explosion Hazards for ISFSI, Revision 2
- Calculation No. 13090401-SC-002, Evaluation of Auxiliary Building Decontamination Area Floor Capacity for HI-TRAC Loads, Revision 7
- Calculation No. HI-2084188, Dose versus Distance from a HI-STORM 100S Version B Containing the MPC-32, Revision 3
- Calculation No. HI-2084189, HI-STORM CoC Radiation Protection Program Dose Rate Limits, Revision 2
- Calculation No. HI-2084218, Cask Handling Weights and Cask Handling Dimensions at D.C. Cook, Revision 4
- Calculation No. HI-2094273 – Seismic Analysis of Suspended HI-Trac in Spent Fuel Pool, Revision 3
- Calculation No. HI-2094279, Seismic Stability and Floor Evaluation of HI-TRAC in SFP, Revision 2
- Calculation No. HI-2114808, Seismic Stability Analysis of Loaded HI-STORM on Goldhofer at DC Cook, Revision 1
- Calculation No. HI-2115043, Dynamic Analysis of Laterally restrained HI-Storm / HI-TRAC Stack for MPC Transfer for the D. C. Cook Nuclear Plant, Revision 0
- Calculation No. HI-2125197, Evaluation of Effects of Wheeled VCT Fire on HI-STORM 100S Version B, Revision 1
- Calculation No. SD-990513-006, Coupled Seismic Analysis Of Aux Bldg Crane And Support Structure, Revision 2
- Calculation No. SD-991112-001, Qualification of Aux Bldg Superstructure to OBE and SSE Seismic Loads Plus Design Basis Crane Loads, Revision 3
- Calculation No. SD-991214-002, Review of the Auxiliary Building Overhead Crane design for Larger North-South Seismic Accelerations, Revision 3
- Calculation SD-991112-001, Qualification of Aux Bldg Superstructure to OBE and SSE Seismic Loads Plus Design Basis Crane Loads, Revision 3
- Calculation SD-991214-002, Review of the Auxiliary Building Overhead Crane design for Larger North-South Seismic Accelerations, Revision 3
- D.C. Cook Nuclear Plant Independent Spent Fuel Storage Installation (ISFSI) 10CFR72.212 Evaluations Report, Revision 0
- Donald C. Cook Nuclear Plant Emergency Plan, Revision 29
- EC 49518, Auxiliary Building East Crane Uprate And Upgrade Modification, Revision 0
- EC 49520, Plant Design and Licensing Bases Changes for ISFSI, Revision 0
- EC 49525, DC Cook Dry Cask Loading Campaign #1, Revision 0
- EC-0000049518, Auxiliary Building East Crane Uprate and Upgrade, Revision 0
- Field Condition Report No. 1705-006, NRC Dry Run #1 Welding, April 27, 2012
- GT 2012-5083, Revise Dry Cask Calc 13090401-R-M-009 Transporter Fires, April 17, 2012
- GT 2012-8605, Procedure Enhancements from Dry Run #3 MPC Unloading, July 13, 2012
- GT 2012-8968, East Aux Building Crane – Cask Loading Area Interlocks, July 23, 2012
- HI-2084065, Conformed Spec for Braidwood Byron Cook Fermi LaSalle & Perry VCT's, Revision 4
- HI-2114871, Analysis Supporting Response to NRC Vacuum Drying Violation Notice, Revision 0
- HSP-504, Procedure to Perform Closure Welds on the MPC and MPC Lid, Revision 13
- HSP-505, Control and Issuance of Weld Filler Metal for MPC Site Welding Services, Revision 2

- HSP-506, Liquid Penetrant Examination for MPC Field Closure Welding, Revision 5
- HSP-507, Visual Weld Examination for MPC Field Closure Welding, Revision 3
- HSP-508, Repair of Deposited Weld Metal for MPC Field Closure Welding, Revision 1
- HSP-509, Procedure for MPC Seal Weld Removal in the Field, Revision 1
- HSP-510, Grinding Control Procedure for Site MPC Welding Operations, Revision 1
- HSP-513, Base Metal Repair Procedures for MPC Field Closure Welding, Revision 3
- MSLT-MPC-HOLTEC, Helium Leak Detection, Revision 3665-DCC-00
- NFG-DCO-11-01, Decay Heat Calculations to Support Dry Cask Storage Campaign #1, Revision 0
- PMI-6010, Radiation Protection Plan, Revision 20
- PMP-5020-MHP-001, Lifting and Rigging Program, Revision 30
- PMP-6010-ALA-001, ALARA Program – Review of Plant Work Activities, Revision 23
- PQR 1146, Machine Hot Wire GTAW on P-No. 8 Stainless Steels, Revision 1
- Quality Assurance Program Description, Revision 22
- Report C6766, Crane Seismic Report, Cask Handling Crane, 150/20 Ton Capacity, S/N 12115, Revision 3
- Report C6766, Crane Seismic Report, Cask Handling Crane, 150/20 Ton Capacity, S/N 12115, Reqn 79508, Revision 3
- Report No. 50174-10-001, Review of Documents Supporting Compliance with NUREG-0554, Revision 3
- VTD-WHCO-0015, Whiting Corporation Operating Instructions for Auxiliary Building (East) Crane, Revision 0
- Whiting Corporation Project No.: C12115.55, Main Hoist Rope Failure / Critical Load Drop Evaluation, Crane S/N 12115, Revision 1
- WO55394496-15, Remove and Restore Vent Port Cap on MPC-141, August 6, 2012
- WPS 246LW, Machine GTAW – Hot Wire on MPC Closure Welds, Revision 1

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access Management System
AISC	American Institute of Steel Construction
ALARA	As-Low-As-Is-Reasonably-Achievable
AR	Action Request
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CLB	Current Licensing Basis
CoC	Certificate of Compliance
ESFAS	Engineered Safety Feature Actuation System
HI-STORM	Storage Cask
HI-TRAC	Transfer Cask
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
ISFSI	Independent Spent Fuel Storage Installation
MPC	Multi-Purpose Canister
MRS	Monitored Retrievable Storage
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NOED	Notice of Enforcement Discretion
NRC	U.S. Nuclear Regulatory Commission
OBE	Operating Basis Earthquake
ODCM	Offsite Dose Calculation Manual
PARS	Publicly Available Records System
PI	Performance Indicator
RCS	Reactor Coolant System
RETS	Radiological Effluent Technical Specification
SDP	Significance Determination Process
SFP	Spent Fuel Pool
SSE	Safe Shutdown Earthquake
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
WO	Work Order

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Sincerely,

/RA/

John B. Giessner, Chief
Branch 4
Division of Reactor Projects

Docket Nos. 50-315; 50-316; and 072-00072
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Letter to L. Weber from J. Giessner dated October 30, 2012.

SUBJECT: D. C. COOK NUCLEAR POWER PLANT, UNITS 1 AND 2 – NRC INTEGRATED INSPECTION REPORT 05000315/2012004; 05000316/2012004; 07200072/2012001; 07200072/2012003; AND 07200072/2012004

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