March 6, 2008

EA-07-129

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

SUBJECT: D. C. COOK NUCLEAR POWER PLANT, UNITS 1 AND 2
NRC INSPECTION REPORT 05000315/2007007(DRS);
05000316/2007007(DRS) AND EXERCISE OF ENFORCEMENT DISCRETION

Dear Mr. Rencheck:

On February 20, 2008, the U. S. Nuclear Regulatory Commission (NRC) completed an inspection at your D. C. Cook Nuclear Power Plant, Units 1 and 2. The enclosed report documents the inspection results, which were discussed on February 20, 2008, with Mr. J. Petro and other members of your staff.

This inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The purpose of this inspection was to follow up on an unresolved item associated with the Unit 1 Reactor Vessel Head Replacement Inspection conducted from September 5 through October 13, 2006. The unresolved item was related to determining whether the D. C. Cook reactor vessel head lifts had been conducted within limitations necessary to ensure safety as established in the plant's licensing basis for heavy loads. Within this area, the inspection consisted of selected examination of program documents, correspondence, procedures, and representative records.

By letter dated September 14, 2007, the Nuclear Energy Institute (NEI) informed the NRC of industry approval of a formal initiative that specifies actions each plant will take to ensure that heavy load lifts continue to be conducted safely and that plant licensing bases accurately reflect plant practices. The NRC staff believes implementation of the initiative will resolve uncertainty in the licensing bases for heavy load handling, and enforcement discretion related to the uncertain aspects of the licensing basis is appropriate during the implementation of the initiative. As a result, the NRC issued Enforcement Guidance Memorandum (EGM) 07-006, "Enforcement Discretion for Heavy Load Handling Activities," on September 28, 2007.

Based on the results of this inspection, two NRC-identified issues of very low safety significance were identified, both of which involved violations of NRC requirements. However, your staff implemented actions in accordance with EGM 07-006 to warrant enforcement discretion. Therefore, consistent with EGM 07-006, we are exercising enforcement discretion in accordance with Section VII.B.6 of the NRC Enforcement Policy and are, therefore, not issuing any enforcement action for these violations.

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

Sincerely,

/RA/

David E. Hills, Chief
Engineering Branch 1
Division of Reactor Safety

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

Enclosure: Inspection Report 05000315/2007007(DRS); 05000316/2007007(DRS)
w/Attachment: Supplemental Information

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

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-315; 50-316
License Nos:	DPR-58; DPR-74
Report Nos:	05000315/2007007(DRS); 05000316/2007007(DRS)
Licensee:	Indiana Michigan Power Company
Facility:	D. C. Cook Nuclear Power Plant, Units 1 and 2
Location:	Stevensville, MI 49127-9726
Dates:	May 21, 2007 through February 20, 2008
Inspectors:	J. Neurauter, Senior Reactor Inspector
Approved by:	David E. Hills, Chief Engineering Branch 1 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000315/2007007(DRS); 05000316/2007007(DRS); 05/21/2007 – 02/20/2008; D. C. Cook Nuclear Power Plant, Units 1 and 2; Reactor Vessel Head Replacement Inspection.

This inspection followed up on an unresolved item from the modification portion of the Unit 1 Reactor Vessel Head Replacement Inspection conducted from August 5 through October 13, 2006. The inspection was conducted by a Region III reactor inspector. The inspection identified two violations of regulatory requirements of very low safety significance. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be "Green" or be assigned a severity level after review by NRC management, using qualitative measures of risk in accordance with Appendix M of IMC 0609. Consistent with the intent of NRC Enforcement Guidance Memorandum 07-006, "Enforcement Discretion for Heavy Load Handling Activities," dated September 28, 2007, enforcement discretion is being exercised for these violations in accordance with Section VII.B.6 of the NRC Enforcement Policy without any enforcement action; as a result, these violations are not being treated as findings. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

No findings of significance were identified.

B. Licensee-Identified Violations

No findings of significance were identified.

REPORT DETAILS

4. OTHER ACTIVITIES (OA)

Cornerstone: Initiating Events

4OA5 Reactor Vessel Head Replacement Inspection (71007)

.1 Historical Reactor Vessel Head Lifts

a. Inspection Scope

In following up on unresolved item (URI) 05000315/2006007-05, the inspector reviewed the circumstances surrounding the failure of the licensee (currently Indiana Michigan Power Company (I&M)) to update the Final Safety Analysis Report (FSAR) with the licensing basis reactor vessel head lift limitations of a 1978 analysis, WCAP-9198, "Reactor Vessel Head Drop Analysis," pertinent to control of heavy loads over the reactor vessel. This analysis had been completed by Westinghouse and discussed by the former licensee (Indiana Michigan Electric Company (I&MEC)) in an August 27, 1982, letter to the NRC. In addition, the inspector reviewed the licensee's corrective actions related to control of heavy loads over the reactor vessel for both Unit 1 and Unit 2. The inspection consisted of selected examination of program documents action requests (ARs), correspondence, procedures, and representative records.

b. Findings

Introduction: The inspector identified a Severity Level IV violation of 10 CFR 50.71(e) for the licensee's failure to update the FSAR in 1983 with the licensing basis reactor vessel head lift limitations pertinent to the Westinghouse WCAP-9198 analysis of postulated drops of the reactor vessel head onto the reactor vessel. This failure contributed to the licensee inappropriately conducting subsequent reactor vessel head lifts outside of licensing basis analyzed assumptions.

Description: In September 2006, an NRC inspector identified that the licensee had not considered the design basis limitations of a Westinghouse 1978 reactor vessel head drop analysis, WCAP-9198, "Reactor Vessel Head Drop Analysis," for the original head in its then upcoming replacement of the Unit 1 reactor vessel head. These limitations also had not been considered in previous lifts of the reactor vessel heads for both units.

The licensee provided this analysis, previously completed by Westinghouse in response to an NRC question pertaining to a RESAR-414 plant: to provide an analysis of the consequences of dropping the reactor vessel head assembly during refueling operations. Portions of the results from the analysis were referenced by the licensee in an August 27, 1982, letter to the NRC: a drop of the original head on the vessel would not exceed acceptance criteria from NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," Paragraph 5.1. The August 27, 1982, letter applied to the vessel heads of both units.

An attachment to the August 27, 1982, letter responded to an NRC request for additional information related to the Phase II response to Generic Letter (GL) 80 -113 and GL 81-07, as follows:

> "Analyses performed to demonstrate compliance with Criteria I through III should conform with the guidelines of NUREG-0612, Appendix A. Justify any exception taken to these guidelines, and provide the specific information requested in Attachment 2, 3, or 4, as appropriate, for each analysis performed."

The licensee responded, in-part:

> "An evaluation was done of the various load handling configurations inside containment. The results of this evaluation, given in Table 2.3-4-c, show that no credible load drops could occur, which would violate Acceptance Criteria I through III."

Table 2.3-4-c indicated that the load drop load cases were taken from WCAP-9198. Also, Acceptance Criteria I through III were taken from NUREG-0612, Paragraph 5.1:

> Release of radioactive material: No
>
> Keff increase over 0.95: No
>
> Reactor coolant boundary damage: No
>
> The basis for comparison to the above acceptance criteria was documented to be: "Subject of Westinghouse WCAP-9198"

In addition, WCAP-9198 load drop scenario parameters were indicated in Table 2.3-4-c: weight, lift heights, and fuel assembly protection during heavy lift, (air or water above the reactor vessel flange).

Prior to the fall 2006 Unit 1 refueling outage, the licensee typically lifted the reactor vessel head in excess of 38 feet above the reactor vessel flange and did not flood the reactor cavity. This lift height was significantly greater than the bounding cases analyzed in WCAP-9198 for both air and water flooding of the reactor cavity. The inspectors also determined that the licensee had not incorporated the WCAP-9198 analysis into the FSAR on February 28, 1983, for the required FSAR update, or obtained a license amendment for lifting the reactor vessel head outside the bounding parameters evaluated in WCAP-9198. As a result, these analysis limitations [lift height and reactor cavity medium (air versus water)] were neither recognized nor incorporated into plant administrative controls and related documents.

To demonstrate that the original Unit 1 reactor vessel head could be safely removed, the licensee performed additional analysis, SD-060607-001, "Sensitivity Evaluation of Reactor Head Drop Analyses Presented in WCAP-9198, Revision 0," representative of the minimum lift height required to remove the existing reactor vessel head from the reactor vessel. The analysis evaluated a 340,000 pound load drop of 34 feet

(10 feet through air and 24 feet through water). After review of the evaluation by the inspector and other NRC staff, the licensee removed the existing reactor vessel head in accordance with these analyzed parameters with additional commitments and compensatory measures in place. These commitments and compensatory measures were documented in a licensee letter to the NRC dated September 22, 2006.

To demonstrate that the replacement Unit 1 reactor vessel head could be safely installed, the licensee performed additional analysis, SD-060926-001, “Analysis of Postulated Reactor Head Load Drop onto the Reactor Vessel Flange,” prior to installing the replacement reactor vessel head onto the reactor vessel flange. The analysis evaluated 340,000 pound load drops of 33 feet, 1.5 inches (9 feet, 1.5 inches through air and 24 feet through water) and 15 feet through air. After review of the evaluation by the NRC, the licensee installed the replacement reactor vessel head in accordance with analyzed parameters with additional commitments and compensatory measures in place. These commitments and compensatory measures were documented in a licensee letter to the NRC dated September 22, 2006.

To address the licensing basis requirements related to WCAP-9198, the licensee entered the concern into their corrective action program as AR No. 00803123, “Clarify Licensing Basis Regarding Head Drop Analysis” and AR No. 00802936, “Place Procedures on Administrative Hold.”

The NRC, in a letter dated November 24, 2006, requested additional information from the licensee relative to why the WCAP-9198 analysis of reactor vessel head drop should not be included in the D. C. Cook licensing basis (i.e., should not be included in the Updated Final Safety Analysis Report (UFSAR)), or state that the licensee planned to include such in the next update of the UFSAR. The licensee’s response, dated February 20, 2007, indicated, in part, that the licensee did not consider the WCAP-9198 reactor vessel head drop analysis, referenced in their August 27, 1982, letter to be a part of their licensing basis, therefore no update to the UFSAR was required. On May 17, 2007, Region III, in consultation with NRC Office of Nuclear Reactor Regulation and Office of Enforcement, determined that the analysis reactor vessel head drop was required to be in the UFSAR.

In support of the Unit 2 reactor vessel head replacement during the fall 2007 refueling outage, licensee analysis SD-070405-001; “Analysis of Postulated Head Drop onto the Reactor Vessel Flange,” included a bounding evaluation of historical reactor vessel head lifts for Unit 1 and Unit 2 that demonstrated the structural integrity of the reactor vessel pressure boundary and reactor vessel support system for a 340,000 pound load drop of 38 feet, 1.5 inches through air. After review of the evaluation by the inspector and other NRC staff, the licensee removed the original reactor vessel head and installed the replacement reactor vessel head in accordance with analyzed parameters.

<u>Analysis</u>: The inspector determined that the failure to update the FSAR in 1983, which contributed to the licensee inappropriately conducting subsequent reactor vessel head lifts outside of pertinent licensing basis analyzed assumptions, was a performance deficiency warranting a significance evaluation. The issue was determined to be more than minor because the failure to update the FSAR as required by 10 CFR 50.71(e) resulted in the licensee performing reactor vessel head moves from 1983 to 2005 outside the licensing basis limitations to ensure safety.

Because failures to comply with 10 CFR 50.71(e) are considered to be non-compliances that potentially impede or impact the regulatory process, this issue was dispositioned using the traditional enforcement process instead of the Significance Determination Process. Therefore, the inspector evaluated this issue using Example D.6 of Supplement I of the NRC Enforcement Policy and determined this issue to be a Severity Level IV issue of very low safety significance. Specifically, licensee analysis SD-070405-001 demonstrated the structural integrity of the reactor vessel pressure boundary and reactor vessel support system for Unit 1 and Unit 2 historical reactor vessel head lifts; therefore, the failure to update the FSAR did not result in unacceptable procedures used to remove and install the reactor vessel head.

Enforcement: Title 10 CFR 50.71(e), states, in part, that each person licensed to operate a nuclear power reactor shall update periodically, as provided in paragraphs (e)(3) and (4) of this section, the FSAR originally submitted as part of the application for the operating license, to assure that the information included in the FSAR contains the latest material developed. This submittal shall contain all the changes necessary to reflect information and analyses submitted to the Commission by the licensee or prepared by the licensee pursuant to Commission requirement since the submission of the original FSAR or, as appropriate, the last updated FSAR. The updated FSAR shall be revised to include the effects of: all changes made in the facility or procedures as described in the FSAR; all safety evaluations performed by the licensee either in support of requested license amendments or in support of conclusions that changes did not involve an unreviewed safety question; and all analyses of new safety issues performed by or on behalf of the licensee at Commission request. The updated information shall be appropriately located within the FSAR.

Contrary to these requirements, on February 28, 1983, the licensee updated the FSAR, but did not include all of the latest material developed. This FSAR update did not contain changes necessary to reflect information and analyses submitted to the Commission by the licensee at Commission request. Specifically, the licensee submitted a letter dated August 27, 1982, to the NRC in response to GL 80-113 and GL 81-07, pertaining to safety issues involving the control of heavy loads at nuclear power plants. The August 27, 1982, letter indicated that if the reactor vessel head had dropped onto the vessel, the results would have been within the acceptance criteria specified in NUREG-0612, Paragraph 5.1: (I) no release of radioactive material, and (II) Keff not over 0.95, and (III) no reactor coolant boundary damage with reference to Westinghouse Report, WCAP-9198, as the basis. In addition, bounding load drop parameters from WCAP-9198 analyses were indicated in Attachment Table 2.3-4-c to the August 27, 1982, letter: reactor vessel head lift weight, lift heights, and fuel assembly protection during heavy lift (air or water above the reactor vessel flange).

By letter dated September 14, 2007, the Nuclear Energy Institute (NEI) informed the NRC of industry approval of a formal initiative that specifies actions each plant will take to ensure that heavy load lifts continue to be conducted safely and that plant licensing bases accurately reflect plant practices. The NRC staff believes implementation of the initiative will resolve uncertainty in the licensing bases for heavy load handling, and enforcement discretion related to the uncertain aspects of the licensing basis is appropriate during the implementation of the initiative. As a result, the NRC issued Enforcement Guidance Memorandum (EGM) 07-006, “Enforcement Discretion for Heavy Load Handling Activities,” on September 28, 2007. The inspector determined that the licensee implemented the following actions in accordance with EGM 07-006 to warrant enforcement discretion:

- For all heavy load lifts within the reactor building, the licensee defined and implemented safe load paths, load handling procedures, and standards for training of crane operators, use of special lifting devices, use of slings, and design, inspection, testing, and maintenance of the reactor building polar crane.

- For reactor vessel head lifts, the licensee developed load drop analyses that bounded the planned lifts with respect to load weight, load height, and media present under the load. The licensee implemented procedures for moving the load that reflected the applicable safety basis.

- The licensee established administrative controls and risk assessments for the movement of heavy loads as required to implement the requirements of 10 CFR 50.65(a)(4).

- On September 17, 2007, the licensee completed change request report UCR-1886 that revised the UFSAR to include a description of the basis for conducting safe reactor vessel head lifts, i.e., reactor vessel head drop calculations with limitations and assumptions incorporated into applicable load handling procedures.

Therefore, consistent with the intent of EGM 07-006, enforcement discretion is being exercised for this violation in accordance with Section VII.B.6 of the NRC Enforcement Policy without any enforcement action. As a result, this issue is not being treated as a finding.

.2 <u>Unit 1 Replacement Reactor Vessel Head Lift</u>

a. <u>Inspection Scope</u>

In following up on URI 05000315/2006007-05, the inspector reviewed the circumstances surrounding the licensee’s failure to assure that the correct design and licensing bases were identified for the Unit 1 replacement reactor vessel head. The inspection consisted of selected examination of program documents ARs, correspondence, procedures, and representative records.

b. <u>Issues</u>

<u>Introduction</u>: The inspector identified a Green violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for a failure to translate pertinent reactor vessel head lift limitations for the Unit 1 replacement reactor vessel head into specifications, drawings, procedures, and instructions to assure that deviations from the original reactor vessel head drop analysis were controlled.

Description: The inspector identified that the licensee had not recognized the need to incorporate WCAP-9198 design basis limitations for reactor vessel head lift, other than reactor vessel head weight, into specifications, drawings, procedures, and instructions for the Unit 1 reactor vessel head replacement. During design work associated with the Unit 1 reactor vessel head replacement modification planned for the fall 2006 refueling outage, the licensee had several opportunities to identify and correct this issue as demonstrated by the following:

- The licensee supplemented WCAP-9198 with a qualitative evaluation, MED-REE-3803, "An Evaluation of the Permanent Reactor Vessel Head Radiation Shield," dated November 12, 1990, to support the addition of 20,000 pounds due to the permanent head shielding modification. A 340,000 pound bounding load drop was used in evaluation MED-REE-3803.

- Licensee Engineering Information Record 51-9006092-001, "D. C. Cook Unit 1 RV Head Drop Evaluation," dated February 20, 2006, concluded that the replacement reactor vessel head lift weight of 315,700 pounds was acceptable; the head drop MED-REE-3803 evaluation and WCAP-9198 (Revision 1 issued in October 2004) analysis remained valid; and no further evaluation of a head drop event was necessary.

- The licensee's engineering specification ES-MECH-0908-QCN, "Reactor vessel head installation - Licensing Considerations," effective date June 21, 2006, Paragraph 4.3.2 stipulated that if the fully assembled service structure weight exceeded 340,000 pound weight limit, the analyses contained in MED-REE-3803 and WCAP-9198 (Revision 1) shall be re-evaluated.

- Licensee 10 CFR 50.59 Screen 2006-0042-00, dated May 30, 2006, compared the 315,700 pound weight for the replacement Unit 1 reactor vessel head against the 340,000 pound load evaluated in MED-REE-3803 and concluded that the existing reactor vessel head lift rig assembly had sufficient capacity and was qualified for providing the lifting function during refueling outages.

The inspector concluded that, absent NRC identification and intervention, the licensee would have likely removed the original head and installed the replacement head outside the bounding cases analyzed in WCAP-9198 during the fall 2006 refueling outage.

As a result of the NRC's concerns, the licensee:

- Performed additional analysis, SD-060607-001, "Sensitivity Evaluation of Reactor Head Drop Analyses Presented in WCAP-9198, Revision 0," representative of the minimum lift height required to remove the existing reactor vessel head from the reactor vessel. The analysis evaluated a 340,000 pound load drop of 34 feet (10 feet through air and 24 feet through water). After review of the evaluation by the inspectors, and other NRC staff, the licensee removed the existing reactor vessel head in accordance with these analyzed parameters with additional commitments and compensatory measures in place. These commitments and compensatory measures were documented in a licensee letter to the NRC dated September 22, 2006.

- Performed additional analysis, SD-060926-001, “Analysis of Postulated Reactor Head Load Drop onto the Reactor Vessel Flange,” prior to installing the replacement reactor vessel head onto the reactor vessel flange. The analysis evaluated 340,000 pound load drops of 33 feet, 1.5 inches (9 feet, 1.5 inches through air and 24 feet through water) and 15 feet through air. After review of the evaluation by the NRC, the licensee installed the replacement reactor vessel head in accordance with analyzed parameters with additional commitments and compensatory measures in place. These commitments and compensatory measures were documented in a licensee letter to the NRC dated September 22, 2006.

- Initiated corrective action program documents AR No. 00802936, dated September 20, 2006, “Place Procedures Listed on Administrative Hold” and AR No. 00803123, dated September 25, 2006, “Clarify Licensing Basis Regarding Head Drop Analysis.”

Analysis: The inspector determined that the failure to ensure that appropriate limitations from the WCAP-9198 analysis were identified and incorporated into plant procedures and/or instructions for reactor head removal and installation during the Unit 1 reactor vessel head replacement design work was a licensee performance deficiency warranting a significance evaluation. The inspector determined that this issue was greater than minor in accordance with Inspection Manual Chapter (IMC) 0612, Appendix B, “Issue Screening” because it was associated with the design control attribute of the Initiating Events Cornerstone and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown. The NRC determined that the reactor vessel head drop issues were not suitable for the Significance Determination Process analysis. Therefore, this issue was reviewed by NRC management, using qualitative measures of risk in accordance with Appendix M of IMC 0609, and was determined to be a Green issue of very low safety significance. Specifically, as a result of questions from the NRC, the licensee implemented reactor vessel head lift height and reactor cavity medium limitations for the Unit 1 reactor vessel head replacement which resulted in a very low safety significance.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” states, in-part, that measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in Section 50.2, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions to assure that deviations from quality standards are controlled. Title 10 CFR 50.2 states, in part, that design bases means that information which identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be requirements derived from analysis (based on calculation and/or experiments) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals.

Contrary to these requirements, prior to September 2006 the licensee failed to establish measures to assure that the design activities correctly translated the design basis for the reactor vessel head drop analysis into specifications, drawings, procedures, and instructions for the Unit 1 reactor vessel head replacement. Specifically, the licensee failed to assure that the deviations in reactor vessel head lift elevation and reactor vessel cavity medium (water versus air), specific parameters used in analysis of the effects of a postulated accident, for replacement of the Unit 1 reactor vessel head were controlled in accordance with the design basis established in WCAP-9198. As indicated in the Enforcement section of 4OA5.1.b above, the inspector determined that the licensee implemented actions in accordance with EGM 07-006 to warrant enforcement discretion. Therefore, consistent with the intent of EGM 07-006, enforcement discretion is being exercised for this violation in accordance with Section VII.B.6 of the NRC Enforcement Policy without any enforcement action. As a result, this issue is not being treated as a finding.

.3 (Closed) URI 05000315/2006007-05: Licensing Basis Requirements of Reactor Vessel Head Drop Analysis

This issue is discussed in Sections 4OA5.1 and .2 above. The URI is closed.

4OA6 Management Meetings

.1 Exit Meeting Summary

On February 20, 2008, the inspector presented the inspection results to Mr. J. Petro and other members of the licensee staff. The licensee acknowledged the issues presented. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. The inspector acknowledged the proprietary and confidential materials reviewed during the inspection, indicated that proprietary or confidential information would not be included in the inspection report, and indicated that copies of proprietary or confidential documents would be shredded.

.2 Interim Exit Meetings

No interim exits were conducted.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
R. Crane, Regulatory Affairs Specialist
R. Meister, Regulatory Affairs Specialist
J. Petro, Regulatory Affairs Manager
T. Satyan-Sharma, Principal Engineer – Design Engineering

Nuclear Regulatory Commission
D. Hills, Chief, Engineering Branch 1
B. Kemker, Senior Resident Inspector
J. Lennartz, Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened, and Discussed

None

Closed

05000315/2006007-05	URI	Licensing Basis Requirements of Reactor Vessel Head Drop Analysis. (Section 4OA5.3)

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather, that selected sections of 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.

10 CFR 50.59 Screen No. 2006-0042-00; Document No. 1-MOD-55003, D. C. Cook Unit - 1 Enhanced Service Structure (ESS); Revision 0

AR 00802936; Place Administrative Procedures on Hold; dated September 20, 2006

AR 00803123; Clarify Licensing Basis Regarding Head Drop Analysis; dated September 25, 2006

AR 00820787; Industry Initiative on Heavy Load Lifts; dated October 17, 2007

Calculation MED-REE-3803; An Evaluation of the Permanent Reactor Vessel Head Radiation Shield; dated November 12, 1990

Calculation SD-060607-001, Sensitivity Evaluation of Reactor Head Drop Analyses Presented in WCAP9198, Revision 0; Revision 4

Calculation SD-060926-001, Analysis of Postulated Reactor Head Load Drop onto the Reactor Vessel Flange, Revision 0

Calculation SD-070405-001, Analysis of Postulated Reactor Head Load Drop onto the Reactor Vessel Flange, Revision 0

Engineering Information Record 51-9006092-001; DC Cook Unit 1 Reactor Vessel Head Drop Evaluation; dated February 20, 2006

Engineering Specification ES-MECH-0908-QCN; Reactor Vessel Head Installation - Licensing Considerations; effective date June 21, 2006

FSAR Change Request No. UCR-1886; Incorporate Reference to Reactor Vessel Head Drop Calculations into UFSAR; Revision 0

Letter from I&MEC to NRC; Subject: Donald C. Cook Nuclear Plant Unit Nos. 1 and 2, Control of Heavy Loads - Phase II.a; dated August 27, 1982

Letter from I&M to NRC; Subject: Donald C. Cook Nuclear Plant (CNP) Unit 1 and Unit 2, Response to Request for Information Regarding Reactor Vessel Head Lift in the CNP Licensing Basis; dated February 20, 2007

Letter from NEI to NRC, Subject: Industry Initiative on Heavy Load Lifts; dated September 14, 2007

Procedure No. FP-AEP-R6; D. C. Cook Unit 1 Refueling Procedure; dated July 14, 1982

Procedure No. FP-AEP-R7; D. C. Cook Unit 1 Refueling Procedure; dated July 21, 1983

Procedure No. 12-OHP-4050-FHP-023; Reactor Vessel Head Removal with Fuel in the Vessel; Revision 2

Procedure No. 12-OHP-4050-FHP-034; Reactor Vessel Head Installation with Fuel in the Vessel; Revision 6

WCAP-9198; Reactor Vessel Head Drop Analysis; Revision 0

WCAP-9198; Reactor Vessel Head Drop Analysis; Revision 1

LIST OF ACRONYMS USED

ADAMS	Agency-wide Document Access and Management System
AR	Action Request
CFR	Code of Federal Regulations
DRS	Division of Reactor Safety
EGM	Enforcement Guidance Memorandum
FSAR	Final Safety Analysis Report
GL	Generic Letter
IMC	Inspection Manual Chapter
I&M	Indiana Michigan Power Company
I&MEC	Indiana Michigan Electric Company
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records System
SDP	Significance Determination Process
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item