UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE ROAD, SUITE 210
LISLE, IL 60532-4352

July 8, 2013

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

SUBJECT: DONALD C. COOK NUCLEAR POWER PLANT, UNITS 1 AND 2, COMPONENT DESIGN BASES INSPECTION 05000315/2013010; 05000316/2013010

Dear Mr. Weber:

On May 24, 2013, the U.S. Nuclear Regulatory Commission, (NRC) completed a follow-up inspection of two Unresolved Items (URIs) related to the 2012 Component Design Bases Inspection (CDBI) at your Donald C. Cook Nuclear Power Plant, Units 1 and 2. These URIs were documented in Inspection Report 05000315/2012007; 05000316/2012007 as items 05000315/2012007-04; 05000316/2012007-04; and 05000315/2012007-05; 05000316/2012007-05. The enclosed report documents the results of this inspection, which were discussed on May 24, 2013, with Mr. S. Lies, and other members of your staff.

The inspection examined additional licensing basis information provided by your staff related to the Steam Generator Tube Rupture design basis accident.

Based on the results of this inspection, one NRC-identified finding of very low safety significance was identified. The finding involved two violations of NRC requirements. These violations are being treated as Non-Cited Violations (NCVs), consistent with Section 2.3.2 of the Enforcement Policy. The current Enforcement Policy is included on the NRC's Web site at (http://www.nrc.gov/about-nrc/regulatory/enforcement/enforce-pol.html).

If you contest the subject or severity of 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 Resident Inspector at the Donald C. Cook Nuclear Power Plant.

In addition, if you disagree with the cross-cutting aspect assigned to the finding, 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 the Donald C. Cook Nuclear Power Plant.

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 System (PARS) component of NRC's Agencywide Documents Access and Management System (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA by P. L. Louden for/

Gary L. Shear, Director
Division of Reactor Safety

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

Enclosure: Inspection Report 05000315/2013010; 05000316/2013010
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	05000315; 05000316
License Nos:	DPR-58; DPR-74
Report No:	05000315/2013010; 05000316/2013010
Licensee:	Indiana Michigan Power Company
Facility:	Donald C. Cook Nuclear Power Plant, Units 1 and 2
Location:	Bridgman, MI
Dates:	December 31, 2012 – May 24, 2013
Inspectors:	C. Tilton, Senior Reactor Engineer R. Baker, Operations Engineer
Approved by:	B. Jose, Acting Chief Engineering Branch 2 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000315/2013010; 05000316/2013010; 12/31/2012 – 05/24/2013; Donald C. Cook Nuclear Power Plant, Units 1 and 2; Component Design Bases Inspection.

The inspection focused on the resolution of two Unresolved Items (URIs) opened during the Component Design Bases Inspection documented in Inspection Report 05000315/2012007; 05000316/2012007. The inspection was conducted by two regional engineering inspectors. One Green finding was identified by the inspectors. The finding was considered to have two Non-Cited Violations (NCVs) of NRC regulations. The significance of inspection findings are indicated by their color (i.e., Greater than Green, or Green, White, Yellow, Red) and determined using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process (SDP)," dated June 2, 2011. Cross-cutting aspects are determined using IMC 0310, "Components Within the Cross-Cutting Areas," dated October 28, 2011. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy dated January 28, 2013. 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

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance, with two associated NCVs of Technical Specification (TS), Section 5.4.1, "Procedures," and TS 3.7.4, "Steam Generator (SG) Power Operated Relief Valves (PORVs)," for the failure to implement design measures which were consistent with the licensing bases for a Steam Generator Tube Rupture (SGTR) concurrent with a Loss of Offsite Power (LOOP) to the station. Specifically, the licensee's emergency operating procedures (EOPs) 1(2) OHP-4023-E-3, "Steam Generator Tube Rupture," failed to provide adequate actions to mitigate the consequences of a SGTR, coincident with a LOOP, in sufficient time to prevent overfilling the ruptured steam generator. Additionally, the licensee failed to declare the affected unit's SG PORVs inoperable and complete the required actions when the non-safety-related control air compressor (CAC) was made unavailable and incapable of providing its required support function. With the unit's CAC unavailable, the SG PORVs would not be capable of being remotely operated from the control room during a SGTR concurrent with the LOOP. The licensee entered this issue into their corrective action program and completed modifications to establish Nitrogen as another motive force to support SG PORV operability.

  This performance deficiency was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was evaluated using the SDP in accordance with IMC 0609, Appendix A, "The Significance Determination Process for Findings at Power." Based on the Detailed Risk Evaluation required, the inspectors determined the finding was of very low safety significance (Green) because the resulting change in the Core Damage Frequency (CDF) was equal to 2.4E-8/yr. The inspectors determined the cause of this finding involved the crosscutting area of human performance, the component of decision making, and the aspect of conservative assumptions, H.1(b) in that the licensee did not adopt a requirement to demonstrate that the proposed action is safe in order to proceed rather than a requirement that it is unsafe in order to disapprove the action. Specifically, the

licensee incorrectly assumed the unaffected unit's plant air system (not backed by the emergency diesel generators) would be available during the SGTR scenario to supply motive power to the affected unit's SG PORVs. This assumption failed to take into account the licensing basis requirement of considering a SGTR and a loss of offsite power to the station (both units). (Section 1R21.b.(1))

B. Licensee-Identified Violations

No violations were identified.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R21 Component Design Bases Inspection (71111.21)

a. Inspection Scope

This inspection was focused on resolving two Unresolved Items (URIs) identified during the August 2012 Component Design Bases Inspection (CDBI). These URIs were documented in Inspection Report 05000315/2012007; 05000316/2012007 as items 05000315/2012007-04; 05000316/2012007-04, "Concerns with Ensuring Margin to Overfill in a Ruptured SG," and 05000315/2012007-05; 05000316/2012007-05, "Concerns with Operability of SG PORVs with Control Air Compressor Unavailable." Both of these issues are related to the Steam Generator Tube Rupture (SGTR) scenario and were left unresolved pending further review of licensing basis information.

The inspectors reviewed additional licensing basis information provided by the licensee. The information below documents the resolution of these two URIs.

b. Findings

(1) Failure to Implement Design Measures Consistent with the Licensing Bases for a Steam Generator Tube Rupture Concurrent with a Loss of Offsite Power to Both Units

Introduction: A finding of very low safety-significance, with two associated NCVs of Technical Specification (TS) Section 5.4.1, "Procedures," and TS 3.7.4, "Steam Generator (SG) Power Operated Relief Valves (PORVs)," was identified by the inspectors for the failure to implement design measures which were consistent with the licensing bases for a SGTR concurrent with a Loss of Offsite Power (LOOP) to the station. Specifically, the licensee's emergency operating procedures (EOPs) 1(2) OHP-4023-E-3, "Steam Generator Tube Rupture," failed to provide adequate actions to mitigate the consequences of a SGTR, coincident with a LOOP, in sufficient time to prevent overfilling the ruptured steam generator. Additionally, the licensee failed to declare the affected unit's SG PORVs inoperable and complete the required actions when the non-safety-related control air compressor (CAC) was made unavailable and incapable of providing its required support function.

Description: During the August 2012 CDBI, the inspectors identified two URIs. These issues were related to the design basis event of a SGTR concurrent with a LOOP.

Each unit's compressed air system provides the motive force for SG PORV operation. This system is described in Updated Final Safety Analysis Report (UFSAR) Section 9.8.2. It includes a single CAC and single plant air compressor (PAC) on each unit. The CACs can only service air loads on their respective unit, while the PACs pressurize a shared header and service loads on both Units 1 and 2. A backup PAC that can supply any service air or control instrument air load is also available.

The inspectors noted that the unit specific CAC, although non-safety-related, could be powered from on-site emergency power. The inspectors determined that the CAC would

be the only available source of immediate motive power for the SG PORVs during a SGTR concurrent with a LOOP event impacting both units. The inspectors observed that there were no controls on the unavailability of the CACs; therefore, the licensee could remove a CAC from service for an indefinite time period. Without a CAC, the SG PORVs would not be readily available to mitigate the SGTR concurrent with a LOOP.

Following several discussions, the licensee documented its understanding of the licensing bases in a document titled, "DC Cook CDBI Response to Question 2012-CDBI-298" (ML12320A544). The licensee stated that the licensing bases assumed a LOOP for the unit with the SGTR (i.e., the affected unit) only, and the licensing basis does not require consideration of a single failure for components credited with mitigation for the SGTR analysis. Therefore, the non-safety-related PAC, shared between both units and not powered by on-site emergency power, would provide the motive power for the PORVs.

The inspectors disagreed with the licensee's interpretation that a LOOP was a unit specific initial condition during a SGTR event. The inspectors consulted with the Office of Nuclear Reactor Regulation (NRR) personnel and subsequently issued a concurrence Task Interface Agreement (TIA) (ML13011A382). The TIA, dated December 7, 2012, documents the NRC's conclusion that the licensing bases for Donald C. Cook Nuclear Power Plant requires an assumption of a SGTR concurrent with a station-wide LOOP (i.e., LOOP impacting both units). On January 11, 2013, Inspection Report 05000315/2012007; 05000316/2012007 (ML13011A401) was issued documenting two URIs (05000315/2012007-04; 05000316/2012007-04, "Concerns with Ensuring Margin to Overfill in a Ruptured SG," and 05000315/2012007-05; 05000316/2012007-05, "Concerns with Operability of SG PORVs with Control Air Compressor Unavailable"). These issues were unresolved because the licensee indicated the conclusions of the TIA were incorrect and additional time was needed to provide supporting information regarding its position.

On February 8, 2013, the licensee provided additional information to support its position. The licensee stated the postulated scenario of a SGTR with dual unit LOOP and the CAC out of service is beyond the design and licensing basis and, therefore, not subject to evaluation via the inspection or licensing process. The licensee also explained that as described in the UFSAR, Donald C. Cook Nuclear Power Plant is designed with compressed air as a shared system and if a CAC is out of service and that unit experiences a SGTR with a LOOP, the source of control air would be the unaffected unit's PAC. The licensee further stated exclusion of the PAC as a valid source of compressed air to mitigate the SGTR event would be considered a single failure and beyond the current design and licensing basis.

The inspectors consulted further with NRR staff and reviewed additional information provided by the licensee. The inspectors noted the following:

- The inspectors agreed that certain older operating plants are credited with the use of non-safety-related equipment to mitigate events. In these cases, the licensee was required to demonstrate the non-safety-related equipment would reasonably be available and use of the equipment was bound by a safety-related path.

- The licensee's statement regarding the original safety evaluation report for Unit 1 implied the SGTR was not assumed to occur when the Unit 1 CAC was out of

service for maintenance supports the inspectors' position. As described above, the licensee would be in a limiting condition for operation if the CAC was unavailable due to its support function to the SG PORVs. Limiting conditions for operation are assumed in accident analysis when applying single failure assumptions. Since the SGTR event for Donald C. Cook Nuclear Power Plant does not include a single failure, the analysis assumes a motive force for the PORVs. The CAC is assumed and required to be functional (i.e., the PORVs OPERABLE) during a SGTR event with a LOOP.

- The inspectors agreed the installation of the Transformer No. 5 improved the reliability of the preferred offsite power source; however, this had no effect on the assumption of a loss of offsite power impacting both units.

- The inspectors agreed that Chapter 1 of the UFSAR provides a high-level summary of the principal design features and safety criteria for the two units and does not identify accident analyses assumptions. This argument supports the inspectors' position that the licensee cannot take credit for the unaffected unit's non-safety-related PAC unless explicitly approved by the NRC and described in the SGTR analysis.

- As stated in the TIA, the inspectors noted the licensing basis of Donald C. Cook Nuclear Power Plant's alternating current (ac) power to plant auxiliaries (e.g., offsite power) allows for a complete loss of the offsite grid and that loss of all (non-emergency) ac power to both units is a credible event and part of Donald C. Cook Nuclear Power Plant's design and licensing basis. In addition, even though the facility has five independent sources of offsite power to both units, current plant operation does not prevent the same sources from being aligned to the units at a specific time. Therefore, a potential configuration exists of operating both units with the same two sources of offsite power. Losing both sources of offsite power is a credible event and part of Donald C. Cook Nuclear Power Plant's licensing basis.

Based on the above, the inspectors concluded that a LOOP is a station event, not a unit-specific event. Furthermore, as applicable to the SGTR analysis, a LOOP is an initial condition of the event and should not be considered as a single failure. The inspectors concluded the statements documented in the TIA (ML13011A382) remain valid.

A review of the facility's unavailability records for the Unit 1 and Unit 2 CACs from January 1, 2000, to August 20, 2012, identified five instances for Unit 1 CAC and eight instances for Unit 2 CAC where the unavailability of the CAC was in excess of the TS allowed outage time of 24 hours for two or more SG PORVs being inoperable. In three of those instances for Unit 1 (April 18, 2001, for 79.8 hrs; November 23, 2003, for 133.2 hrs; and April 7, 2008, for 108.9 hrs) and two of those instances for Unit 2 (February 12, 2003, for 71.8 hrs; and January 16, 2006, for 103.5 hrs) the unavailability time of the CAC was in excess of the total TS allowed outage times of 54 hours to place the Unit in a Mode where the LCO does not apply.

During this event, the operators follow EOP 1(2) OHP-4023-E-3, "Steam Generator Tube Rupture." Step 7 of this procedure, directs the operators to perform a rapid cooldown by fully opening all SG PORVs on the 3 intact SGs. With the CAC unavailable during a SGTR concurrent with a LOOP, Step 7 of EOP 1(2) OHP-4023-E-3 would lead to a

response not obtained (RNO). Per procedure, this RNO directs the operators to establish backup Nitrogen. Because establishing backup Nitrogen was a manual operation, the inspectors concluded that this evolution appeared to require additional time not accounted for in the margin to overfill (MTO) calculation. The MTO analysis documented in TH-00-06, “DC Cook Unit 1 Steam Generator Tube Rupture with Operator Actions,” allows a maximum of 52 minutes to mitigate the accident from the start of the event to the point where there is no more water flowing out of the rupture. TH-00-03, “DC Cook Unit 2 Steam Generator Tube Rupture with Operator Actions,” allows a maximum of 51 minutes to mitigate the accident. The MTO analyses assume the SG PORVs open immediately on demand (zero time allotted). On Friday, July 27, 2012, a non-licensed operator performed a walkthrough of the procedure for establishing backup Nitrogen in order to locally open the SG PORVs. This evolution took a total of 27 minutes. This additional time confirmed that the licensee did not take this RNO evolution into account in the MTOs calculations.

In response to these identified issues, the licensee entered this concern into its CAP as Action Request (AR) 2012-9233 for further evaluation. In addition, since the original inspection, the licensee completed a plant modification which connected the existing Nitrogen system with the compressed air system. This modification enabled the nitrogen to be immediately available (without manual operator action in the field) in the event of a loss of all compressed air system pressure and will be reviewed for adequacy under the baseline inspection program.

Analysis: The inspectors determined that the failure to implement design measures consistent with the licensing bases for a SGTR concurrent with a LOOP to both units was reasonably within the licensee’s ability to foresee, correct, and prevent, and therefore was a performance deficiency. This performance deficiency was assessed using Inspection Manual Chapter (IMC) 0612, Appendix B, “Issue Screening,” and determined to be more than minor because it was associated with the Mitigating Systems Cornerstone attribute of equipment performance and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, with the CAC on the affected unit unavailable, if a SGTR with a LOOP to the facility occurred, the normal method of cooldown of the reactor coolant system using the SG PORVs would not be immediately available which would lead to an increased risk of a SG overfill condition and potential for increased dose consequences.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, Appendix A, “The SDP for Findings at Power,” Exhibit 2, “Mitigating Systems Screening Questions,” issued June 19, 2012. The inspectors answered "Yes" to the Question A.2, “Does the finding represent a loss of system and/or function?” because with a CAC on the affected Unit unavailable due to maintenance, if a SGTR event with a LOOP to the facility occurred, it would render the normal method of cooldown of the reactor coolant system using the SG PORVs not immediately available or unreliable which would lead to an increased risk of a SG overfill condition. As a result, a Detailed Risk Evaluation was required.

The Donald C. Cook Nuclear Power Plant Standardized Plant Analysis Risk (SPAR) Model Version 8.20 and Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) Version 8.0.8.0 software was used to evaluate the risk-significance of this finding. The exposure time for the finding was determined by the most limiting time that a CAC on one of the Units was unavailable in a one-year time

period. Based on review of plant data, the most limiting time that a CAC on one of the Units was unavailable in a one-year time period was bounded by 30 days. Using information from the SPAR model and the unavailability of 30 days of a CAC due to maintenance, the following was obtained:

Description	Value
Initiating Event Frequency for SGTR Event	2.07E-3/yr
Unavailability Due to Maintenance of Control Air Compressor (30 days in one-year period)	8.22E-2/yr
Probability of a LOOP Given a Reactor Trip	5.29E-3
Probability of a Dual Unit LOOP (Switchyard-Centered)	1.94E-1

The capability to align a nitrogen backup to the SG PORVs to allow valve operation was credited. The human error probability (HEP) of aligning nitrogen to the SG PORVs was determined using the SPAR-H method (per NUREG/CR-6883). For diagnosis, stress was determined to be high. For action, stress was determined to be high, complexity was determined to be moderate, and experience/training was determined to be low. This resulted in an HEP of aligning nitrogen to the SG PORVs of 3.2E-2. If the nitrogen backup to the SG PORVs is required to be aligned, it is expected that SG overfill would occur during a SGTR event due to the amount of time required to align the nitrogen, resulting in the late termination of break flow from the reactor coolant system to the secondary side of the SG. Per NUREG-0844, an SG overfill event was evaluated with a failure probability of 0.1 that a main steam (MS) safety valve would stick-open due to water relief through the valve. For conservatism it was assumed that if a MS safety valve stuck-open, then core damage would result. In addition, the probability of the failure of the SG PORVs to open (5.56E-3 per valve per the SPAR model) was taken into account.

Using the above information, the delta core damage frequency associated with the finding was determined to be 2.4E-8/yr.

Based on the Detailed Risk Evaluation, the inspectors determined that the finding was of very low safety significance (Green).

The inspectors determined the cause of this finding involved the cross-cutting area of human performance, the component of decision making, and the aspect of conservative assumptions, H.1(b), in that the licensee did not adopt a requirement to demonstrate that the proposed action is safe in order proceed rather than a requirement that it is unsafe in order to disapprove the action. Specifically, as recent as April 20, 2012, the licensee incorrectly assumed the unaffected unit's plant air system (i.e., not backed by the emergency diesel generators) would be available during the SGTR scenario to supply motive power to the affected unit's SG PORVs. This assumption failed to take into account the licensing basis requirement of considering a SGTR and a loss of offsite power to the station (both units).

Enforcement: The inspectors identified the following two violations of NRC requirements were associated with this finding:

- Technical Specification 5.4.1, "Procedures," requires, in part, that written procedures shall be established, implemented, and maintained covering emergency operating procedures (EOPs) required to implement the

requirements of NUREG-0737, “Clarification of TMI Action Plan Requirements,” and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33.

Section I.C.1 of NUREG-0737, “Clarification of TMI Action Plan Requirements,” states, in part, that emergency procedures are required to be consistent with the actions necessary to cope with the transients and accidents analyzed, including SGTR accidents.

NUREG-0737, Supplement 1, addresses development of plant specific EOPs from Generic Technical Guidelines, which includes plant specific information.

Title 10 CFR Part 50, Appendix B, Criterion III requires, in part, measures be established to assure that applicable design bases are correctly translated into procedures and instructions.

Final Safety Analysis Report, Section 14.2.1, “Steam Generator Tube Rupture,” states, in part, that a SGTR accident will not result in the overfilling of the ruptured SG, in that, SG volume will be controlled using the PORVs.

Contrary to the above, as of July 27, 2012, the licensee’s EOPs 1(2) OHP-4023-E-3, “Steam Generator Tube Rupture,” did not provide actions to mitigate the consequences of a SGTR in sufficient time to prevent overfilling the ruptured steam generator. Specifically, the procedure did not account for the affected unit’s CAC being unavailable and for the additional time required to establish backup nitrogen to operate the SG PORVs.

- Technical Specification 3.7.4, “SG PORVs,” requires in part, that four SG PORVs shall be OPERABLE in Modes 1, 2, and 3. Condition B requires that if two or more SG PORVs are inoperable in Mode 1, 2, or 3, all but one SG PORV must be restored to OPERABLE status within 24 hours. Condition C requires in part, that if the Required Action and associated Completion Time of Condition B are not met, the Unit must be placed in Mode 3 within 6 hours and in Mode 4 (e.g., a condition without the reliance upon SGs for heat removal) within 24 hours.

  As defined in the facility’s Unit 1 and Unit 2 TS, “A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function are also capable of performing their related support function(s).” Because the CAC will be the only functional motive force for the SG PORVs during a SGTR event with a LOOP, the CAC is considered support equipment to ensure the SG PORVs will operate to perform the mitigating function of cooling the reactor coolant system. The inspectors concluded this equipment is required to be functional for the SG PORVs to be considered OPERABLE.

  Contrary to the above, for Unit 1 on April 18, 2001; November 23, 2003; and April 7, 2008; and for Unit 2 on February 12, 2003 and January 16, 2006, the licensee failed to meet the conditions specified in TS 3.7.4, “SG PORVs.” Specifically, the licensee failed to declare all four of the affected unit’s SG

PORVs inoperable when that unit's CAC was unavailable with the unit in Mode 1, and failed to enter the applicable TS LCO condition statement or complete the required actions prior to exceeding the appropriate completion time.

Because these violations are of very low safety significance (Green) and were entered into the licensee's CAP, these violations are being treated as NCVs consistent with the NRC Enforcement Policy. (NCV 05000315/2013010-01, 05000316/2013010-01; Failure to Maintain Emergency Operating Procedures for Mitigating the Consequences of a SGTR per TS Section 5.4.1, "Procedures": NCV 05000315/2013010-02, 05000316/2013010-02; Failure to Enter the Limiting Condition for Operations and Perform Required Actions per TS 3.7.4, "SG PORVs")

4. OTHER ACTIVITIES

4OA5 Other Activities

.1 (Closed) Unresolved Item 05000315/2012007-04; 05000316/2012007-04: Concerns with Ensuring Margin to Overfill in a Ruptured Steam Generator

This URI is closed to NCV 05000315/2013010-01; 05000316/2013010-01, "Failure to Maintain Emergency Operating Procedures for Mitigating the Consequences of a SGTR per TS Section 5.4.1, "Procedures," as documented in Section 1R21.b.(1) of this report.

.2 (Closed) Unresolved Item 05000315/2012007-05; 05000316/2012007-05: Concerns with Operability of SG PORVs with Control Air Compressor Unavailable

This Unresolved Item is closed to Non-Cited-Violation 05000315/2013010-02; 05000316/2013010-02, "Failure to Enter the Limiting Condition for Operations and Perform Required Actions per Technical Specifications 3.7.4, "SG PORVs," as documented in Section 1R21.b.(1) of this report.

4OA6 Management Meeting

.1 Exit Meeting Summary

On May 24, 2013, the inspectors presented the inspection results to Mr. S. Lies and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed none of the information reviewed was proprietary.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

S. Lies, Executive Vice President
J. Gebbie, Site Vice President
M. Scarpello, Regulatory Affairs and Performance Improvement Department Manager
W. Hodge, I&C Design Engineering Supervisor
S. Mitchell, Licensing Activity Coordinator (Compliance)

Nuclear Regulatory Commission

A. M. Stone, Acting Deputy Director, Division of Nuclear Materials Safety
J. Ellegood, Senior Resident Inspector
P. LaFlamme, Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Closed

05000315/2012007-04; 05000316/2012007-04	URI	Concerns with Ensuring Margin to Overfill in a Ruptured Steam Generator.
05000315/2012007-05; 05000316/2012007-05	URI	Concerns with Operability of SG PORVs with Control Air Compressor Unavailable

Opened/Closed

05000315/2012010-01; 05000316/2013010-01	NCV	Failure to Maintain Emergency Operating Procedures for Mitigating the Consequences of a SGTR per TS Section 5.4.1, “Procedures”. (Section 1R21.b.(1))
05000315/2012010-02; 05000316/2013010-02	NCV	Failure to Enter the Limiting Condition for Operations and Perform Required Actions per TS 3.7.4, “SG PORVs.” (Section 1R21.b.(1))

Discussed

None.

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 stated in the body of the inspection report.

Number	Description or Title	Revision
AR 2012-9233	SGTR DBA analysis may not be met with some unavailable equipment	7/27/12
Licensee's White Paper	DC Cook Steam Generator Tube Rupture Margin to Overfill Licensing Basis Position	9/12/12

LIST OF ACRONYMS USED

ac	Alternating Current
ADAMS	Agencywide Documents Access and Management System
AR	Action Request
CAC	Control Air Compressor
CAP	Corrective Action Program
CDBI	Component Design Bases Inspection
CDF	Core Damage Frequency
DBA	Design Basis Accident
DRS	Division of Reactor Safety
EOP	Emergency Operating Procedure
HEP	Human Error Probability
IMC	Inspection Manual Chapter
LCO	Limiting Condition for Operation
LOOP	Loss of Offsite Power
MS	Main Steam
MTO	Margin to Overfill
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
PAC	Plant Air Compressor
PARS	Publicly Available Records System
PORV	Power Operated Relief Valve
RASP	Risk-Assessment Standardization Project
RTO	Response Not Obtained
SAPHIRE	Systems Analysis Programs for Hands-On Integrated Reliability Evaluations
SDP	Significant Determination Process
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SPAR	Standardized Plant Analysis Risk
TIA	Task Interface Agreement
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item

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 System (PARS) component of NRC's Agencywide Documents Access and Management System (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA by P. L. Louden for/

Gary L. Shear, Director
Division of Reactor Safety

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

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