UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 E LAMAR BLVD
ARLINGTON, TX 76011-4511

February 11, 2016

Shane M. Marik, Vice President
Omaha Public Power District
Fort Calhoun Station FC-2-4
P.O. Box 550
Fort Calhoun, NE 68023-0550

SUBJECT: FORT CALHOUN STATION – NRC CONFIRMATORY ACTION LETTER FOLLOW UP INSPECTION 05000285/2016007

Dear Mr. Marik:

On January 15, 2016, the U.S. Nuclear Regulatory Commission (NRC) completed a Confirmatory Action Letter follow-up team inspection your Fort Calhoun Station (FCS) and on January 14, 2016, discussed the results of this inspection with Mr. Todd Tierney and other members of your staff. The inspection team documented the results of this inspection in the enclosed inspection report.

During this inspection, the NRC examined activities conducted under your license as they relate to public health and safety with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

This inspection focused on assessing activities related to the implementation of the commitments described in Confirmatory Action Letter (CAL) EA-13-243, issued December 17, 2013 (ML13351A395). CAL EA-13-243 confirmed the Omaha Public Power District’s (OPPD’s) commitments to ensure the improvements realized during the previous extended outage remain in place, and performance continues to improve at the facility. Specifically, this inspection reviewed the action items associated with the subject commitments to determine which could be closed, and this inspection report describes the results of those reviews.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390, “Public Inspections, Exemptions, Requests for Withholding,” a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC’s Public

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Sincerely,

/RA/

Jeffrey Sowa, Chief (Acting)
Project Branch D
Division of Reactor Projects

Docket: 50-285
License: DPR-40

Enclosure: NRC Inspection Report 05000285/2016007
w/Attachment: Supplemental Information

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cc w/ encl: Electronic Distribution

ADAMS ACCESSION NUMBER: ML16042A542

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OFFICE	DRP/B	DNMS	DRP/D	DRP/D			
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DATE	2-10-16	2-9-16	2/11/16	2/11/16			

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Letter to S.Marik from Jeffrey Sowa, dated February 11, 2016

SUBJECT: FORT CALHOUN STATION – NRC CONFIRMATORY ACTION LETTER FOLLOW UP INSPECTION 05000285/2016007

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000285
License:	DPR-40
Report:	05000285/2016007
Licensee:	Omaha Public Power District
Facility:	Fort Calhoun Station
Location:	9610 Power Lane Blair, NE 68008
Dates:	January 11 – 15, 2016
Inspectors:	B. Hagar, Senior Project Engineer (Lead) D. Dodson, Senior Resident Inspector, Wolf Creek Station B. Baca, Health Physicist
Approved By:	Jeffrey Sowa, Chief (Acting) Branch D, Division of Reactor Projects

SUMMARY

IR 05000285/2016007; 01/11/16 – 01/15/16; Fort Calhoun Station; Confirmatory Action Letter Follow-up Inspection.

The inspection activities described in this report were performed from January 11-15, 2016, by two inspectors from the NRC's Region IV office, and during the several weeks preceding January 11, 2016, by a health physicist in the NRC's Region IV office. During this inspection, the inspectors did not identify a finding.

The significance of inspection findings is indicated by their color (Green, White, Yellow, or Red), which is determined using Inspection Manual Chapter 0609, "Significance Determination Process." Their cross-cutting aspects are determined using Inspection Manual Chapter 0310, "Components Within the Cross-Cutting Areas." Violations of NRC requirements are dispositioned in accordance with the NRC Enforcement Policy. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process."

REPORT DETAILS

4. OTHER ACTIVITIES (OA)

4OA4 Confirmatory Action Letter (CAL) Inspection Activities (92702)

The inspection team assessed and verified certain commitments described in the Confirmatory Action Letter (CAL) issued December 17, 2013 (ML13351A395). That CAL stated that it would remain in effect until the NRC verifies that Omaha Public Power District (OPPD) effectively implements the commitments identified below:

1. OPPD commits to implement those actions detailed in the December 2, 2013, letter titled, "Integrated Report to Support Restart of Fort Calhoun Station and Post-Restart Commitments for Sustained Improvement" (ML13336A785), associated with the following areas:

   - Organizational Effectiveness, Safety Culture, and Safety Conscious Work Environment
   - Problem Identification and Resolution
   - Performance Improvement and Learning Programs
   - Design and Licensing Basis Control and Use
   - Site Operational Focus
   - Procedures
   - Equipment Performance
   - Programs
   - Nuclear Oversight
   - Transition to the Exelon Nuclear Management Model and Integration into the Exelon Nuclear Fleet

2. OPPD commits to complete the following actions detailed in the Flooding Recovery Action Plan: 1.2.3.21, 1.2.3.82, and 4.4.3.1 through 4.4.3.3. These actions entail:

   - Item 1.2.3.21 - Inspect tank and equipment on demineralized water tank for damage

   - Item 1.2.3.82 - Perform independent spent fuel storage installation route load test

   - Item 4.4.3.1 - Gather flood response lessons learned through condition report reviews to determine if procedure or strategy changes should be implemented

   - Item 4.4.3.2 - Review flood design basis and determine if the 2011 flood event provides additional information that should drive design basis changes

   - Item 4.4.3.3 - Implement procedure and strategy changes as indicated by the lessons learned review conducted

3. OPPD commits to complete actions 4.5.1.14 and 4.5.1.15 (tracked through 4.5.3.06) detailed in the Flooding Recovery Action Plan, "Perform HELB [High Energy Line Break] analysis of Auxiliary Steam in the Auxiliary Building" and "Implement resolution of Auxiliary Steam piping in the Auxiliary Building."

4. OPPD commits to:

   - Evaluate the structural design margin for the containment internal structures, and reactor cavity and compartments, and resolve any deficiencies in accordance with its corrective action program (CAP).

   - Regarding Beam 22A and Beam 22B in the containment internal structures, resolve any deficiencies in accordance with the CAP.

   - Regarding the reactor head stand, prior to the next use of the reactor head stand, OPPD will evaluate the structural design margin for the head stand and resolve any deficiencies in accordance with the CAP.

The sections below report the status of these commitments. Specifically,

- Section 4OA4.1 reports the status of those actions detailed in the December 2, 2013, letter titled, "Integrated Report to Support Restart of Fort Calhoun Station and Post-Restart Commitments for Sustained Improvement."

- Section 4OA4.2 reports the status of the subject actions detailed in the Flooding Recovery Action Plan.

- Section 4OA4.3 reports the status of the subject commitments associated with Auxiliary Steam piping in the Auxiliary Building.

- Section 4OA4.4 reports the status of the subject containment internal structures.

- Section 4OA4.5 lists the action items within the CAL that remain open after this inspection.

.1 <u>Actions detailed in the December 2, 2013, letter titled, "Integrated Report to Support Restart of Fort Calhoun Station and Post-Restart Commitments for Sustained Improvement" (ML13336A785)</u>

In the subject letter, OPPD characterized the subject actions as "Key Drivers for Achieving and Sustaining Excellence." The subsections below describe each of those key drivers. Each subsection begins with a subsection number and the key driver title in bold text, and within each subsection are one or more parts that correspond to the key driver action items listed in the subject letter. Each subsection part begins with an underlined header that includes the item number, the title, and, in parentheses, the Plant Integrated Improvement Matrix (PIIM) Action Item (AI) number that the licensee used in the subject letter to identify the key driver action items. Also,

- if the NRC had previously closed the action item, then within the subsection part is only a statement that identifies the inspection report (IR) in which the NRC had previously closed the action item; however,

- if the NRC had not previously closed the action item, then within the part are statements that describe (1) the inspection scope, (2) the most-notable observations that resulted from inspecting the action item, and (3) the assessment results.

For the action items inspected by the team, the team verified implementation via the following activities, as applicable:

- verifying that the action item descriptions correspond to the action item descriptions in Enclosure 3 of OPPD's December 2, 2013, letter;

- reviewing documents produced or revised by the action item and/or records resulting from implementation of the action item;

- verifying completion of the action item as scheduled;

- assessing the licensee's effective use of appropriate performance metrics to demonstrate performance improvement; and

- where applicable, performing independent verification of improved performance.

.1.1 Organizational Effectiveness, Safety Culture and Safety Conscious Work Environment

Item 1.a: Organizational Effectiveness (2013-0014)

Closed in IR 05000285/2014009 (ML14318A886), Section 4OA4.1.

Item 1.b: Station Safety Culture/Safety Conscious Work Environment (2013-0006)

Closed in IR 05000285/2014009 (ML14318A886), Section 4OA4.1.

.1.2. Problem Identification and Resolution

Item 2.a: Corrective Action Program (CAP) Excellence Plan – Problem Identification (2013-0055)

Closed in IR 05000285/2015008 (ML15071A115), Section 4OA5.b.2.

Item 2.b: CAP Excellence Plan – Root Cause and Apparent Cause Quality (2013-0065)

Closed in IR 05000285/2015008 (ML15071A115), Section 4OA5.b.1.

Item 2.c: CAP Excellence Plan – Corrective Action Closure (2013-0062)

Closed in IR 05000285/2015008 (ML15071A115), Section 4OA5.b.1.

.1.3. Performance Improvement and Learning Programs

Item 3.a: Performance Improvement (2013-0015)

Closed in IR 05000285/2015008 (ML15071A115), Section 4OA5.b.2.

Item 3.b: Human Performance (2013-0061)

Closed in IR 05000285/2014009 (ML14318A886), Section 4OA4.3.

.1.4. Design and Licensing Basis Control and Use

Item 4.a: Design and Licensing Basis (2013-0086)

(1) Inspection Scope

As described in IR 05000285/2014009 (ML14318A886), Section 4OA4.4, the NRC previously inspected these action items with satisfactory results:

- AI 2013-05570-010, Strengthen the Engineering Assurance Group to improve the oversight of engineering products that affect the design or licensing basis.
- AI 2013-05570-025, Complete Phase 2 of the key calculation identification and improvement process. Phase 2 of the process evaluates the critical calculation's defined purpose and methodology, defined acceptance criteria, and appropriateness of the results and conclusions.
- AI 2013-05570-067, Develop and implement an aggregate station performance indicator to measure the effectiveness of maintenance and use of licensing and design bases information.
- AI 2013-05570-079, Decide the appropriate Design Basis Document (DBD) model for Fort Calhoun Station.
- AI 2013-05570-091, Perform a technical assessment of modifications performed between January 1, 1989, and January 1, 2007, on a population of the top six risk significant systems that provides a 95/95 confidence level that no nuclear safety issues have been introduced into the plant.

In IR 05000285/2014009, the NRC closed AI 2013-05570-049 ("Improve the engineering support personnel training regarding the design and licensing basis") with the comment that upon final closure, the NRC would review this action item for adequacy.

During this inspection, besides reviewing the adequacy of AI 2013-05570-049, the team also reviewed implementation of these action items:

- AI 2013-05570-026, Identify and define the current licensing bases and assure licensing bases documentation remains current, accurate, complete, and retrievable.

- AI 2013-05570-052, Deliver the modified training to the engineering support personnel.

- AI 2013-05570-057, Develop performance metrics to trend and trigger action on the performance of the use, implementation, and identification of design and licensing bases issues such as, effective and ineffective 50.59 evaluations, and procedure inadequacies related to design and licensing bases.

- AI 2013-05570-076, Identify and define the design bases and assure design bases documentation remains current, accurate, complete, and retrievable.

- AI 2013-05570-092, Complete Phase 3 of the Key Calculation Project. Phase 3 consists of revising any deficient critical calculation or engineering analysis identified from Phase 2, as needed.

- AI 2013-05570-093, Validate the design and licensing basis has been translated into plant operation by verifying that the operation, surveillance, and maintenance of the safety-related components do not compromise the design and licensing basis.

- AI 2013-17439-003, Ensure Design Engineering performs at least one engineering self-assessment on a risk significant system in 2014.

- AI 2013-17439-004, Ensure Design Engineering performs at least one engineering self-assessment on a risk significant system in 2015.

- AI 2013-17439-005, Assign condition reports to ensure Design Engineering continues to perform an engineering self-assessment on risk significant systems each year.

(2) Observations and Findings

- Inspection of AIs 2013-05570-057, 2013-17439-003, and 2013-17439-004 resulted in no notable observation.

- In CR 2013-05570-049, the licensee characterized the action item in a way that was different from how they had characterized it in their December 2, 2013, letter. Specifically, in their December 2, 2013, letter, they characterized this action as:

  > “Modify engineering support personnel initial and continuing training addressing the design and licensing basis record types and retrieval.”

  However, in CR 2013-05570-049, they characterized it as:

  > “CAPR-3- Modify the Engineering Support Personnel Training (ESPT) initial and continuing training programs to incorporate CAPR-1 and CAPR-2. Training shall include items 1, 2 and 3 from CAPRs 1 and 2 to address the identification of design and licensing bases, record types that are included, and the method of retrieval.”

(In the text above, “CAPR” stands for “corrective action to prevent recurrence.)

Thus, determining the adequacy of CR 2013-05570-049 was beyond the scope of this inspection, so the NRC deferred that determination until a later inspection.

- By completing AIs 2013-17439-003 & -004, the licensee demonstrated that they can effectively schedule and complete self-assessments of risk significant systems. This provides confidence that they will complete AI 2013-17439-005 on its due date.

(3) Assessment Results

The NRC considers AIs 2013-05570-057, 2013-17439-003, 2013-17439-004, and 2013-17439-005 to be closed.

The NRC deferred inspecting the adequacy of CR 2013-05570-049 until a later inspection.

At the time of this inspection, the licensee had scheduled completion of the remaining action items on July 20, 2018:

AI Number	Description
2013-05570-026	Identify and define the current licensing bases and assure licensing bases documentation remains current, accurate, complete, and retrievable.
2013-05570-076	Identify and define the design bases and assure design bases documentation remains current, accurate, complete, and retrievable.
2013-05570-092	Complete Phase 3 of the Key Calculation Project. Phase 3 consists of revising any deficient critical calculation or engineering analysis identified from Phase 2, as needed.
2013-05570-093	Validate the design and licensing basis has been translated into plant operation by verifying that the operation, surveillance, and maintenance of the safety-related components do not compromise the design and licensing basis.

Because inspecting the adequacy of CR 2013-05570-049 is not complete and the activities listed in the table above are not complete, this item remains open.

.1.5. Site Operational Focus

Item 5.a: Site Operational Focus, Operational Decision Making and Anticipating System Response (2013-0037)

Closed in IR 05000285/2014009 (ML14318A886), Section 4OA4.5.

.1.6. Procedures

Item 6.a: Procedure Quality and Procedure Management (2013-0012)

Closed in IR 05000285/2014009 (ML14318A886), Section 4OA4.6.a.

Item 6.b: Abnormal and Emergency Operating Procedures (2013-0031)

Closed in IR 05000285/2014009 (ML14318A886), Section 4OA4.6.b.

Item 6.c: Transition to the Exelon Nuclear Management Model and Integration into the Exelon Nuclear Fleet (2013-0077)

Closed in IR 05000285/2014009 (ML14318A886), Section 4OA4.6.c.

.1.7. Equipment Performance

Item 7.a: Tornado Protection (2013-0041)

Closed in IR 05000285/2014009 (ML14318A886), Section 4OA4.7.a.

Item 7.b: Equipment Service Life (2013-0088)

Closed in IR 05000285/2014009 (ML14318A886), Section 4OA4.7.b.

Item 7.c: Equipment Reliability/Containment Internal Structures (2013-0013)

(1) Inspection Scope

The team reviewed the implementation of AI 2012-04392-014, "Restore the design criteria for the Internal Structure of Containment, including any needed plant modifications to beam 22A and B."

(2) Observations and Findings

In CR 2012-04392-014 AI, the licensee characterized this action item as,

> "Resolve discrepancies for the Internal Structure of Containment, including any needed plant modifications. Implement design modifications to restore the Containment Internal Structure (CIS) to within its design basis requirements."

On December 23, 2015, the licensee submitted LIC-15-0142, "Supplement of License Amendment Request 15-03; Revise Current Licensing Basis to Use ACI Ultimate Strength Requirements," (ML15363A042), to supersede License Amendment Request (LAR) 15-03, "Revise Current Licensing Basis to Use ACI Ultimate Strength Requirements," dated August 31, 2015. To allow the NRC time to respond to that request, the licensee extended the due date for CR 2012-04392-014 AI to December 15, 2016.

The licensee is tracking the implementation part of this action item in CR 2012-04392-045 AI, which they described as, "Implement design modifications to restore the Reactor Cavity and Compartments (RC&C) to within its design basis requirements during the next refueling outage (RFO 27)."

Because resolution of CR 2012-04392-045 AI is linked to the resolution of CR 2012-04392-014 AI, the licensee also extended its due date to December 15, 2016.

(3) Assessment Results

Because CR 2012-04392-014 AI is not complete, CAL item 7.c remains open.

Item 7.d: Equipment Reliability/Equipment Performance (2013-0027)

(4) Inspection Scope

The team reviewed the implementation of the following action items:

- AI 2012-08134-039, Perform an interim effectiveness review of the Plant Health Committee process and performance.

- AI 2012-08134-040, Perform a final effectiveness review of the Plant Health Committee process and performance.

(5) Observations and Findings

For the interim effectiveness review (EFR) described in AI 2012-08134-039:

- One acceptance criterion was to identify no self-assessment area for improvement related to performance monitoring being consistently performed.

- On December 2, 2014, as documented in Focused Area Self-Assessment (FASA) RA 2013-1147-004, the licensee declared that the subject EFR had failed, because although the site had transitioned to Exelon procedure ER-AA-2001, "Plant Health Committee," insufficient time had passed to fully implement that procedure and align plant-health-committee behaviors. Consequently, the licensee scheduled a follow-up interim effectiveness review via AI 2012-08134-085.

- On April 8, 2015, the licensee closed AI 2012-08134-085 with the note that they had completed the EFR using procedure PI-AA-125-1004, "Effectiveness Review Manual," and had determined that the EFR had been effective.

For the EFR described in AI 2012-08134-040:

- One acceptance criterion was for the Equipment Reliability Index (ERI) to be in the second quartile of industry performers or better.

- On June 25, 2015, the licensee determined that this EFR had failed, because the site's ERI of 75 was within the industry's fourth quartile. To address this failure, the licensee initiated CR 2015-08257. Via that CR, the licensee initiated AI 2012-08134-086 to require an EFR using procedure PI-AA-125-1004 with a replacement acceptance criterion and a due date of January 17, 2017.

(6) Assessment Results

AI 2012-08134-039 is closed because the licensee has completed the action.

AI 2012-08134-040 is closed because:

- the actions described in AI 2012-08134-086 are sufficient to fully address the actions originally described in AI 2012-08134-040,
- AI 2012-08134-086 is on schedule such that it will be completed on or before the due date,
- AI 2012-08134-086 involves only a final effectiveness review, and
- the licensee has demonstrated that they can successfully complete effectiveness reviews.

Because both AI 2012-08134-039 and AI 2012-08134-040 are closed, CAL item 7.d is closed.

Item 7.e: Electrical Equipment Qualification (EEQ)/High Energy Line Break (2013-0021)

Closed in IR 05000285/2014009 (ML14318A886), Section 4OA4.7.e.

Item 7.f: Safety System Functional Failures (2013-0056)

Closed in IR 05000285/2014009 (ML14318A886), Section 4OA4.7.f.

Item 7.g: Cables and Connections (2013-0033)

(1) Inspection Scope

As described in IR 05000285/2014009 (ML14318A886), Section 4OA4.7, the NRC previously inspected this PIIM Action Item but did not close it. Instead, that inspection produced several comments that are summarized below. Therefore, the team reviewed the implementation of the following action items:

- AI 2012-08617-011, Provide procedural expectations and guidance to electrical craft for handling aged electrical cables.
- AI 2012-03544-014, Develop a change management plan to implement the cables and connections program.
- AI 2012-08134-026, Execute plans to recover the EEQ and cable aging management programs.

- AI 2009-4216-020, Perform an effectiveness review of the strategy for maintaining dry those safety-related and important-to-safety cables susceptible to wetting.

- AI 2013-17441-001, Complete an assessment report on Cables and Connections Program.

- AI 2013-17441-002, Complete an assessment report on Verification of Material Condition of Medium & Low Voltage Safety Related Cables Submerged.

(2) Observations and Findings

- Inspection of AIs 2012-03544-014, 2013-17441-001, and 2013-17441-002 resulted in no notable observation.

- In IR 05000285/2014009, Section 4OA4.7.g, the NRC noted that the licensee had closed AI 2012-08617-011 without completing some of the required actions. The licensee initiated CR 2014-06939 to address this issue.

  The inspectors' review of CR 2014-06939 verified that the licensee had completed the actions they had previously missed.

- In IR 05000285/2014009, Section 4OA4.7.g, the NRC noted that the licensee had closed AI 2012-08134-026 without ensuring that the program owners for the Electrical Equipment Qualification program and the Cable Ageing Management program were properly qualified. In response to the team's observation, the licensee initiated CR 2014-9499 to address this issue.

  The inspectors' review of CR 2014-9499 verified that for both of the subject programs, both of the program owners and both of the backup program owners now are qualified.

- In IR 05000285/2014009, Section 4OA4.7.g, the NRC noted that the licensee had failed to accurately transcribe the action associated with AI 2009-4216-020 from its December 2, 2014, letter to the NRC (ADAMs Accession Number ML13336A785) into AI 2009-4216-020. Specifically, the inspectors noted that instead of completing an effectiveness review of the strategy for maintaining dry those safety-related and important-to-safety cables susceptible to wetting, the licensee had ensured that the long-term strategies for the subject cables and for keeping the subject manholes dry were in place. In response to the team's notation, the licensee initiated CR 2014-09009 to document that no action item had implemented the subject effectiveness review.

  The inspectors' review of AI 2014-09009-002 verified that the licensee had satisfactorily completed an effectiveness review of the subject strategy and that the subject review had concluded that effective implementation of the corrective actions for CR 2009-4216 had protected the subject safety related-cables from submergence.

(3) Assessment Results

The team considers CAL item 7.g closed.

.1.8. Programs

Item 8.a: Engineering Rigor (2013-0011)

Closed in IR 05000285/2014009 (ML14318A886), Section 4OA4.8.a.

Item 8.b: Equipment Safety Classification and Safety Related Equipment Maintenance (2013-0036)

Closed in IR 05000285/2014009 (ML14318A886), Section 4OA4.8.b.

Item 8.c: Electrical Bus Modifications and Maintenance (2013-0016)

Closed in IR 05000285/2014009 (ML14318A886), Section 4OA4.8.c.

Item 8.d: Deficiencies in Design and Implementation of Fundamental Regulatory Required Processes (2013-0007)

Closed in IR 05000285/2014009 (ML14318A886), Section 4OA4.8.d.

Item 8.e: Design Change 10 CFR 50.59 Practices (2013-0066)

Closed in IR 05000285/2014009 (ML14318A886), Section 4OA4.8.e.

Item 8.f: Piping Code and System Classification and Analysis (2013-0071)

(1) Inspection Scope

As described in IR 05000285/2014009 (ML14318A886), Section 4OA4.8.f, the NRC previously inspected the following action items with satisfactory results:

- AI 2012-07724-022, Review all Class I piping modifications since April 8, 1994, and document the effectiveness of the procedure for ensuring that thermal fatigue analysis was performed.

- AI 2012-07724-025, Review the United States of America Standard (USAS) B31.7 and ASME III code reconciliation and correct any code discrepancies.

During this inspection, the team reviewed the implementation of the following action items:

- AI 2012-07724-023, Provide calculations documenting thermal fatigue analysis on the Class I piping systems for primary plant sampling, reactor coolant gas vent, reactor coolant, safety injection, and waste disposal in accordance with United States of America Standards (USAS) B31.7 Draft 1968.

(2) Observations and Findings

IR 05000285/2014009 (ML14318A886), Section 4OA4.8.g stated that the licensee would not be able to complete AI 2012-07724-023 until the NRC completes its review of Licensee Amendment Request 14-04.

In response to Licensee Amendment Request 14-04, on August 10, 2015, the NRC issued Amendment No. 283 to the licensee's operating license. This amendment allows the licensee to perform pipe-stress analyses of non-reactor coolant system safety-related piping in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, 1980 Edition (no Addenda) as an alternative to the current Code of Record (i.e., USAS 831.7, 1968 (DRAFT) Edition).

Subsequently, the licensee assigned to AI 2012-07724-023 a due date of June 1, 2016.

(3) Assessment Results

Until the licensee completes AI 2012-07724-023, CAL Item 8.f remains open.

Item 8.g: Vendor Manual and Vendor Information Control Program (2013-0060)

Closed in IR 05000285/2014009 (ML14318A886), Section 4OA4.8.g.

Item 8.h: Safeguards Information Digital Storage Control (2013-0009)

Closed in IR 05000285/2014009 (ML14318A886), Section 4OA4.8.h.

Item 8.i: Operability Determination (2013-0107)

(1) Inspection Scope

As described in IR 05000285/2014009 (ML14318A886), Section 4OA4.8.i and in 05000285/2015008 (ML15071A115), NRC inspection teams reviewed the implementation of the following action items:

- AIs 2013-19752-001, -037; -038; -039; and -040; as part of the quarterly training curriculum review committee agenda, review operability determination performance indicators from the Engineering Assurance Group and the Operability Determination Quality Review Board. This will be a repeated action through 2014.

- AI 2013-19752-002, Conduct oral boards of all operators who make immediate operability determinations or screen condition reports.

- AI 2013-19752-005, Develop interim guidance for resolving unclear operability references. Include relating the use of prompt operability determinations with CAP, and current procedure direction, and its level of detail.

- AI 2013-19752-006, Formalize the Operability Determination Quality Review Board into a Fort Calhoun Station procedure.

- AI 2013-19752-007, Develop a method for ensuring that immediate operability determinations which fail the minimum Operability Determination Quality Review Board acceptance criterion (<70% unsupported operability determination) are re-performed by the On-Shift Crew.

- AI 2013-09494-036, Institute a change to NOD-QP-31 (or equivalent Exelon document) which incorporates clear and complete directions for completion of each applicable step of supporting process forms.

- AI 2013-19752-010, Develop specific guidance that directs personnel screening plant conditions or equipment failures to ensure actions are taken as required by the technical specifications (What to do when “this fails” procedure).

- AI 2013-19752-011, Screen the population of Fort Calhoun Station surveillances and relate these to the associated limiting condition for operations they support.

- AI 2013-19752-012, Review existing testing criteria, direction, or methodologies against industry norms.

- AI 2013-19752-013, Review material previously contained in Technical Data Book (TDB) VIII to ensure it resides in other documents that are clearly linked to the associated technical specification limiting condition for operations.

- AI 2013-19752-021, -022, -023, and -024; Conduct a common factors analysis of immediate operability determinations quarterly with results and actions approved by the MRC. Action will be on-going through 2014.

- AI 2013-19752-025, -026, -027, and -028; Conduct a common factors analysis of prompt operability determinations quarterly with results and actions approved by the MRC. Action will be on-going through 2014.

- AI 2013-19752-029, -030, -031, and -032; Present to Plant Review Committee (PRC) licensee event reports, results of operability determination performance metrics, and common factor analysis no less than semi-annually. Action will be on-going through 2014.

- AI 2013-19752-033, Immediate Operability Determination Engineering Assurance Group Assessment Performance Indicator of “green” with no more than one immediate operability determinations score greater than 2.0 per month (on average) for the period of June 1 through December 31, 2014.

- AI 2013-19752-034, Immediate Operability Determination Engineering Assurance Group Failure Rate Performance Indicator of “green” with no more than one immediate operability determinations failure per month (on average) for the period of June 1 through December 31, 2014.

- AI 2013-19752-035, Operability Determination Quality Review Board Operability Determination Performance Indicator of “green” with average Immediate Operability Determination (IOD)/Immediate Functionality Assessment (IFA) score > 90% per month for a period of June 1 through December 31, 2014.

- AI 2013-19752-036, Operability Determination Quality Review Board Operability Determination Failure Rate Indicator “green” with < 1 failure per month (on average) for a period of June 1 through December 31, 2014.

(2) Observations and Findings

Inspection of these action items resulted in no notable observation.

(3) Assessment Results

The team considers CAL item 8.i closed.

.1.9. Nuclear Oversight

Item 9.a: Nuclear Oversight Effectiveness (2013-0010)

Closed in IR 05000285/2014009 (ML14318A886), Section 4OA4.9.a.

.1.10. Transition to the Exelon Nuclear Management Model and Integration into the Exelon Nuclear Fleet

Item 10.a: Transition to the Exelon Nuclear Management Model and Integration into the Exelon Nuclear Fleet (2013-0077)

Closed in IR 05000285/2014009 (ML14318A886), Section 4OA4.10.a

.2 Actions detailed in the Flooding Recovery Action Plan

(1) Inspection Scope

As described in IR 05000285/2014004 (ML14317A777), Section 4OA4.5, NRC inspectors reviewed the implementation of the following action items:

- Action Request (AR) 49712-11, Item 4.4.3.1: Gather flood response lessons learned through CR reviews to determine if procedure or strategy changes should be implemented.

- AR 49712-13, Item 4.4.3.3: Implement procedure and strategy changes as indicated by the lessons learned review conducted above.

The team reviewed the implementation of the following action items:

- AI 2011-8950-025, Item 1.2.3.21: Inspect tank and equipment on DI tank for damage

- AI 2012-16067-002, Item 1.2.3.82: [Independent Spent Fuel Storage Installation; ISFSI] haul route load test

- AI 2012-03366-002, Item 4.4.3.2: Review Flood Design Basis and determine if the 2011 flood event provides additional information that should drive design basis changes

(2) Observations and Findings

IR 05000285/2014004 (ML14317A777), Section 4OA4.5, states that the NRC had no notable observation about AR 49712-11 or AR 49712-13.

Concerning AI 2011-8950-025 and AI 2012-03366-002, the team had no notable observation.

Regarding AI 2012-16067-002:

- This action item described a “proof test” of the haul route with the actual Transfer Trailer and the tow vehicle used for moving the ISFSI casks and with a combined weight greater than 110 percent of the Transfer Trailer and tow vehicle’s combined weight with a dry cask and irradiated fuel.

- On September 26, 2014, and under work request 216509, the licensee completed a haul route test using a forklift that weighed approximately 37 percent of the weight of the Transfer Trailer and tow vehicle with a dry cask and irradiated fuel. The licensee noted that the load traveled at a walking pace with two engineers continuously monitoring the contact of the forklift’s tires with the haul route. The licensee reported that the engineers did not observe any sign of washout along the haul route, and did not note any haul route surface reaction or degradation of any kind.

  Also, on July 30, 2015, an NRC inspector walked down the haul route and did not note any degradation of any kind in either the haul route or the areas immediately adjacent to the haul route.

- The “Close Comments” associated with AI 2012-16067-002 say, in part:

  > “CR 2012-16067-003 will update RE-RR-DFS-003, “Loaded DSC/TSC From Auxiliary Building to ISFSI Operations”, to ensure the appropriate haul route proofing takes place prior to the next FCS Dry Fuel Storage Campaign. The current estimated scheduled start date for the next FCS Dry Fuel Storage Campaign is 2018.”

- The inspectors reviewed CR 2012-16067-003, and verified that the licensee has updated RE-RR-DFS-003 as described above; the result was Revision 11 to that procedure.

(3) Assessment Results

The team closed AI 2011-8950-025, AI 2012-03366-002, AR 49712-11, and AR 49712-13.

Although the licensee has not yet completed a load test as originally described in AI 2012-16067-002 and as currently described in AI 2012-16067-003,

- licensee and NRC visual examinations have not identified any evidence of haul path degradation, and

- Revision 11 to procedure RE-RR-DFS-003 indicates that completing the load test is a scheduled and required element of the licensee's next dry fuel storage campaign.

Therefore, the "Actions detailed in the Flooding Recovery Action Plan" are closed.

.3 <u>Actions Associated with Auxiliary Steam Piping in the Auxiliary Building</u>

(1) Inspection Scope

The team reviewed implementation of the following action items:

- AR 49722-33, 4.5.1.14: Perform [High Energy Line Break] analysis of Auxiliary Steam piping in the auxiliary building.

- AR 61005, 4.5.1.15: Implement resolution of Auxiliary Steam piping in the auxiliary building.

(2) Observations and Findings

The licensee closed AR 49722-33 to AR 61005, and completed associated action item 4.5.1.14 via AI 2011-5244-015, which the licensee characterized as,

> "Perform analysis or calculation and implement required activities which fully qualifies the [Auxiliary] Steam and Condensate Return lines in the Intake Structure."

However, their actions addressed more than just the Intake Structure. Specifically, as described in an attachment to AI 2011-5244-015,

- In calculation FC 08353, "Environmental Effects from an Auxiliary Steam & Condensate Return Line Crack in the Intake Structure," the licensee determined that the plant's designers had routed Auxiliary Steam and Condensate Return lines through the Service Building to the Intake Structure. The licensee postulated a High Energy Line Crack (HELC) in each of those lines, and calculated the resulting environmental effects. They concluded that the environmental limits in the Raw Water Pump Rooms containing the Raw Water Pumps (AC-10A/B/C/D) and Strainers (AC-12A/B) (the only components within the Intake Structure to which Electrical Environmental Qualification (EEQ) applies) are not exceeded.

  Analysis of the Intake Structure was not part of the licensee's commitment, but was part of their response to that commitment.

- In calculation FC 08350," Environmental Effects from Condensate Return Line Cracks in the Auxiliary Building," the licensee determined that Auxiliary Steam lines are routed through rooms 69, 81, and 82. Regarding those rooms:

    - The licensee postulated a HELC in Auxiliary Steam piping in room 69, and determined that the resulting environment would remain mild.

    - The licensee did not evaluate room 81 based on previous analyses of Main Steam Line Breaks (MSLBs) that occur within room 81, because the MSLBs are bounding for HELCs with respect to temperature, pressure, and humidity. The licensee documented their analyses of MSLBs in design analysis FC07889.

    - The licensee did not evaluate room 82 because that room does not contain any credited safe-shutdown equipment, but noted that according to design analysis FC07889, room 82 would experience a harsh environment due to a postulated MSLB in Room 81.

    - The licensee evaluated dynamic and wetting effects of Auxiliary Steam HELCs in analysis EA 13-037, discussed further below.

- The unit's Controlled Access Area Ventilation air supply housing VA-17 is located in room 69, and the unit's Uncontrolled Access Area Ventilation air supply housing VA-19 is located in room 81. A HELC within one of those housings could result in distribution of increased heat load and humidity to other areas of the plant via the ventilation system. In calculation FC 08462, the licensee further evaluated the impact of a HELC in AS piping within both of those air supply housings, and concluded that the effect of the HELC on airflow within VA-17 and VA-19 would be negligible. The negligible effect of the HELC on airflow and the control and capacity of heating coils VA-36A(B) and VA-43A(B) prompted the licensee to conclude that a postulated HELC within VA-17 and VA-19 would not have a significant effect on either the Auxiliary Building Controlled Access Area or Uncontrolled Access Area.

- In calculation FC 08384, "Environmental Effects from Condensate Return Line Cracks in the Auxiliary Building," the licensee determined that Condensate Return lines exist in rooms 4, 6, 19, 26, 30, 31, 56, 69, 81, and 82. The licensee did not evaluate rooms 81 and 82 because of the bounding environment due to a postulated MSLB in Room 81, as discussed above. The licensee postulated a HELC in Condensate Return piping in rooms 4, 6, 19, 26, 30, 31, 56, and 69 and analyzed the resulting environments with respect to EEQ limits. The licensee determined that after the postulated HELC,

    - rooms 4, 6, 19, 26, 56, and 69 remain mild environments, and

    - rooms 30 and 31 would become harsh environments due to relative humidity.

The licensee determined that the harsh environments in rooms 30 and 31 were acceptable because those rooms contain no credited safe-shutdown equipment.

The licensee evaluated dynamic and wetting effects of CR cracks in Rooms 4, 6, 19, 26, 30, 31, 56, 69, 81, and 82 in design analysis EA 13-037 (see below).

- In analysis EA 13-037, "Auxiliary Steam and Condensate Return High Energy Line Break Dynamic and Wetting Effects in the Auxiliary Building," the licensee concluded that no AS or CR HELC in the Auxiliary Building would produce dynamic wetting effects that would adversely affect the safe shutdown of the station. However, the licensee identified the need for and implemented these design changes:

    - EC 64326: Because room 19 contains the station air compressors and safety related Auxiliary Feedwater (AFW) pumps and because EA 13-037 determined that water may drip on and thus fail the AFW pumps as a result of a HELC. The licensee re-routed Condensate Return piping to traverse through the Turbine Building instead of room 19.

      The inspectors walked down the previous and current Condensate Return piping routes.

    - EC 62956: To preclude postulating and evaluating a HELC in the Emergency Diesel Generator (EDG) rooms, the licensee cut and capped the Auxiliary Steam supply lines to those rooms. The licensee also installed electric heaters to replace the steam heaters to which AS had previously supplied steam.

      The inspectors walked down the previous Auxiliary Steam steam line routes, and observed both the subject caps on the steam lines and the subject electric heaters.

The licensee completed action item 4.5.1.15 "Implement resolution of Auxiliary Steam piping in the auxiliary building," by implementing the engineering changes discussed above. As noted above, NRC inspectors walked down the implemented changes.

(3) Assessment Results

The "Actions Associated with Auxiliary Steam Piping in the Auxiliary Building" are closed.

.4 <u>Actions Associated with Containment Internal Structures</u>

(1) Inspection Scope

The team reviewed the implementation of the following action items:

- “Evaluate the structural design margin for the containment internal structures, and reactor cavity and compartments, and resolve any deficiencies in accordance with its corrective action program (CAP).”

- “Regarding Beam 22A and Beam 22B in the containment internal structures, resolve any deficiencies in accordance with the CAP.”

- “Regarding the reactor head stand, prior to the next use of the reactor head stand, OPPD will evaluate the structural design margin for the head stand and resolve any deficiencies in accordance with the CAP.”

(2) Observations and Findings

The licensee addressed these issues in accordance with the CAP as shown below:

Issue	Action Item(s)
“Evaluate the structural design margin for the containment internal structures, and reactor cavity and compartments…”	CR 2012-04392-014 AI, Item 4 & CR 2012-04392-045 AI
“Regarding Beam 22A and Beam 22B in the containment internal structures, resolve any deficiencies…”	CR 2012-04392-048 AI, Item 4
“Regarding the reactor head stand, prior to the next use of the reactor head stand, OPPD will evaluate the structural design margin for the head stand and resolve any deficiencies…”	CR 2012-04392-049 AI, Item 4

The discussions below summarize the results of the inspectors’ reviews of these action items.

<u>CR 2012-04392-014 AI:</u> The licensee characterized this action item as,

> “Resolve discrepancies for the Internal Structure of Containment, including any needed plant modifications. Implement design modifications to restore the Containment Internal Structure (CIS) to within its design basis requirements.”

The licensee’s analysis of this issue determined that a LAR was warranted to allow alternate provisions used in the analyses of the internal structures. Accordingly, on December 23, 2015, the licensee submitted LAR 15-0142 (ML15363A042). To allow time for the NRC to respond to that LAR, the licensee set the due date for CR 2012-04392-014 AI to December 15, 2016.

<u>CR 2012-04392-045 AI:</u> The licensee characterized this action item as,

> “Implement design modifications to restore the Reactor Cavity and Compartments (RC&C) to within its design basis requirements during the next refueling outage (RFO 27).”

The licensee set the date for this AI also to December 15, 2016, to allow time to resolve LAR 15-03 and its supplement.

CR 2012-04392-048 AI: The licensee characterized this action item as,

> "Resolve any identified discrepancies concerning Beams 22A and 22B in the Containment Internal Structure."

As described in OPPD letter LIC-15-0042, the licensee revised this commitment to,

> "Regarding Beam 22A and Beam 22B in the containment internal structures, prior to resuming power operation following Fort Calhoun Station Unit 1 Refueling Outage 28, OPPD will restore full structural design margin as described in the Fort Calhoun Station licensing basis."

The licensee extended the due date for this AI to December 15, 2016, due to the complexity of design and the licensee's inability to obtain the necessary pre-engineered components in sufficient time to complete the activities.

CR 2012-04392-049 AI: The licensee prepared and implemented:

- Calculation FC 08389, "New RVH Support Frame Analysis and Design," Revision 1, and

- Engineering Change EC 58237, "Containment Internal Structure RVH Stand Area,"

The inspectors determined that OPPD evaluated the structural design margin for the reactor head stand by completing and implementing the analyses associated with FC08389 and EC 58237.

(3) Assessment Results

- CRs 2012-04392-014 AI, 2012-04392-045 AI, and 2012-04392-048 AI remain open, because the associated actions are not complete.

- CR 2012-04392-049 AI is closed.

.5 Action Items Within the CAL that Remain Open

Sections 4OA4.1 through 4OA4.4 above identified several action items that remain open after this inspection. To summarize, the table below lists those items. In this table, the "Ref." column refers to the 4OA4 section of this report that discusses each item. The table is sorted by Due Date.

Ref.	AI Number	Description	Due Date
4OA4.1.4 (Item 4.a)	2013-05570-049	CAPR-3- Modify the Engineering Support Personnel Training (ESPT) initial and continuing training programs to incorporate CAPR-1 and CAPR-2. Training shall include items 1, 2 and 3 from CAPRs 1 and 2 to address the identification of design and licensing bases, record types that are included, and the method of retrieval	See note 1
4OA4.1.8 (Item 8.f)	2012-07724-023	Provide calculations documenting thermal fatigue analysis on the Class I piping systems for primary plant sampling, reactor coolant gas vent, reactor coolant, safety injection, and waste disposal in accordance with USAS B31.7 Draft 1968	June 1, 2016
4OA4.4	2012-04392-014 (item 4) & 2012-04392-045	Evaluate the structural design margin for the containment internal structures, and reactor cavity and compartments, and resolve any deficiencies in accordance with its corrective action program (CAP)	Dec. 15, 2016
4OA4.4	2012-04392-048	Regarding Beam 22A and Beam 22B in the containment internal structures, resolve any deficiencies in accordance with the CAP	Dec. 15, 2016
4OA4.1.4	2013-05570-026	Identify and define the current licensing bases and assure licensing bases documentation remains current, accurate, complete, and retrievable.	July 20, 2018
4OA4.1.4	2013-05570-076	Identify and define the design bases and assure design bases documentation remains current, accurate, complete, and retrievable.	July 20, 2018
4OA4.1.4	2013-05570-092	Complete Phase 3 of the Key Calculation Project. Phase 3 consists of revising any deficient critical calculation or engineering analysis identified from Phase 2, as needed.	July 20, 2018

4OA4.1.4	2013-05570-093	Validate the design and licensing basis has been translated into plant operation by verifying that the operation, surveillance, and maintenance of the safety-related components do not compromise the design and licensing basis.	July 20, 2018

Note 1: As discussed in section 4OA4.1.4, the NRC deferred determining the adequacy of this action item until a later inspection.

4OA5 Other Activities

.1 <u>(Closed) Unresolved Item (URI) 05000285/2013008-27, "Continuous Monitoring Capability of Post Accident Main Steam Radiation Monitor RM-064"</u>

The NRC opened this unresolved item because NRC inspectors had questioned the capability of post-accident radiation monitor RM-064 to provide representative measurements due to the system configuration. Specifically, the inspectors suspected that the system configuration could represent a failure to ensure continuous effluent monitoring of the main steam lines following a steam-generator-tube-rupture accident concurrent with a reactor trip and a loss of offsite power.

RM-064 is the licensee's post-accident gaseous effluent release monitor corresponding to NUREG 0707 Section II.F.1.1 "High Range Noble Gas Effluent Monitor." That monitor provides plant operators and emergency planning agencies with information on plant releases of noble gases during and following an accident.

Under CR 2013-04442, the licensee performed an engineering technical evaluation that was based on existing radiological analysis Calculation FC06820 used to analyze the steam generator accident. In this technical evaluation, the licensee removed many of the conservative assumptions that they had included in Calculation FC06820. Based on this evaluation and engineering judgment, the licensee determined that the mixing would be sufficient and the concentration would be adequate to provide a representative radiation measurement. However, the inspectors raised several questions about this conclusion, and, in response, the licensee initiated Condition Reports 2013-04442, 2013-05515, 2013-06267, and 2013-10507 to address those questions.

Subsequently, on August 20, 2013, the licensee completed engineering analysis EA 13-021 to document their determination that RM-064 has a constant mass flow rate, the sample is uniformly mixed, sufficient pressure exists to drive the sample through the detector volume, and the radioactive concentrations measured would be conservative at maximum safety valve flow rate. In addition, the licensee determined the Geiger Mueller detector would have a conservative factor of 1.8 – 2.0. The licensee determined a correction factor of 0.825 was necessary to adjust the conservatism and create a more-representative sample measurement. The licensee revised the RM-064 primary source and electronic calibration procedures to incorporate the new correction factor.

The inspectors reviewed the associated condition reports, engineering evaluations, and drawings, and walked down the RM-064 monitoring system. The inspectors compared the licensee's assumptions and calculations with industry standards (ANSI N13.1 and ASTM D1066), licensee technical specifications, USAR, and regulations. The inspectors reviewed the licensee's postulated accident assumptions and offsite dose calculations with respect to a steam generator tube rupture concurrent with a reactor trip and loss-of-offsite-power event. The inspector concluded that the post-accident radiation monitor RM-064 would provide representative high range gaseous effluent measurements during a design-basis steam-generator-tube-rupture event concurrent with a reactor trip and a loss-of-offsite-power event.

.2 (Closed) VIO 2013-011-01, "Continued Failure to Classify Intake Structure Sluice Gates as Safety Class 3"

The NRC issued this cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," because after the NRC had issued the corresponding non-cited violation in inspection report 05000285/2012002 (ML13045B055), the licensee had remained in noncompliance until the inspection documented in inspection report 05000285/2013011 (ML13070A399). To address this violation, the licensee initiated CR 2013-05620.

The inspectors reviewed the Apparent-Cause Evaluation documented in CR 2013-05620 and the corresponding corrective actions. Those corrective actions included upgrading the sluice gate classification to "Limited Critical Quality Element" (with NRC approval via Amendment 282), installing safety-related floodwater inlet valves, and updating the USAR and station procedures accordingly. The inspectors determined that through these actions, the licensee restored compliance to 10 CFR 50, Appendix B, Criterion III. This violation is closed.

.3 (Closed) Licensee Event Report (LER) 2012-021-0, "HCV-2987, HPSI Alternate Header Isolation Valve"

On February 9, 2013, the licensee submitted this LER after identifying that valve HCV-2987, High Pressure Safety Injection Alternate Header Isolation, would not have been able to fulfill its design safety function because of higher-than-acceptable friction in the valve packing. The licensee's associated root-cause-analysis evaluation, documented in CR-2012-01601-017, stated that in 2008, the licensee had completed a "FlowScan" diagnostic evaluation of HCV-2987 that had showed abnormally high stem-packing friction, but at the time, the licensee did not compare the value to approved calculations and did not take actions to correct the issue. Consequently, HCV-2987 remained inoperable from 2008 until July 16, 2013, when the licensee replaced the existing valve stem packing with a new configuration having a lower packing-friction coefficient. The subject report also stated that the root cause of the extended inoperability of HCV-2987 was that the licensee had failed to compare FlowScan data with approved calculations and take corrective actions. One of their corrective actions to prevent recurrence, which they designated as corrective action to prevent recurrence number one (CAPR-1), was to write and implement a new procedure characterized as ER-FC- 410-AD-SETPOINT, "Air-Operated Valve Setpoint Control," Revision 0. This procedure established setpoint control for all Category-I air-operated valves, and required, in part, that the licensee review all diagnostic test results for compliance with the setpoint criteria for all diagnostic tests performed.

The licensee discovered this condition while reviewing various design calculations as part of an extent-of-condition review for the condition documented in CR 2011-9945-006 AI, which was that the licensee had discovered non-conservative pressure-regulator settings for a different valve.

The inspectors considered that the licensee’s root-cause-analysis evaluation and CAPR-1 revealed that until the licensee implemented ER-FC- 410-AD-SETPOINT, Revision 0, the licensee had failed to establish a procedure that required the licensee to review all diagnostic test results for compliance with the setpoint criteria for all diagnostic tests performed. The licensee’s failure to establish that procedure was a performance deficiency that constituted a violation of Technical Specification 5.8.1. The inspectors dispositioned that violation as the licensee-identified violation documented in section 4OA7 below.

LER 2012-021-0 is closed.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On January 14, 2016, the inspectors presented the inspection results to Mr. Todd Tierney and other members of the licensee staff. The licensee acknowledged the results presented. The licensee confirmed that the inspectors had returned or destroyed any proprietary information reviewed.

4OA7 Licensee-Identified Violations

Listed below is one violation of very low safety significance (Green) that was identified by the licensee and is a violation of NRC requirements, which meets the criteria of the NRC Enforcement Policy for being dispositioned as a non-cited violation.

- Technical Specification 5.8.1 requires in part, that procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2. That appendix states, in part, that maintenance that can affect the performance of safety-related equipment should be performed in accordance with written procedures appropriate to the circumstances. Contrary to the above, maintenance that can affect the performance of safety-related equipment was not performed in accordance with written procedures appropriate to the circumstances. Specifically, maintenance that can affect the performance of safety-related valves was not performed in accordance with a procedure that required the licensee to review all diagnostic test results for compliance with the setpoint criteria for all diagnostic tests performed.

  As described in CR-2012-01601-017, the licensee restored compliance by writing and implementing procedure ER-FC- 410-AD-SETPOINT, “Air-Operated Valve Setpoint Control,” Revision 0. This procedure requires, in part, that the licensee review all diagnostic test results for compliance with the setpoint criteria for all diagnostic tests performed.

The licensee's failure to complete maintenance that can affect the performance of safety-related valves in accordance with written procedures appropriate to the circumstances was a performance deficiency that is more-than-minor because it adversely affected the Procedure Quality attribute of the Mitigating Cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, this performance deficiency resulted in valve HCV-2987, High Pressure Safety Injection Alternate Header Isolation, being not able to fulfill its design safety function from February, 2013, through July, 2013. Using Manual Chapter 0609, Attachment 4, "Initial Characterization of Findings," dated June 19, 2012, the inspectors determined that the finding should be processed through Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 1, 2012. Using Appendix A, Exhibit 2, "Mitigating Systems Screening Questions," the inspectors determined that the finding was not a design or qualification deficiency but represented a loss of train function for greater than the outage time allowed by Technical Specifications. Therefore, a Region IV senior reactor analyst performed a detailed risk evaluation in accordance with Manual Chapter 0609, Appendix A, Section 6.0, "Detailed Risk Evaluation." The analyst determined that the condition of valve HCV-2987 inoperability would affect only the plant's response to a large-break loss-of-coolant accident followed by the failure of the instrument air system. The analyst calculated the initiating-event frequency to be 2.63 x 10^{-10} /year. Also, the analyst determined that the finding did not affect external initiator risk and would not involve a significant increase in the risk of a large, early release of radiation. Therefore, this violation has very low (Green) safety significance.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

R. Beck, Manager Chemistry
B. Blome, Manager Site Regulatory Assurance
C. Cameron, Supervisor, Regulatory Compliance
J. Cate ,Manager, Engineering Special Programs
M. Frans, Manager, Special Assignment
C. Gotschall, Corrective Action Program Coordinator
E. Matzke, Senior Regulatory Assurance Engineer
T. Tierney, Plant Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Closed

05000285/2013008-27	URI	Continuous Monitoring Capability of Post-Accident Main Steam Radiation Monitor RM-064
05000285/2013011-01	VIO	Continued Failure to Classify Intake Structure Sluice Gates as Safety Class 3
05000285/2012-021-0	LER	HCV-2987, [High Pressure Safety Injection] Alternate Header Isolation Valve

LIST OF DOCUMENTS REVIEWED

4OA4 Confirmatory Action Letter (CAL) Inspection Activities

Procedures

Number	Title	Revision
AOP-01	Acts of Nature	46
CC-AA-201	Plant Barrier Control Program	10
CC-FC-309-1011-AD-MEI-8	Piping Design	0
PED-CSS-3	Procuring, Applying and Inspecting Protective Coatings Inside Reactor Containment Building	7
RE-RR-DFS-003	Loaded DSC/TSC From Auxiliary Building to ISFSI Operations	11

Condition Reports (CRs)

2012-03366	2012-04392	2012-07724	2014-06939	2014-08532
2014-9499				

Action Items (AIs)

2012-04392-014	2012-04392-034	2012-04392-045	2012-04392-048	2012-04392-049

Action Requests (ARs)

49712	61014

Calculations

Number	Title	Revision
FC08350	Environmental Effects from an Auxiliary Steam Line Crack in Room 69	13
FC08384	Environmental Effects from Condensate Return Line Cracks in the Auxiliary Building	11

Other Documents

Number	Title	Revision/Date
50.59 Screening Number 14-027	Containment Internal Structure RVH Stand Area	0

11405-S-19	Reactor Plant Operating Fl. Plan El. 1045’-0” and 1060’-0” Outline	20
11405-S-41	Reactor Plant Operating Fl. Plan El. 1045’-0” & 1060’-0” Reinforcement – Sheet 1	4
14Q4249-CAL-001	New RVH Support Frame Analysis and Design	2
14Q4249-CAL-001	New RVH Support Frame Analysis and Design	7
Amendment No. 283 to the site’s operating license	FORT CALHOUN STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: ADOPT AMERICAN SOCIETY OF MECHANICAL ENGINEERS BOILER AND PRESSURE VESSEL CODE, SECTION III, AS AN ALTERNATIVE TO THE CURRENT CODE OF RECORD (TAC NO. MF4160)	August 10, 2015
CR 2012-04392-014	Request for MRC Approval of Due Date Extension	April 30, 2015
CR 2012-04392-045	Request for MRC Approval Due Date Extension	April 30, 2015
EC 58236	Containment Internal Structure Column Interferences (East Side)	0
EC 58237	Containment Internal Structure RVH Stand Area	0
EC 58237	Containment Internal Structure RVH Stand Area	3A
FC08189	Evaluation of Operability Containment Internal Structures (CIS) (SA Calc No. 12Q4070-CAL-009, R6)	3A
FC08389	New RVH Support Frame Analysis and Design	0
FC08389	New RVH Support Frame Analysis and Design	1
LIC-15-0042	Revision to Post-Restart CAL Commitment for Containment Internal Structure Beam 22A and Beam 22B	April 30, 2015
LIC-15-0142	Supplement of License Amendment Request 15-03; Revise Current Licensing Basis to Use ACI Ultimate Strength Requirements	December 23, 2015
SO-G-21	Modification Control	99

4OA5 Other Activities

Condition Reports (CRs)

2013-04442	2013-05515	2013-06267	2013-10507

Other Documents

Number	Title	Revision/Date
Calculation FC 06820	Site Boundary and Control Room Doses following a Steam Generator Tube Rupture Accident Using Alternate Source Term	1
Engineering Analysis 13-021	Determination of Representative Sampling of the Post-Accident Main Steam Line Monitor RM-064	1

Miscellaneous Documents

Number	Title	Revision/Date
ANSI N13.1	Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities	1999
ASMT D1066	Standard Practice for Sampling Steam	11
NUREG 0707	Clarification of TMI Action Plan Requirements	1980
USAR 11.2.3.11	Radioactive Waste and Radiation Protection and Monitoring – Radiation Protection and Monitoring – Post Accident Main Steam Line Monitor	20
USAR 11.3	Radioactive Waste and Radiation Protection and Monitoring – Radiological Effluent Requirements	5
USAR 14.1.4	Safety Analysis – General – Radiation Monitoring During Accident Conditions	23
USAR 14.14	Safety Analysis - Steam Generator Tube Rupture Accident	15