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

September 24, 2015

Louis P. Cortopassi, Site 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 REACTIVE INSPECTION REPORT 05000285/2015011

Dear Mr. Cortopassi:

On August 14, 2015, the U.S. Nuclear Regulatory Commission (NRC) completed a Special Inspection at the Fort Calhoun Station, to evaluate the facts and circumstances surrounding a failure of two auxiliary feedwater system valves. Based upon the risk and deterministic criteria specified in NRC Management Directive 8.3, "NRC Incident Investigation Program," the NRC initiated a Special Inspection in accordance with Inspection Procedure 93812, "Special Inspection." The basis for initiating the special inspection and the focus areas for review are detailed in the enclosed Special Inspection Charter. The determination that the inspection would be conducted was made by the NRC on June 22, 2015, and the onsite inspection started on June 29, 2015. The enclosed report documents the inspection findings that were discussed on August 14, 2015, with members of your staff. Inspectors documented the results of this inspection in the enclosed inspection report.

NRC Inspectors documented three findings of very low safety significance (Green) in this report. All of these findings involved violations of NRC requirements. The NRC is treating these violations as non-cited violations consistent with Section 2.3.2.a of the NRC Enforcement Policy.

If you contest the violations or significance of these non-cited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region IV; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC resident inspector at the Fort Calhoun Station.

If you disagree with a cross-cutting aspect assignment in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region IV; and the NRC resident inspector at the Fort Calhoun Station.

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

Sincerely,

/RA/

Geoffrey Miller, Chief
Project Branch D
Division of Reactor Projects

Docket No.: 50-285
License No.: DPR-40

Enclosures:
Inspection Report 05000285/2015011
w/ Attachments:
1. Supplemental Information
2. Special Inspection Charter

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Letter to L. Cortopassi from G. Miller dated September 24, 2015

SUBJECT: FORT CALHOUN STATION – NRC SPECIAL INSPECTION REPORT 05000285/2015011

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OEWEB Resource@nrc.gov
OEWEB Resource (Sue.Bogle@nrc.gov)
Technical Support Assistant (Loretta.Williams@nrc.gov)
RIV/ETA: OEDO (Cindy.Rosales-Cooper@nrc.gov)
RIV RSLO (Bill.Maier@nrc.gov)
ACES (R4Enforcement.Resource@nrc.gov)
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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	05000285
License:	DPR-40
Report:	05000285/2015011
Licensee:	Omaha Public Power District
Facility:	Fort Calhoun Station
Location:	9610 Power Lane Blair, NE 68008
Dates:	June 29 through August 14, 2015
Inspectors:	R. Kumana, Resident Inspector M. Langelier, P.E., Resident Inspector
Approved By:	Geoffrey B. Miller Chief, Project Branch D Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000285/2015011; 06/29/2015 – 08/14/2015; Fort Calhoun Station, Special Inspection to Evaluate Causes of the Failures of Two Auxiliary Feedwater System Valves.

The report covered one week of onsite inspection and in-office review through August 14, 2015, by inspectors from the NRC's Region IV office. Three Green findings were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

Cornerstone: Mitigating Systems

- Green. The inspectors reviewed a Green, self-revealing, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to evaluate the suitability of materials utilized during the design review process. Specifically, the licensee failed to identify during the design review process that replacement valve internal seal materials for the steam generator auxiliary feed containment isolation valves would not be suitable for high temperature conditions that the valves would experience in service, and as a result, caused both trains of the safety-related auxiliary feedwater system to become inoperable during hot standby conditions. The licensee entered this issue into their corrective action program as Condition Report CR-2015-07564 and replaced the valve internals with material that had been previously installed in valves HCV-1107A and HCV-1108A before the modification.

  The inspectors determined that the licensee's failure to evaluate the suitability of the materials used during the design review process for the steam generator auxiliary feed containment isolation valves was a performance deficiency. The performance deficiency is more than minor, and therefore a finding, because it is associated with the equipment performance attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Specifically, the licensee's failure to properly evaluate the suitability of CTFE for use in the steam generator auxiliary feed containment isolation valves led to the failure of HCV-1107A and HCV-1108A and rendered both safety-related trains of auxiliary feedwater inoperable.

  Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process for Findings At-Power", dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions", the inspectors determined that the finding required a detailed risk evaluation since the finding represented a loss of system and/or function. A Region IV senior reactor analyst performed the detailed risk evaluation in accordance with Appendix A, Section 6.0, "Detailed Risk Evaluation." The detailed risk evaluation result is a finding of very low safety significance (Green). The calculated change in core damage frequency of 2.3×10^{-7} was dominated by a loss of offsite power; common cause failure of the auxiliary feedwater discharge air-operated valves; failure of diesel-driven auxiliary feedwater pump FW-54; failure of the feed and bleed operation; and failure of operators to manually override a steam generator isolation signal and establish a flowpath for the main feedwater system. The analyst determined that the finding did not involve a significant impact to external initiators because of the short exposure time, or a significant increase in the risk of a large, early release of radiation. The finding has an operating experience cross-cutting aspect in the problem identification and resolution cross-cutting area since the organization did not systematically and effectively collect, evaluate, and implement relevant internal and external operating experience in a timely manner. Specifically, readily available internal operating experience on the high temperature conditions that valves HCV-1107A and HCV-1108A experienced during normal operations was not utilized during the design change process [P.5]. (Section 4OA5.2)

- Green. The inspectors identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's failure to follow the operability determination procedure. Specifically, the licensee failed to establish a valid technical basis for operability of auxiliary feed containment isolation valves HCV-1107A and HCV-1108A. Following the valves' failure on June 5, the licensee replaced the failed valve elastomers with new PTFE seals and nitrile O-rings. The licensee then performed an operability evaluation that considered the effect of high temperatures from a main steam line break on the valve elastomers. The inspectors found that the evaluation was not sufficient because it did not determine that the new O-rings would function under all potential temperature conditions and did not consider the function of the other valve components. The licensee entered these issues in their corrective action program as Condition Report CR-2015-08362 and revised their operability evaluation.

  The licensee's failure to follow the operability determination procedure was a performance deficiency. The performance deficiency is more than minor, and therefore a finding, because it is associated with the equipment performance attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Specifically, the licensee failed to sufficiently address the capability of the steam generator auxiliary feed containment isolation valves HCV-1107A and HCV-1108A to perform their safety function, requiring significant further analysis to demonstrate operability. Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," and Appendix A, Exhibit 2, "Mitigating Systems Screening Questions," the finding was determined to be of very low safety significance (Green) because although the finding was a deficiency affecting design or qualification, but the mitigating structure, system or component maintained its operability. The finding has a consistent process cross-cutting aspect in the human performance cross-cutting area since the organization did not use a consistent, systematic approach to make decisions and incorporate risk insights appropriately. Specifically, the licensee failed to re-evaluate the operability decision when new information on the conditions and susceptibility affecting valves HCV-1107A and HCV-1108A during normal operations was available [H.13]. (Section 4OA5.5)

- Green. The inspectors identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for the licensee's failure to correct a condition adverse to quality. Specifically, the licensee failed to take corrective actions after identifying that the steam generator auxiliary feed containment isolation valves were not rated for the maximum temperature they would experience in service. The inspectors determined that on February 2, 2015, an NRC inspector questioned the licensee whether valves HCV-1107A and HCV-1108A were adequately designed for containment temperatures. The licensee determined that the design specification for the valves was 180°F, and the containment temperature following a main steam line break was evaluated to be 374°F. The fact that the valve was not designed for the most limiting conditions was a non-conforming condition of a safety related component, and was a condition adverse to quality. However, the licensee did not initiate a condition report to resolve and correct the condition. Additionally, the inspectors determined that in 2002, the licensee initiated Condition Report CR-2002-02124 after identifying elevated temperatures in the auxiliary feedwater piping. This condition report documented that the design specification for the two valves was 180°F and had been exceeded in service. Although the condition report description recommended modifying the design of the valves, the licensee did not take actions to correct the condition. In both of these instances, the licensee recognized that the valve design temperature was not adequate for its application, but did not take action to resolve the discrepancy. The inspectors determined that although the inadequate design was a non-conforming condition, the valves were not inoperable until the licensee installed inappropriate elastomer material during the 2015 refueling outage as a result of inadequate design control. The licensee entered the failure to identify and correct the non-conforming design in their corrective action program as Condition Report CR-2015-08523.

The licensee's failure to take corrective action for a non-conforming condition was a performance deficiency. The performance deficiency is more than minor, and therefore a finding, because it is associated with the design control attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Specifically, the licensee failed to take corrective actions to ensure an adequate design for the steam generator auxiliary feed containment isolation valves HCV-1107A and HCV-1108A. Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," and Appendix A, Exhibit 2, "Mitigating Systems Screening Questions," the finding was determined to be of very low safety significance (Green) because although the finding was a deficiency affecting the design or qualification of a mitigating system, structure, or component, the system, structure, or component maintained its operability. The finding has a basis for decisions cross-cutting aspect in the human performance cross-cutting area since leaders and individuals did not verify their understanding or question the basis of decisions. Specifically, the licensee failed to understand the potential significance of the non-conforming design of the valves and the basis for not taking corrective actions [H.10]. (Section 4OA5.6)

REPORT DETAILS

<u>Basis for Special Inspection</u>

On June 5, 2015, while in Mode 3, Fort Calhoun Station performed auxiliary feedwater surveillance testing prior to startup from a refueling outage. During the testing, auxiliary feed containment isolation valve HCV-1107A failed to open on demand. This valve provides the auxiliary feedwater injection path to steam generator RC-2A for both the turbine and motor driven auxiliary feedwater pumps. Because this valve feeds both trains of auxiliary feedwater to steam generator RC-2A, the licensee declared both trains inoperable and entered technical specification (TS) 2.5(1)D. This technical specification requires that actions be taken to immediately restore at least one train of auxiliary feedwater to operable, and suspends all mode changes required by other limiting conditions of operations. The licensee also declared auxiliary feedwater containment isolation valve HCV-1108A, the injection valve to steam generator RC-2B, inoperable based on irregular operation while opening.

During the refueling outage, the licensee rebuilt both of the affected valves using new types of elastomers in the valve seals. The licensee changed the low friction seal material from polytetrafluoroethylene (PTFE, commonly referred to as Teflon®) to polychlorotrifluoroethylene (CTFE) because of the potential for PTFE to degrade under high radiation flux. Additionally, O-ring seals within the valve were changed from nitrile to polyvinylidene fluoride (FKM, commonly referred to as Viton®) due to temperature limitations of the nitrile O-rings. The new seals were not rated for the temperature to which they were subjected to during plant heatup and subsequently deformed, causing the valves to bind.

The NRC used Management Directive 8.3, "NRC Incident Investigation Program," to evaluate the level of NRC response for this event. In evaluating the deterministic criteria in the Management Directive, the NRC determined that: (1) the event led to the loss of a safety function, in that the event involved the common-cause loss of both trains of AFW to both steam generators, and (2) the event involved possible adverse generic implications, in that the event involved the common-cause failures of elastomer seals at high temperatures. The senior reactor analyst determined the Conditional Core Damage Probability for the event to be 3.0×10^{-6}.

Based on these deterministic criteria, the risk associated with the loss of both trains of auxiliary feedwater flow to both steam generators, and concerns associated with the engineering change processes used by the licensee to replace the original seals, Region IV chartered this Special Inspection to review the facts and circumstances associated with these failures.

The inspectors used NRC Inspection Procedure 93812, "Special Inspection Procedure," to conduct the inspection. The inspections included field walkdowns of equipment, interviews with station personnel, and reviews of procedures, corrective action documents, and design documentation.

OTHER ACTIVITIES

Cornerstone: Mitigating Systems

4OA5 Other Activities: Special Inspection (93812)

The following activities were performed in accordance with the Special Inspection Charter (Attachment 2).

.1 Develop a complete sequence of events or occurrences that are related to the failure of affected valves beginning with the event or occurrence that prompted the licensee to replace the seals in the affected valves, including the licensee's decision to replace the seals via their like-for-like process, and ending with the licensee restoring the valves to operable status after June 5, 2015. (Charter Item 2)

a. Inspection Scope

The inspectors developed and evaluated a timeline of the events leading up to, during, and after the failure of the two auxiliary feed containment isolation valves. This includes troubleshooting activities and restoration of the valves. The inspectors developed the timeline, in part, through a review of work orders, condition reports, station logs, and interviews with station personnel.

b. Sequence of Events

The inspectors created the following timeline during their review of the events related to the failure of the auxiliary feed containment isolation valves on June 5, 2015.

Date/Time	Activity
June 3, 2002	Condition Report CR-2002-02124 identified an elevated piping temperature of 347°F next to valve HCV-1108B. The Condition Report stated that HCV-1108A is much closer to the steam generator injection nozzles and temperature could be close to steam generator temperatures. The Condition Report identified that the design specification sheet for valves HCV-1107A & B and HCV-1108A & B show maximum temperature specifications as 120°F and 180°F for the B and A valves respectively. The Condition Report also stated that the nitrile elastomers in the valves are not rated for such high temperatures. Recommended corrective actions in the Condition Report to replace or modify valves HCV-1107A and HCV-1108A to be compatible with a steam environment up to 547°F and 1000 psig were not completed.

Date/Time	Activity
January 11, 2013	Engineering Analysis EA 12-024 evaluated design temperature requirements for the elastomers in valves HCV-1107A and HCV-1108A. The analysis incorrectly concluded that the elastomers were required to meet a design rating of 140°F.
October 25, 2013	Condition Report CR-2013-19342 identified that valves HCV-1107A and HCV-1108A contain low friction seals made of PTFE which is susceptible to degradation from high radiation exposure. The Condition Report recommended the replacement of the seals with a more radiation resistant material.
April 24, 2014	Memorandum NED-14-063 DEN voided Engineering Analysis EA 12-024, stating that incorrect design temperatures were determined for valves HCV-1107A and HCV-1108A and that elastomers will need to be replaced.
October 7, 2014	The licensee made initial contact with Fisher (valve manufacturer) requesting a proposal for a more radiation resistant alternate material for the low friction seals in valves HCV-1107A and HCV-1108A.
April 7, 2015	Replacement CTFE seals are received on site and accepted into warehouse inventory.
April 17, 2015	Item Equivalency Evaluation (IEE) 63861 is prepared and completed for CTFE low friction seals.
May 4, 2015	Modification FC-1173D is completed and approved for FKM O-ring replacement of nitrile O-rings in valves HCV-1107A and HCV-1108A.
May 8, 2015	New CTFE low friction seals and FKM O-rings are installed in valves HCV-1107A and HCV-1108A via Work Orders 550466 and 504047.
May 20, 2015	The cold stroke post-maintenance tests of valves HCV-1107A and 1108A are completed satisfactorily via surveillance test procedure OP-ST-AFW-3010.
June 1, 2015 03:36	Reactor Coolant System temperature reached 300°F during the heatup coming out of the outage. Auxiliary feedwater was required to be operable. Surveillance test for turbine driven pump was required within seven days.
June 2, 2015 18:12	The plant entered Mode 3 - Reactor Coolant System temperature greater than 515°F.
June 5, 2015 13:30	Valve HCV-1107A failed to stroke during performance of surveillance test OP-ST-AFW-3010. The valve was immediately declared inoperable.

Date/Time	Activity
June 5, 2015 13:30	Valve HCV-1108A stroked open as part of surveillance test OP-ST-AFW-3010. The valve exhibited jerky movements and did not smoothly open.
June 5, 2015 13:30	Operators entered Technical Specification LCO 2.5(1)D for both trains of auxiliary feedwater inoperable.
June 5, 2015 16:51 to 17:13	Thermal imaging pictures taken of valves HCV-1107A and HCV-1108A showed maximum temperatures on the valves of 427°F and 269°F respectively.
June 5, 2015 18:00	Valve HCV-1108A was declared inoperable after an extent of condition evaluation of the valve HCV-1107A failure because it did not operate smoothly.
June 5, 2015 20:35	Commenced cooldown of the reactor coolant system to effect repairs of valves HCV-1107A and HCV-1108A.
June 5, 2015 21:10	Exited Mode 3. Reactor coolant system temperature less than 515°F.
June 6, 2015 03:45	Operators exited Technical Specification LCO 2.5(1)D because reactor coolant system temperature was had decreased below 300°F and auxiliary feedwater was no longer required.
June 6, 2015 04:56	Entered Mode 4. RCS Temperature less than 210°F.
June 6, 2015	PTFE low friction seals and nitrile O-rings were reinstalled in valves HCV-1107A and HCV-1108A valves via Work Orders 554904 and 554906.
June 6, 2015 23:00	Operators commenced plant heatup.
June 7, 2015 01:30	The cold stroke post-maintenance tests of valves HCV-1107A and HCV-1108A were completed satisfactorily per surveillance procedure OP-ST-AFW-3010.
June 7, 2015 02:25	Operators declared valves HCV-1107A and HCV-1108A operable but degraded. Operations requests an operability evaluation by engineering in Condition Report CR-2015-07613.
June 7, 2015 03:59	Exited Mode 4, transitioning to Mode 3. RCS Temperature greater than 210°F
June 7, 2015 10:15	Reactor coolant system temperature greater than 300°F. Auxiliary Feedwater was required to be operable.
June 8, 2015 09:20	Entered Mode 3. Reactor coolant system temperature greater than 515°F.

Date/Time	Activity
June 8, 2015 16:07	A hot stroke post-maintenance test of valves HCV-1107A and HCV-1108A per procedure PE-RR-VX-0414S was completed satisfactorily.
June 9, 2015 03:09	Entered Mode 2. Reactor critical.
June 9, 2015 12:09	Entered Mode 1 with reactor power greater than 2%.
June 11, 2015 04:45	The hot strokes of valves HCV-1107A and HCV-1108A per surveillance test procedure OP-ST-AFW-3010 were completed satisfactorily.

.2 Assess the adequacy of the equivalency evaluation and parts-change processes used by the licensee during the material selection for the replacement elastomers to be installed in the AFW injection valves. (Charter Item 3)

a. Inspection Scope

The inspectors reviewed the item equivalency evaluations that were used for material selection of the replacement elastomers installed in the auxiliary feed containment isolation valves including the information that was used to make the determinations. The inspectors evaluated the licensee's implementation of their procedures and processes for item equivalency evaluations, material selection, and configuration changes.

b. Observations

The inspectors determined that the human performance risk analysis tools available (i.e. HU-AA-1212, "Technical Task Risk/Rigor Assessment, Pre-Job Brief, Independent Third Party Review, and Post-Job Review") were not being utilized in the most conservative manner. The human performance risk analysis procedure begins with determining the consequences of performing the equivalency evaluation incorrectly. For both evaluations, the consequences were rated as low and no further human performance risk evaluation was conducted. The inspectors concluded that the human performance failure consequences should have been rated higher based on the safety significance of the affected equipment. This may have required further evaluation of human performance risks associated with the item equivalency evaluation and non-significant change evaluation, but did not constitute a finding or violation of NRC requirements. This was documented in Condition Report CR-2105-11244.

The item equivalency evaluation process includes a brief discussion about design margin. This discussion states "A replacement item can function differently than the original item and still be evaluated as an equivalent item, since the [structure, system, or component] design function can be different than actual function. The difference pertains to margins associated with that function." It also states that the basis for this statement is that the documented design will include any margin for the item that may be necessary. In the case of the low friction seal change, the original PTFE seal had a temperature rating of 500°F, whereas the replacement seal had a temperature rating of 390°F. The information available to the evaluator showed a design temperature of the auxiliary feedwater injection valve of 180°F and a design basis accident temperature of 374°F. As the perceived design requirement was 374°F, the evaluator accepted the reduction in margin. In this case, the margin reduction resulted in failure of the valve. The inspectors determined that, although other factors were associated with the incorrect evaluation (e.g. the valve not having a design specification appropriate for the conditions), this evaluation did not effectively maintain design margin through a systematic and rigorous process. Although the margin reduction aspects of the inadequate item equivalency evaluation did not constitute a finding, one finding was identified based on the associated design control aspects and is documented below.

c. Findings

Introduction. The inspectors reviewed a Green, self-revealing, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to evaluate the suitability of materials utilized during the design review process. Specifically, the licensee failed to identify during the design review process that replacement valve internal seal materials for the steam generator auxiliary feed containment isolation valves would not be suitable for high temperature conditions that the valves would experience in service, and as a result, caused both trains of the safety-related auxiliary feedwater system to become inoperable during hot standby conditions.

Description. On June 5, 2015, the licensee was performing surveillance testing to verify operability of the auxiliary feedwater system prior to critical operations. During the performance of surveillance test OP-ST-AFW-3010, "Auxiliary Feedwater System Quarterly Category A and B Valve Exercise Test", the steam generator auxiliary feed containment isolation valve HCV-1107A that supplies steam generator RC-2A failed to open on demand. Additionally, the steam generator auxiliary feed containment isolation valve HCV-1108A that supplies steam generator RC-2B, failed to open in a smooth manner (i.e. jerky movements were observed). The licensee initially declared valve HCV-1107A inoperable and entered Technical Specification 2.5 "Steam and Feedwater Systems" Limiting Condition of Operation 2.5.1(D) since this valve supplies both safety-related trains of auxiliary feedwater required to meet the decay heat removal safety function for steam generator RC-2A. As part of an extent of condition evaluation, the licensee also declared valve HCV-1108A inoperable because it did not operate smoothly when stroked. Thermography temperature readings of the valves showed HCV-1107A to be 430°F and HCV-1108A to be 270°F shortly after the event. These temperatures were beyond the design temperature rating of the valves of 180°F. The licensee cooled down the reactor plant to less than 160°F to make repairs to the steam generator auxiliary feedwater containment isolation valves utilizing a non-safety flow path via the diesel-driven auxiliary feedwater pump. Following reactor plant cooldown, the licensee inspected both valves and determined that the low friction seals had failed under high temperature conditions which prevented proper valve operation.

During the station's most recent refueling outage (27RFO), the licensee rebuilt valves HCV-1107A and HCV-1108A and replaced the existing polytetrafluoroethylene (PTFE) low friction seals and nitrile O-rings with polychlorotrifluoroethylene (CTFE) low friction seals and polyvinylidene fluoride (FKM) O-ring materials. The existing PTFE low friction seals had been identified in Condition Report 2013-19342 as subject to degradation under a high radiation flux, and the licensee generated Item Equivalency Evaluation IEE-35549 to replace the existing low friction seals with a more radiation resistant material. The existing nitrile O-rings had been identified in Condition Report CR 2015-00159 as having an operating temperature limit of 180°F which was not high enough for the temperature conditions that could be present on the valve. The licensee generated Engineering Change EC 66130 to replace the existing O-rings with a more temperature resistant material. Following the failure of surveillance test OP-ST-AFW-3011 on June 5, 2015, as part of a special inspection, the inspectors reviewed the modifications associated with HCV-1107A and HCV-1108A and determined that the licensee failed to evaluate the suitability of the materials used based on the following:

- The inspectors identified that the modification IEE-35549 associated with the valve seals utilized a voided design basis document as a temperature input to evaluate the acceptability of using a material that was rated for a lower temperature than previously installed. The licensee voided Engineering Analysis EA 12-024, “Determination of Design Temperature Requirement for Elastomers in Valves HCV-1107A and HCV-1108A,” on April 24, 2014, prior to the approval of modification IEE-35549, and a memorandum attached to the electronic document indicated that the reference contained incorrect information including the maximum design temperature of the elastomers in the valves of 140°F.

- The inspectors identified that both modification IEE-35549 and engineering change EC 66130 did not account for the actual maximum fluid temperatures that the valve internals could be exposed to due to leakage from the steam generator auxiliary feedwater injection check valves. The steam generator auxiliary feedwater injection check valves are tested for seat leakage per surveillance test procedure OP-ST-AFW-3013, “Auxiliary Feedwater Injection Check Valves FW-163 and FW-164 Close Test,” with a maximum allowable leakage of 1 gallon per minute (gpm). This procedure identified leak rates during testing of approximately 0.06 gpm and 0.25 gpm respectively for check valves FW-163 and FW-164. Leakage past these check valves was likely the source of the elevated temperatures of valves HCV-1107A and HCV-1108A which caused the low friction seals to fail. These elevated temperatures were above the design specification of the valves of 180°F.

The licensee entered this issue into their corrective action program as Condition Report CR-2015-07564. The licensee replaced the valve internals with new elastomers of the same type of materials previously installed in these valves. The licensee determined that the use of these materials is acceptable until the next refueling outage. Additionally, the licensee evaluated the acceptability of the use of CTFE in other valves identified in a preliminary extent of condition and found those uses to be acceptable. The licensee is in the process of performing a root cause analysis of the event to determine additional corrective actions, extent of condition, and extent of cause. Following the event, the inspectors determined that the licensee had previously identified the presence of elevated fluid temperatures at valves HCV-1107A and HCV-1108A due to check valve leakage, and that these temperatures were greater than temperatures described in design basis documents for the valves. Specifically, Condition Report CR-2002-02124 identified that the auxiliary feed containment isolation valves (HCV-1108A&B) experienced temperatures above their design specification after unexpected high temperatures were found on the piping near HCV-1108B (auxiliary feed containment isolation valve outside containment) due to back leakage through the check valves from the steam generators. Additionally, Condition Reports CR-2003-00400 and CR-2013-21956 identified a steam plume coming from auxiliary feed containment isolation valve HCV-1108A, which is an indication of high temperatures on the valve that exceeded the design specification. The licensee failed to take into account this operating experience in their evaluation of the use of CTFE in the valves.

Analysis. The inspectors determined that the licensee's failure to evaluate the suitability of the materials used during the design review process for the steam generator auxiliary feed containment isolation valves was a performance deficiency. The performance deficiency is more than minor, and therefore a finding, because it is associated with the equipment performance attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Specifically, the licensee's failure to properly evaluate the suitability of CTFE for use in the steam generator auxiliary feed containment isolation valves led to the failure of HCV-1107A and HCV-1108A and rendered both safety-related trains of auxiliary feedwater inoperable.

Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process for Findings At-Power", dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions", the inspectors determined that the finding required a detailed risk evaluation since the finding represented a loss of system and/or function. A Region IV senior reactor analyst performed the detailed risk evaluation in accordance with Appendix A, Section 6.0, "Detailed Risk Evaluation." The detailed risk evaluation result is a finding of very low safety significance (Green). The calculated change in core damage frequency of 2.3×10^{-7} was dominated by a loss of offsite power; common cause failure of the auxiliary feedwater discharge air-operated valves; failure of diesel-driven auxiliary feedwater pump FW-54; failure of the feed and bleed operation; and failure of operators to manually override a steam generator isolation signal and establish a flow path for the main feedwater system. The analyst determined that the finding did not involve a significant impact to external initiators because of the short exposure time, or a significant increase in the risk of a large, early release of radiation. The finding has an operating experience cross-cutting aspect in the problem identification and resolution cross-cutting area since the organization did not systematically and effectively collect, evaluate, and implement relevant internal and external operating experience in a timely manner. Specifically, readily available internal operating experience on the high temperature conditions that valves HCV-1107A and HCV-1108A experienced during normal operations was not utilized during the design change process [P.5].

Enforcement. Title 10 of the *Code of Federal Regulations*, Part 50, Appendix B, Criterion III, "Design Control" requires, in part, that measures shall be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems and components. Contrary to the above, from October 15, 2013 to May 5, 2015, the licensee failed to ensure that measures were adequate during the design review process for the selection and review for suitability of application of materials that are essential to the safety-related functions of the steam generator auxiliary feedwater valves. Specifically, the licensee failed to identify during the design review process that the internal elastomeric materials selected for the steam generator auxiliary feed containment isolation valves would not be suitable for the high temperature conditions the valves experienced, and as a result caused both trains of the safety-related auxiliary feedwater system to become inoperable. The licensee replaced the elastomers in the valves and restored them to service. Because this violation was of very low safety significance (Green) and was entered into the licensee's corrective action program as

Condition Report CR-2015-07564, this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the Enforcement Policy: NCV 05000285/2015011-01, “Failure to Ensure the Suitability of Replacement Materials during the Design Review Process.”

.3 Review procedures the licensee has developed to implement material substitutions (e.g., lubricants, gaskets, fasteners, etc.) without additional reviews. (Charter Item 4)

a. Inspection Scope

The inspectors reviewed the licensee procedures for item equivalency evaluations and non-significant configuration changes that were used for material substitutions. These reviews included procedures for human performance risk analysis that were associated with the evaluations. The inspectors also reviewed changes the licensee implemented to their procedures and processes as a result of the auxiliary feedwater valve failures.

b. Observations

The inspectors determined that the item equivalency process found in procedure SM-FC-300-1001, “Procurement Engineering Process and Responsibilities,” Revision 0, includes a review of external operating experience but does not require review of operating experience internal to Fort Calhoun Station. In the case of the item equivalency evaluation for the low friction seal replacement, no internal operating experience review was conducted. Had this internal operating experience review been conducted, relevant operating experience regarding operating temperature of the auxiliary feed containment isolation valves may have prevented approval of the new low friction seal material that was not rated for the environment of the auxiliary feed containment isolation valves. Specifically, Condition Report CR-2002-02124 identified that the auxiliary feed containment isolation valves (HCV-1108A&B) experienced temperatures above their design specification after unexpected high temperatures were found on the piping near HCV-1108B (auxiliary feed containment isolation valve outside containment) due to back leakage through the check valves from the steam generators. Additionally, Condition Reports CR-2003-00400 and CR-2013-21956 identified a steam plume coming from auxiliary feed containment isolation valve HCV-1108A which is an indication of high temperatures on the valve that exceeded the design specification. This did not constitute a finding or violation, but contributed to the finding discussed in the previous section. This was documented by the licensee in Condition Report CR-2015-11244.

c. Findings

No findings were identified.

.4 Assess the control room operator response and Technical Specification compliance following the failure of the AFW injection valves. (Charter Item 5)

a. Inspection Scope

The inspectors reviewed control room operator logs and event reports for details of the operator response, and compared the response to the actions required by the licensee’s technical specifications. The inspectors also interviewed the control room supervisor who was on shift during the event, and some of the operators and maintenance personnel who were performing the testing of the valves.

b. Observations

The inspectors concluded that the licensed operators responded appropriately to the event and that the licensee complied with the requirements of their technical specifications.

c. Findings

No findings were identified.

.5 Evaluate the progress of the licensee's cause evaluation for the valve failures that occurred on June 5, 2015, and any corrective actions completed or planned. Also include the licensee's completed and planned extent-of-condition and extent-of-cause reviews, and determine if the licensee is conducting the cause evaluation at a level of detail commensurate with the safety significance of the issue. (Charter Item 6)

a. Inspection Scope

At the time of the inspection, the licensee had not completed their cause evaluation for the event. The inspectors interviewed the licensee's cause evaluation team and reviewed the preliminary corrective actions taken by the licensee. The inspectors evaluated the cause evaluation team's planned extent-of-condition and extent-of-cause reviews. The inspectors evaluated the licensee's corrective actions and cause evaluation efforts against their corrective action program guidance.

The inspectors reviewed the licensee's immediate corrective actions for the failed valves, and reviewed the operability evaluation performed after replacing the valve components.

b. Observations

The inspectors determined that the licensee was evaluating the event appropriately and in accordance with their corrective action program guidance. The inspectors also determined that the cause evaluation team's planned extent-of-condition and extent-of-cause reviews should adequately address the safety issue if implemented as planned.

c. Findings

Introduction. The inspectors identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's failure to follow the operability determination procedure. Specifically, the licensee failed to establish a valid technical basis for operability of auxiliary feed containment isolation valves HCV-1107A and HCV-1108A.

Description. The licensee identified in Condition Report CR-2013-19342 that the auxiliary feedwater valve seals contained PTFE, which is susceptible to degradation under high radiation flux. The licensee performed an operability evaluation using procedure OP-FC-108-115, "Operability Determinations," to provide an input to operability of the valves. This evaluation concluded that the valves were operable but degraded because the seals had recently been replaced, and the expected dose from one operating cycle and a postulated accident would not sufficiently degrade the seals to the point of failure. During the refueling outage in May 2015, the licensee replaced the PTFE seals with the CTFE seals that ultimately failed. Following the valves' failure on June 5, the licensee replaced the failed valve elastomers with new PTFE seals and nitrile O-rings. The licensee then initiated Condition Report CR-2015-07583 and performed another operability evaluation partially based on the one previously performed in 2013. The new operability evaluation also considered the effect of high temperatures from a main steam line break on the valve elastomers. The inspectors reviewed this evaluation and identified technical issues with the operability evaluation, including:

- The licensee stated that the maximum temperature that the nitrile O-rings would be exposed to was 374°F based on a post main steam line break event containment environment. The licensee failed to take into account actual operating temperatures that exceeded that value. In addition, the licensee noted that the nitrile O-ring was only rated to 250°F. Furthermore, the licensee documented that nitrile compounds have a melting temperature of ranging from 411°F to 748°F depending on the type of compound, and that the specific compounds used in this application were unknown.

- The licensee failed to evaluate all of the valve components. The valve components (excluding the elastomers) were rated by the manufacturer for a temperature of 325°F. This was below the maximum temperature that the valve would be exposed to. The operability evaluation did not address the effect of high temperature on the metallic components and the mechanical operation of the valves.

The inspectors determined that the procedure required an operability evaluation to be sufficient to address the capability of the structure, system or component to perform its safety function. The inspectors found that the evaluation was not sufficient because it did not determine that the O-rings would function under all potential temperature conditions and did not consider the function of the other valve components. The licensee documented these issues in Condition Report CR-2015-08362 and took actions to revise their operability evaluation. The licensee determined that the maximum valve temperature was 550°F based on steam generator conditions. The licensee determined that a potential O-ring failure would not prevent or interfere with valve operation based on information obtained from the vendor that the specific nitrile compound used would not melt below the maximum valve temperature, and they completed an evaluation concluding that the other valve components also would not fail at the maximum valve temperature.

<u>Analysis</u>. The licensee's failure to follow the operability determination procedure was a performance deficiency. The performance deficiency is more than minor, and therefore a finding, because it is associated with the equipment performance attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Specifically, the licensee failed to sufficiently address the capability of the steam generator auxiliary feed containment isolation valves HCV-1107A and HCV-1108A to perform their safety function, requiring significant further analysis to demonstrate operability. Using

Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," and Appendix A, Exhibit 2, "Mitigating Systems Screening Questions," the finding was determined to be of very low safety significance (Green) because although the finding was a deficiency affecting design or qualification, but the mitigating structure, system or component maintained its operability. The finding has a consistent process cross-cutting aspect in the human performance cross-cutting area since the organization did not use a consistent, systematic approach to make decisions and incorporate risk insights appropriately. Specifically, the licensee failed to re-evaluate the operability decision when new information on the conditions and susceptibility affecting valves HCV-1107A and HCV-1108A during normal operations was available [H.13].

Enforcement. Title 10 of the *Code of Federal Regulations*, Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings" requires, in part, that activities affecting quality shall be accomplished in accordance with prescribed instructions and procedures. Contrary to the above, on June 8, 2015, the licensee failed to accomplish an activity affecting quality in accordance with a prescribed procedure. Specifically, the licensee performed an operability evaluation for the steam generator auxiliary feed containment isolation valves and failed to follow step 4.4.2 of OP-FC-108-115, "Operability Determinations," a quality related procedure. The licensee revised the operability evaluation to address the additional material deficiencies. Because this violation was of very low safety significance (Green) and was entered into the licensee's corrective action program as Condition Report CR-2015-08362, this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the Enforcement Policy: NCV 05000285/2015011-02, "Failure to Establish a Technical Basis for Operability of the Auxiliary Feedwater System"

.6 Review the history of any performance issues associated with the stem seals installed in the affected valves. (Charter Item 7)

a. Inspection Scope

The inspectors reviewed condition reports, work orders, and test records for the affected valves. The inspectors looked for instances when the licensee could have or should have identified the non-conforming design of the valves earlier and resolved the condition prior to their failure. The inspectors also reviewed previous NRC inspection reports and discussed the valve history with licensee personnel and the NRC resident inspectors.

b. Observations

The inspectors determined that there was a history of elevated temperatures on these valves, and that the licensee had identified the potential for the valves to be exposed to temperatures higher than their design temperature several times dating back to 2002. The licensee also had an opportunity to identify the non-conforming design when NRC inspection report 05000285/2013008 identified that the elastomers in the valve actuators were not designed for the appropriate temperatures. Finally, the inspectors noted that earlier in 2015, NRC inspectors had questioned the design temperature of the valve, and the licensee had not recognized the non-conforming condition.

c. Findings

Introduction. The inspectors identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for the licensee's failure to correct a condition adverse to quality. Specifically, the licensee failed to take corrective actions after identifying that the steam generator auxiliary feed containment isolation valves were not rated for the maximum temperature they would experience in service.

Description. The inspectors reviewed the performance history of the steam generator auxiliary feed containment isolation valves. On February 2, 2015, an NRC inspector questioned the licensee whether valves HCV-1107A and HCV-1108A were adequately designed for containment temperatures. The licensee determined that the design specification for the maximum temperature of the valves was 180°F, and the containment temperature following a main steam line break was evaluated to be 374°F. In addition, the piping section adjacent to the valves was rated for 547°F due to its exposure to the high temperature steam generator conditions. The fact that the valve was not designed for the most limiting conditions was a non-conforming condition of a safety related component, and was a condition adverse to quality. However, the licensee did not initiate a condition report or take actions to resolve and correct the condition. Additionally, the inspectors determined that in 2002, the licensee initiated Condition Report CR-2002-02124 after identifying elevated temperatures in the auxiliary feedwater piping. This condition report documented that the design specification for the two valves was 180°F and that this temperature had been exceeded. Although the condition report description recommended modifying the design of the valves, no actions were taken. In both of these instances, the licensee recognized that the valve design temperature was not adequate for its application, but did not take action to resolve the discrepancy.

In addition to instances where the licensee specifically identified the non-conforming design, there were five condition reports written between 2009 and 2014 documenting environmental qualification issues with elastomers and other components in the valve actuators, including an NRC finding documented in NRC Inspection Report 05000285/2013008, that should have led to the realization that the valves themselves were not suitable for their environment. The licensee also identified steam leaking from the valves on at least two occasions, documented in Condition Reports CR-2003-00400 and CR-2013-21956, which should have led the licensee to recognize that the valve temperature was in excess of the design temperature of 180°F. The inspectors determined that although the inadequate design was a non-conforming condition, the valves were not inoperable until the licensee changed the elastomer material during the 2015 refueling outage as a result of the inadequate design control. The licensee documented the failures to identify and correct the non-conforming design in Condition Report CR-2015-08523.

Analysis. The licensee's failure to take corrective action for a non-conforming condition was a performance deficiency. The performance deficiency is more than minor, and therefore a finding, because it is associated with the design control attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Specifically, the licensee failed to take corrective actions to ensure an adequate design for the steam generator auxiliary feed containment isolation valves HCV-1107A and HCV-1108A.

Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," and Appendix A, Exhibit 2, "Mitigating Systems Screening Questions," the finding was determined to be of very low safety significance (Green) because although the finding was a deficiency affecting the design or qualification of a mitigating system, structure, or component, the system, structure, or component maintained its operability. The finding has a basis for decisions cross-cutting aspect in the human performance cross-cutting area since leaders and individuals did not verify their understanding or question the basis of decisions. Specifically, the licensee failed to understand the potential significance of the non-conforming design of the valves and the basis for not taking corrective actions [H.10].

Enforcement. Title 10 of the *Code of Federal Regulations*, Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to the above, prior to June 5, 2015, the measures established by the licensee failed to identify and correct a condition adverse to quality. Specifically, the licensee became aware of a non-conforming condition involving the safety-related auxiliary feedwater containment isolation valves and failed to take action to correct the condition. The licensee documented the non-conforming condition and evaluated the operability of the auxiliary feedwater system. Because this violation was of very low safety significance (Green) and was entered into the licensee's corrective action program as Condition Report CR-2015-08523, this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the Enforcement Policy: NCV 05000285/2015011-03, "Failure to Correct a Non-Conforming Condition Associated with Auxiliary Feedwater Valves"

.7 Review industry operating experience that is associated with similar failures of air-operated valves in general, and the AFW injection valves in particular, and assess the licensee's responses to applicable operating experience reports. (Charter Item 8)

a. Inspection Scope

The inspectors evaluated the licensee's application of industry operating experience related to this event. The inspectors reviewed applicable operating experience reports and generic NRC communications with a specific emphasis on air operated valves to assess whether the licensee had appropriately evaluated the reports for relevance to the facility and incorporated applicable lessons learned into station programs and procedures.

b. Observations

The inspectors reviewed four examples of operating experience relevant to the failure of auxiliary feedwater valves, other air-operated valves, and valve components. The inspectors determined that three of the four examples had been evaluated by the licensee as not being applicable to the Fort Calhoun Station:

- Condition Report OE-2012-01977 documented a failure of a solenoid-operated valve at another site due to the valve not being designed for the given environmental temperature. The licensee screened this condition report as not relevant to any nuclear program because the valve that failed was in a non-safety related application.

- Condition Report OE-2012-00639 documented a failure to perform adequate testing on an air-operated valve for a safety chiller. The licensee screened this condition report as not applicable to the Fort Calhoun Station because the Fort Calhoun Station does not have containment safety chillers.

- Condition Report OE-2011-1209 documented the failure of an air-operated auxiliary feedwater pump discharge valve to open during testing. The licensee screened this condition report as not applicable to the Fort Calhoun Station because the Fort Calhoun Station auxiliary feedwater pump discharge valves are manually operated.

The inspectors disagreed with the licensee's evaluation of the above operating experience. The inspectors observed that, although the specific details of the equipment in these operating events may have differed from the equipment at the Fort Calhoun Station, the operating events demonstrated lessons that apply generically to the design, operation, testing, and maintenance of air-operated valves in use at the station.

The inspectors determined that the licensee did not appropriately evaluate operating experience reports and failed to implement applicable lessons from operating experience relevant to the failure of air-operated valves and, in particular, the auxiliary feedwater containment isolation valves. The inspectors concluded that the failure to implement applicable operating experience did not involve a violation of NRC requirements and did not constitute a finding because none of the operating experience reports reviewed would have prevented the specific failure mechanism of the auxiliary feedwater containment isolation valves. The licensee initiated Condition Report CR-2015-1124 to evaluate the operating experience screening process for the above events.

The inspectors concluded that the failure to implement applicable operating experience did not constitute a finding because none of the operating experience reports reviewed would have prevented the specific failure mechanism of the auxiliary feedwater containment isolation valves and did not involve a violation of NRC requirements. The licensee initiated Condition Report CR-2015-1124 to evaluate the operating experience screening process for the above events.

c. Findings

No findings were identified.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On July 2, 2015, the inspectors debriefed Mr. Cortopassi, Site Vice President, and other members of the licensee's staff. The licensee representatives acknowledged the findings presented.

On August 14, 2015, the inspectors conducted an exit meeting by telephone with Mr. Dean, Plant Manager, and other members of the licensee's staff. The licensee representatives acknowledged the findings presented. The inspectors informed the licensee that no proprietary information would be documented in the report.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

S. Anderson, Manager, Nuclear Projects
D. Bakalar, Manager, Site Security
D. Brehm, Supervisor, Radiation Protection
C. Cameron, Principle, Regulatory Specialist
L. Cortopassi, Site Vice President
E. Dean, Plant Manager
S. Fatora, Director, Site Work Management
H. Goodman, Director, Site Engineering
R. Hugenroth, Supervisor Nuclear Oversight
T. Hutchinson, Reliability Engineer
K. Ihnen, Manager, Nuclear Oversight
K. Kingston, Superintendent Maintenance
R. Lowery, Senior Operations Training Instructor
K. Maassen, Program Engineer
T. Maine, Manager, Radiation Protection
K. Mann, Regulatory Specialist
E. Matzke, Senior Regulatory Engineer
W. McCall, Health Physicist
J. McManis, Manager, Engineering Programs
B. Pearson, Supervisor, Radiological Protection
C. Scofield, Senior Nuclear Design Engineer-Mechanical
T. Simpkin, Manager, Site Regulatory Assurance
S. Swanson, Director, Site Operations

NRC Personnel

M. Schneider, Senior Resident Inspector
D. Loveless, Senior Reactor Analyst

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000285/2015011-01	NCV	Failure to Ensure the Suitability of Replacement Materials during the Design Review Process
05000285/2015011-02	NCV	Failure to Establish a Technical Basis for Operability of the Auxiliary Feedwater System
05000285/2015011-03	NCV	Failure to Correct a Non-Conforming Condition Associated with Auxiliary Feedwater Valves

LIST OF DOCUMENTS REVIEWED

CALCULATIONS

NUMBER	TITLE	REVISION
EA11-026	Design Requirements for Backup Systems for Air Operated Valves	1
EA13-040	Evaluation of Valves with Teflon Subcomponents Located in Radiation Areas	0
EA-FC-06-032	Environmental Parameters for Electrical Equipment Qualification	2
EA-FC-10-020	Electrical Equipment Qualification Radiation Dose Reconstitution Analysis	1
EEQ-H-01	ASCO NP Series Solenoid Valve	22
EEQ-H-02	NAMCO EA180 Limit Switches	24
FC07536	FW-6 and FW-10 Suction and Discharge Piping Friction Loss	0A

CONDITION REPORTS

CR-1998-01148	CR-2002-02124	CR-2004-04204	CR-2007-00550	CR-2009-05356
CR-2009-05780	CR-2012-05509	CR-2012-05779	CR-2012-08621	CR-2012-12118
CR-2012-15703	CR-2013-01396	CR-2013-02611	CR-2013-02611	CR-2013-04193
CR-2013-05570	CR-2013-19342	CR-2013-20953	CR-2013-21956	CR-2014-04954
CR-2014-10156	CR-2014-15695	CR-2015-00159	CR-2015-05807	CR-2015-07235
CR-2015-07562	CR-2015-07564	CR-2015-08362	CR-2015-08362	CR-2015-08363
CR-2015-08522	CR-2015-08523	CR-2015-08527	CR-2015-08571	CR-2015-11244
OE-2010-01573	OE-2011-01209	OE-2012-00639	OE-2012-01977	OE-2013-01681

DRAWINGS

NUMBER	TITLE	REVISION
11405-M-253	Composite Flow Diagram Steam Generator Feedwater and Blowdown	53
Fig 8.1-1	Simplified One Line Diagram Plant Electrical System P&ID	147

PROCEDURES

NUMBER	TITLE	REVISION
AOP-28	Auxiliary Feedwater System Malfunctions	18

PROCEDURES

NUMBER	TITLE	REVISION
CC-AA-103	Configuration Change Control for Permanent Physical Plant Changes	25
CC-AA-104	Document Change Requests	15
CC-FC-102	Design Input and Configuration Change Impact Screening	1
CC-MW-101	Engineering Change Requests	1
HU-AA-1212	Technical Task Risk/Rigor Assessment, Pre-Job Brief, Independent Third Party Review, and Post-Job Review	6
IC-RR-VX-0409	Diagnostic Testing of Air Operated Valves	19
OI-AFW-4	Auxiliary Feedwater Startup and System Operation	90
OP-FC_108-115	Operability Determinations	2
OP-ST-AFW-3007	Auxiliary Feedwater Functional Test of Circuits and Components	16
OP-ST-AFW-3010	Auxiliary Feedwater System Quarterly Category A and B Valve Exercise Test	12
OP-ST-AFW-3013	Auxiliary Feedwater injection Check valves FW-163 and FW-164 Close Test	0
PED-GEI-41	Processing Configuration Changes	20
PED-GEI-60	Preparation Substitute Replacement Items	49
PED-GEI-81A	AOV/MOV Rising Stem Valve Packing Installation	5
PED-QP-2	Configuration Change Control	64
PE-RR-VX-0414S	Inspection and Repair of Safety Related Fisher “HSC” Control Valves	13
PI-AA-115	Operating Experience Program	0
PI-AA-120	Issue Identification and Screening Process	1
PI-AA-125	Corrective Cation Program (CAP) Procedure	2
PI-AA-125-1001	Root Cause Analysis Manual	1
PI-AA-125-1006	Investigation Techniques Manual	1
SM-FC-300	Procurement Engineering Support Activities	0
SM-FC-300-1001	Procurement Engineering Process and Responsibilities	0
SO-M-101	Maintenance Work Control	106

WORK ORDERS

131115	131487	271461	444829	490116
504047	507619	519423	550466	550466
554904	554906	554970		

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION
	Control room logs from May 31, 2015 to June 7, 2015	
	Evaluation and Design & Licensing Basis of Valves HCV-1107A/B & 1108A/B	5/14/15
	Updated Safety Analysis Report	5/5/15
	Valve Packing Data Sheet for HCV-1107A	1
163613-4	Safety Basis Evaluation: Seal, Low-Friction, PCTFE, for 3 inch Type HSC Valves	
199135	Purchase Order for 3-inch Valve Plug	
EC 30568	Fort Calhoun Nuclear Station Unit #1 Safety-Related Check Valve Specification	1
EC 66130	Non-significant Configuration Change for HCV-1107A/B & HCV-1108A/B	5/4/15
EEQ-H-01	ASCO NP Series Solenoid Valve	22
EEQ-H-02	NAMCO EA180 Limit Switches	24
FC-1029-84	SOER 84-3; Auxiliary Feedwater Pumps Disabled by Backleakage	6/19/84
LIC-87-594 Attachment 1	Design Parameters for HCV-1107A and 1108A	8
SDBD-FW-AFW-117	Auxiliary Feedwater	45
SM-AA-300-1001-F-01	Item Equivalency Evaluation for the replacement of Low Friction Seal, Catalog ID 163613-4	3
SPEC 17	Aux. Feedwater Control Valves	12
STM-AFW	Auxiliary Feedwater System	51
TD A391.0270	Instruction Manual for Flowserve Corp. 3”-Class 900 Durabla WLC Check Valve	
TD F130.0280	Instruction Manual for Fisher Controls Design HSC Valve Body	0

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

June 22, 2015

MEMORANDUM TO: Rayomand Kumana

FROM: Troy Pruett, Director /RA/
Division of Reactor Projects

SUBJECT: SPECIAL INSPECTION CHARTER TO EVALUATE CAUSES OF THE FAILURES OF TWO AUXILIARY FEEDWATER SYSTEM VALVES AT THE FORT CALHOUN STATION

In response to failures of two auxiliary feedwater valves during surveillance testing on June 5, 2015, a special inspection will be performed. You are hereby designated as the special inspection team leader. Michael Langelier is also assigned to your team.

A. Basis

On June 5, 2015, the unit was in Mode 3 while the licensee performed auxiliary feedwater (AFW) surveillance testing prior to startup from a refueling outage. During the testing, steam generator AFW injection valve HCV-1107A failed to open on demand. This valve provides the AFW injection path to the RC-2A steam generator for both the turbine and motor driven AFW pumps. Because this valve feeds both trains of AFW to steam generator RC-2A, the licensee declared both trains of AFW inoperable and entered TS 2.5(1) D, which requires that actions be taken to immediately restore at least one train of AFW to operable, and all mode changes required by other LCOs are suspended. The licensee performed an extent-of-condition review and declared valve HCV-1108A (the similar valve in the flow path to steam generator RC-2B) inoperable based on sluggish operation while opening.

During the refueling outage, the licensee rebuilt both of the affected valves using a new type of elastomer in the valve seals. The licensee changed the seal material because the old seals were made of Teflon, and Teflon has been known to degrade under high radiation flux. The new seals were not rated for the temperature to which they were subjected during plant heatup and deformed, causing the valves to bind.

The NRC used Management Directive 8.3, “NRC Incident Investigation Program,” to evaluate the level of NRC response for this event. In evaluating the deterministic criteria of MD 8.3, the NRC determined that: (1) the event led to the loss of a safety function, in that the event involved the common-cause loss of both trains of AFW to the both steam generators, and (2) the event involved possible adverse generic implications, in that the event involved the common-cause failures of Viton elastomer seals at high temperatures. The senior risk analyst determined the Core Damage Probability for the event to 3.0E-6.

Based on these deterministic criteria, the risk associated with the loss of both trains of AFW flow to both steam generators, and concerns associated with the engineering change processes used by the licensee to replace the original seals, Region IV decided that a Special Inspection is the appropriate level of NRC response.

This Special Inspection is chartered to review concerns associated with these failures, as described below.

B. Scope

The inspectors are expected to perform data gathering and fact-finding to address the following:

1. Before the end of the first day on site, provide a recommendation to Region IV management as to whether the inspection should be upgraded to an augmented inspection team response.

2. Develop a complete sequence of events or occurrences that are related to the failure of affected valves beginning with the event or occurrence that prompted the licensee to replace the seals in the affected valves, including the licensee's decision to replace the seals via their like-for-like process, and ending with the licensee restoring the valves to operable status after June 5, 2015.

3. Assess the adequacy of the equivalency evaluation and parts-change processes used by the licensee during the material selection for the replacement elastomers to be installed in the AFW injection valves.

4. Review procedures the licensee has developed to implement material substitutions (e.g., lubricants, gaskets, fasteners, etc.) without additional reviews.

5. Assess the control room operator response and Technical Specification compliance following the failure of the AFW injection valves.

6. Evaluate the progress of the licensee's cause evaluation for the valve failures that occurred on June 5, 2015, and any corrective actions completed or planned. Also include the licensee's completed and planned extent-of-condition and extent-of-cause reviews, and determine if the licensee is conducting the cause evaluation at a level of detail commensurate with the safety significance of the issue.

7. Review the history of any performance issues associated with the stem seals installed in the affected valves.

8. Review industry operating experience that is associated with similar failures of air-operated valves in general, and the AFW injection valves in particular, and assess the licensee's responses to applicable operating experience reports.

9. For any finding identified, gather the information necessary to enable completion of the significance determination process.

C. <u>Guidance</u>

Inspection Procedure 93812, “Special Inspection,” provides additional guidance to be used by the Special Inspection Team. Your duties will be as described in that procedure. In your review of the circumstances surrounding the event, you should emphasize fact-finding, and should not examine the regulatory process. If you identify safety concerns that are not directly related to the event, report those concerns to the Region IV office for appropriate action.

You will formally begin the special inspection with an entrance meeting to be conducted no later than June 29, 2015. You should provide a daily briefing to Region IV management during the course of your inspections and prior to your exit meeting, and plan to conduct an exit meeting during the afternoon of July 2, 2015. A report documenting the results of the inspection should be issued within 45 days of the completion of the inspection. (Your inspection report number will be 2015011.)

If you develop significant new information that warrants review, this charter may be modified.

CONTACT: Geoff Miller, Chief, DRP
817-200-1173