UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
245 PEACHTREE CENTER AVENUE NE, SUITE 1200
ATLANTA, GEORGIA 30303-1257

July 28, 2014

Cheryl A. Gayheart
Vice President
Joseph M. Farley Nuclear Plant
Southern Nuclear Operating Company, Inc.
7388 North State Highway 95
Columbia, AL 36319

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT – NRC COMPONENT DESIGN BASES INSPECTION REPORT 05000348/2014007 AND 05000364/2014007

Dear Mrs. Gayheart:

On June 6, 2014, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Joseph M. Farley Nuclear Plant Units 1 and 2 and discussed the results of this inspection with you and other members of your staff. The team re-exited via telephone on July 23, 2014, with you and other members of your staff. Inspectors documented the results of this inspection in the enclosed inspection report.

NRC inspectors documented seven findings of very low safety significance (Green) in this report. All of these findings involved violations of NRC requirements. Additionally, NRC inspectors documented one Severity Level IV violation with no associated finding.

If you contest the violations or significance of these 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 II; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC resident inspector at the Joseph M. Farley Nuclear Plant.

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 II; and the NRC resident inspector at the Joseph M. Farley Nuclear Plant.

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

Sincerely,

/RA/

Rebecca L. Nease, Chief
Engineering Branch 1
Division of Reactor Safety

Docket Nos.: 50-348, 50-364
License No.: NPF-2, NPF-8

Enclosure: Inspection Report 05000348/2014007; and 05000364/2014007
w/Attachment: Supplementary Information

cc: Distribution via Listserv

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	05000348, 05000364
License Nos.:	NPF-2, NPF-8
Report Nos.:	05000348/2014007 and 05000364/2014007
Licensee:	Southern Nuclear Operating Company, Inc.
Facility:	Joseph M. Farley Nuclear Plant, Units 1 and 2
Location:	Columbia, AL
Dates:	May 5, 2014, through June 6, 2014
Inspectors:	Tonya Lighty, Reactor Inspector (Lead) Néstor Féliz-Adorno, Reactor Inspector Delza Mas, Reactor Inspector Eric Stamm, Senior Reactor Inspector Stanley Kobylarz, Contractor (Electrical) Matityahu Yeminy, Contractor (Mechanical) Phillip Braaten, Reactor Inspector (Training) George MacDonald, Senior Reactor Analyst
Approved by:	Rebecca L. Nease, Chief Engineering Branch 1 Division of Reactor Safety

SUMMARY

IR 05000348/2014007 and 05000364/2014007; 05/05/2014 – 06/06/2014; Joseph M. Farley Nuclear Plant, Units 1 and 2; Component Design Bases Inspection.

This inspection was conducted by a team of five Nuclear Regulatory Commission (NRC) inspectors from Region II and Region III, and two NRC contract personnel. One traditional violation, and seven Green non-cited violations (NCVs) were identified. The significance of inspection findings is indicated by their color (Green, White, Yellow, Red) using the NRC Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," dated June 2, 2011. Cross-cutting aspects are determined using IMC 0310, "Components Within the Cross Cutting Areas," dated December 19, 2013. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy, dated July 9, 2013. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 5, dated February 2014.

Cornerstone: Mitigating Systems

- Green. The team 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 perform an adequate operability evaluation following the discovery that the component cooling water miscellaneous user isolation valves would not isolate the safety-related piping from the non-safety related portion. The licensee entered the issue into their corrective action program as condition report 823056. In 2013, the valve actuators were modified from air to open and close, to a spring to close design so this is not a current operability issue.

  The team determined that the failure to perform an adequate operability evaluation as required by NMP-AD-012, "Operability Determinations and Functionality Assessments," was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of Equipment Performance and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the inspectors had reasonable doubt on the past operability of component cooling water because the operability evaluation relied on assumptions that were not correct, regarding the ability to establish make-up water to the on-service component cooling water train. The team performed a significance screening of this finding using the guidance provided in IMC 0609, "Significance Determination Process," Attachment 0609.04, "Initial Characterization of Findings." The team determined the finding required a detailed risk evaluation in accordance with Exhibit 2, "Mitigating Systems Screening Questions," and Exhibit 4, "External Event Screening Questions." A risk analysis was completed by a regional senior reactor analyst in accordance with the guidance of NRC IMC 0609 Appendix A. A bounding analysis was performed using Farley site specific seismic data and a conditional core damage probability determined using the NRC Farley SPAR PRA model. In addition, NUREG/CR6544 and NUREG/CR4550 show SSC fragility data for generic component types. From Table 1 Generic Seismic Fragilities the data shows that offsite power would be affected at 0.3G, electrical equipment and large flat bottomed storage tanks at approx. 1G, heat exchangers at 1.9 G with motor driven pumps at 2.0 G and piping at 3.8G. The major analysis assumptions included: a one year exposure period, no credit for the reactor coolant pump (RCP) shutdown seals, the performance deficiency was assumed to result in lowering surge tank level and subsequent common cause failure of all three CCW pumps with no recovery, and the miscellaneous header piping and components were assumed to fail from a seismic event of magnitude 0.3 – 0.5 G. The dominant sequence was a loss of RCP seal cooling resulting in an RCP seal LOCA caused by loss of CCW. The risk was mitigated by the low frequency of the seismic initiating event. The analysis determined that the risk increase due to the performance deficiency was an increase in core damage frequency of < 1E-6/year, a GREEN finding of very low safety significance. The team did not identify a cross-cutting aspect associated with this finding because this performance deficiency was not indicative of present licensee performance. (Section 1R21.2b.1)

- Green. The team identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to demonstrate compliance with IEEE 308-1971 for the required independence of 120V vital AC distribution system channels. The licensee entered the issue into their corrective action program as condition report 820528 and performed an immediate determination of operability and determined that the inverters were operable but non-conforming.

  The team determined that the failure to conform to the independence requirements of IEEE 308-1971, to which the licensee was committed, was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of Equipment Performance and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the finding resulted in a condition where there was a reasonable doubt of the operability of the 120V vital AC distribution system channels. In addition, the performance deficiency is similar to example 3j of IMC 0612, Appendix E, "Examples of Minor Issues." The team determined that the finding was of very low safety significance (Green) because it was not a design deficiency resulting in the loss of functionality or operability. The team did not identify a cross-cutting aspect associated with this finding because this performance deficiency is not indicative of present licensee performance. (Section 1R21.2b.2)

- Green. The team identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," for the licensee's failure to promptly correct a lack of documented verification and validation for time critical operator actions which are inputs into design basis plant safety analyses. The licensee entered the issue into their corrective action program as condition report 823401. Initial time validations of the more limiting time critical operator actions have been completed and the remaining Updated Final Safety Analysis Report (UFSAR) described time critical operator actions have been identified and scheduled for validation.

  The team determined the licensee's failure to promptly correct a lack of documented verification and validation for time critical operator actions, which are inputs into design basis plant safety analysis was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of Design Control and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the programmatic failure to ensure design basis operator actions could be accomplished within required time limits could impact the availability and capability of systems that respond to initiating events and result in unanalyzed plant conditions. The team determined that the finding was of very low safety significance (Green) because it was not a design deficiency resulting in the loss of functionality or operability. The team determined this finding was associated with the cross-cutting aspect of Evaluation in the area of Problem Identification and Resolution because following the identification of this deficiency in 2012, the licensee did not adequately evaluate the current operability for mitigating SSCs reliant upon these time critical operator actions described in the UFSAR. [P.2] (Section 1R21.2b.3)

- Green. The team 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 include an appropriate acceptance criterion for ultimate heat sink (UHS) temperature in surveillance procedures. Specifically, the acceptance criterion did not account for instrument uncertainty. The licensee entered the issue into their corrective action program as condition report 810638. As an immediate corrective action, the licensee established an action tracking item for control room operators to declare UHS inoperable if indicated temperature exceeded 90 degrees Fahrenheit. In addition, the licensee performed a historic review and did not find an example where the technical specifications (TS) temperature limit of 95 degrees Fahrenheit was exceeded.

  The team determined the failure to include appropriate acceptance criterion for UHS temperature in surveillance procedures was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of Equipment Performance and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of the UHS system to respond to initiating events to prevent undesirable consequences. Specifically, the failure to account for UHS temperature instrument uncertainty was significant enough to require revision of the associated surveillance procedures to ensure the validity of heat exchanger performance calculations and compliance with TS limits. The team determined the finding was of very low safety significance (Green) because it was not a design deficiency resulting in the loss of functionality or operability. The team did not identify a cross-cutting aspect associated with this finding because it is not indicative of present licensee performance. (Section 1R21.2b.4)

- Green. The team identified a Green non-cited violation of 10 CFR 50.55a(f), "Inservice testing requirements," subsection (4), American Society of Mechanical Engineers Operation and Maintenance of Nuclear Power Plants code, Subsection ISTC-5221, “Check Valves,” with two examples for the licensee’s failure to incorporate adequate acceptance criteria for testing safety-related check valves into the procedures. The licensee entered both examples into their corrective action program as condition reports 816150 and 816303. A review of past pump data and testing indicated the check valves caused no degradation to the high-head safety injection system.

  The team determined the failure to establish acceptance criteria that demonstrates closure of safety-related check valves was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of Design Control and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, testing Unit 1 & 2 refueling water storage tank (RWST) supply to charging header check valves (Q1/2E21V026) using an acceptance criterion of boric acid tank pump discharge pressure greater than 80 psig (normally 115+ psig) with no change in boric acid tank level, may have resulted in the check valves not seating and allowed reverse flow to the RWST. In addition, using an acceptance criterion of no reverse rotation of the charging pump impeller when testing the Unit 1 & 2 charging pump mini-flow check valves (Q1/2E21V0121) and Unit 1 & 2 charging pump discharge check valves (Q1/2E21V0122) may result in the check valves not seating and challenge high head safety injection flow. The team determined that the finding was of very low safety significance (Green) because it was not a design deficiency resulting in the loss of functionality or operability. The team did not identify a cross-cutting aspect associated with this finding because it is not indicative of present licensee performance. (Section 1R21.2b.5)

- <u>Green</u>. The team identified a Green non-cited violation of 10 CFR 50.55a(f), "Inservice testing requirements," subsection (4), American Society of Mechanical Engineers Operation and Maintenance of Nuclear Power Plants code, Subsection ISTC-1300, "Valve Categories," for the licensee's failure to categorize Unit 1 & 2 charging pump suction isolation valves (LCV115 B &D), and Unit 1 & 2 refueling water storage tank (RWST) supply to charging header check valves (Q1/2E21V026) as Class "A" for which seat leakage is limited to a specific maximum amount in the closed position. Specifically, the licensee's inservice testing program did not test safety-related valves to ensure they could perform their safety function in the closed direction and meet seat leakage requirements. The licensee entered the issue into their corrective action program as condition reports 823022 and 815699. A review of past pump data indicated the valve held against system pressure and would not allow a significant reverse flow.

  The team determined that failure of the licensee to properly categorize LCV115 B & D and QV026 in their inservice testing program to ensure they could perform their safety function was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of Design Control and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to properly categorize valves as Category "A" resulting in failure to leak test the valves to ensure reverse flow of containment sump water to the RWST did not result in exceeding the plant's post accident dose rate limits. The team determined the finding was of very low safety significance (Green) because it was not a design deficiency resulting in the loss of functionality or operability. The team did not identify a cross-cutting aspect associated with this finding because it is not indicative of present licensee performance. (Section 1R21.2b.6)

- <u>Severity Level IV</u>. The team identified a Severity Level (SL) IV non-cited violation of 10 CFR 50.71, "Maintenance of Records, Making of Reports," for the licensee's failure to update the Updated Final Safety Analysis Report (UFSAR). Specifically, the UFSAR was not updated to reflect the analysis requested by the NRC in GL 2008-01, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems." The licensee entered the issue into the corrective action program as condition report 823270.

  The team determined the failure to update the UFSAR with the analyses performed for GL 2008-01 was a performance deficiency. Failures to update the UFSAR are dispositioned using the traditional enforcement process instead of the SDP in accordance with IMC 0612, Appendix B, Block TE2, because they potentially impede or impact the regulatory process. Specifically, failures to update the UFSAR challenges the regulatory process because it serves as a reference document used, in part, for recurring safety analyses, evaluating license amendment requests, and in preparation for and conduct of inspection activities. As a result, the team compared the performance deficiency against the examples in Section 6.1 of the NRC Enforcement Policy and determined it constituted a more than minor traditional enforcement violation because it rose to a SL-IV violation. Specifically, SL-IV violation example d.3 stated "A licensee fails to update the UFSAR as required by 10 CFR 50.71(e) but the lack of up-to-date information has not resulted in any unacceptable change to the facility or procedures." The team determined an evaluation for cross-cutting aspect was not applicable because this was a traditional enforcement violation. (Section 1R21.3b.1)

- Green. The team identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to ensure the residual heat removal (RHR) system would be capable to respond to a MODE 4 loss of coolant accident (LOCA). Specifically, low pressure coolant injection may not be available during MODE 4, which is required for a large break LOCA. The licensee entered the issue into their corrective action program as condition report 826059. As an immediate corrective action, the licensee performed an extent of condition to identify other deficient procedures. In addition, the licensee implemented action tracking items in the control room to limit one train of decay heat removal operation while above 212 degrees Fahrenheit.

  The team determined that the failure to ensure that RHR would be capable to respond to a LOCA that initiates in MODE 4 as required by TS 3.5.3., "ECCS - Shutdown," was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Mitigating System cornerstone attribute of Equipment Performance and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, procedures and design for the RHR system did not ensure the capability to perform its emergency core cooling system mitigating function of low pressure injection while in MODE 4 because steam void formation could occur and was not evaluated. The finding was screened in accordance with NRC Inspection Manual Chapter (IMC) 0609 Attachment 4 and was transitioned to IMC 0609 Appendix G as the finding represented a degraded condition, which could occur only during shutdown conditions. NRC IMC 0609 Appendix G Attachment 1 screening determined that the finding represented a potential loss of system safety function and required a phase 2 shutdown risk assessment. A bounding phase 2 shutdown risk assessment was performed by a regional senior reactor analyst in accordance with NRC IMC 0609 Attachment 2. The major assumptions in the analysis included an exposure interval of 5 minutes for Unit 1 only and a bounding conditional core damage probability of 1.0 given a LOCA. The risk was mitigated by the short exposure period and the low probability of a LOCA during shutdown conditions. The result of the analysis was an increase in core damage frequency of < 1E-6/year a GREEN finding of very low safety significance. The team did not identify a cross-cutting aspect associated with this finding because it is not indicative of present licensee performance. (Section 1R21.3b2)

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R21 Component Design Bases Inspection (71111.21)

.1 Inspection Sample Selection Process

The team selected risk-significant components for review using information contained in the licensee's probabilistic risk assessment. In general, this included components that had a risk achievement worth factor greater than 1.3 or Birnbaum value greater than 1E-6. The sample included 16 components, one of which was associated with containment large early release frequency (LERF), and five operating experience (OE) items.

The team performed a margin assessment and a detailed review of the selected risk-significant components to verify that the design bases had been correctly implemented and maintained. Where possible, this margin was determined by the review of the design basis and Updated Final Safety Analysis Report (UFSAR). This margin assessment also considered original design issues, margin reductions due to modifications, or margin reductions identified as a result of material condition issues. Equipment reliability issues were also considered in the selection of components for a detailed review. These reliability issues included items related to failed performance test results, significant corrective action, repeated maintenance, maintenance rule status, Manual Chapter 0326 conditions, NRC resident inspector input regarding problem equipment, system health reports, industry OE, and licensee problem equipment lists. Consideration was also given to the uniqueness and complexity of the design, OE, and the available defense-in-depth margins. An overall summary of the reviews performed and the specific inspection findings identified is included in the following sections of the report.

.2 Component Reviews

a. Inspection Scope

Components

- component cooling water (CCW) heat exchangers (Q1/2P17H0001)
- CCW pumps (Q1/2P17001)
- CCW surge tanks (Q1/2P17T0001)
- service water (SW) isolation valves (Q1/2P16V515, Q1/2P16V517)
- CCW isolation valves (Q1/2P173096 A/B)
- refueling water storage tanks (RWST) (Q1/2F16T0501)
- RWST outlet valves (Q1/2LCV115 B/D)
- high head safety injection (HHSI)/charging pumps (Q1/2E21P0002)
- volume control tank outlet isolation/check valves (Q1/2LCV 115C/E, Q1/2E21V0188)
- 4 kV Bus (Q1/2R15A0006 and Q1/2R15A007)
- HHSI/charging pump motors (Q1/2E21M0001)

- 120V AC vital distribution system inverters and distribution panels (Q1/2R21E009A/B/C/D)
- 1A/1B startup transformer (N1/2R11A0501, N1/2R11A0501)
- sequencers (Q1/2R43E0001, Q1/2R43E0501)
- RWST level Instrumentation (Q1/2F16T0501)

<u>Components with LERF Implications</u>
- HHSI valves (Q1/2E21MOV8107, Q1/2E21MOV8108)

For the 16 components listed above, the team reviewed the plant technical specifications (TS), UFSAR, design bases documents (DBDs), and drawings to establish an overall understanding of the design bases of the components. Design calculations and procedures were reviewed to verify that the design and licensing bases had been appropriately translated into these documents. Test procedures and recent test results were reviewed against DBDs to verify that acceptance criteria for tested parameters were supported by calculations or other engineering documents, and that individual tests and analyses served to validate component operation under accident conditions. Maintenance procedures were reviewed to ensure components were appropriately included in the licensee's preventive maintenance program. System modifications, vendor documentation, system health reports, preventive and corrective maintenance history, and corrective action program documents were reviewed (as applicable) in order to verify that the performance capability of the component was not negatively impacted, and that potential degradation was monitored or prevented. Maintenance Rule information was reviewed to verify that the component was properly scoped, and that appropriate preventive maintenance was being performed to justify current Maintenance Rule status. Walkdowns for accessible components and interviews were conducted to verify that the installed configurations would support their design and licensing bases functions under accident conditions and had been maintained to be consistent with design assumptions.

Additionally, the team performed the following component-specific reviews:

- The team reviewed the capability of the CCW design to withstand the effects of a reactor coolant pump (RCP) thermal barrier break.
- The team reviewed the capability of the CCW design to isolate a failure of the non-seismic piping portion of the system.
- The team reviewed Generic Letter (GL) 2008-01 modifications to assess charging pumps' suction and discharge lines do not contain gas and HHSI pump operation is maintained.
- The team reviewed HHSI pumps control logic during different modes of operation to asses the pumps will operate during the most limiting conditions.
- The team reviewed system logic and interlocks for pump protection and verified the interlocks are tested.
- The team reviewed the potential reverse flow testing requirements of HHSI check valves.
- The team reviewed Farley's evaluation of Information Notice (IN) 2013-18 to assess if potential weld fabrication flaws, stress corrosion cracking, and high stress low cycle fatigue were considered for the RWST and associated piping.

- The team reviewed vortexing calculations to asses if air vortices formed for various combinations of suction for the operating emergency core cooling system (ECCS) pumps.
- The inspectors reviewed the RWST design features for protection from tornado missiles.
- The team reviewed the valves and check valves that prevent reverse flow of highly radioactive water to the RWST.
- The team reviewed motor operated valve (MOV) LCV115B and LCV115D thermal overload relay heater sizing calculations to ensure the heaters were sized correctly and would operate in any design basis accident sequence.
- The team reviewed the, equipment specifications, voltage tap settings, short circuit and voltage drop calculations, protective relay settings, and loading for the transformer.
- The team reviewed the transformer protective relay trip setting calculations to verify adequate primary and backup protections and appropriate coordination margins between upstream and downstream protective devices.
- The team reviewed the logic circuit drawings to verify load shed and load sequencing signals are in accordance with design basis.
- The team reviewed the calculations for loading and voltage drop for the vital inverters, distribution panels, and pump and valve motors, to ensure that sufficient capacity exists for normal and accident loading, and sufficient voltage was available for all loads.
- The team reviewed the 4kV switchgear incoming line breaker settings and coordination with emergency diesel generator breaker and load breakers.
- The team reviewed the Kraus & Naimer (K&N) control switch replacement evaluations to ensure functional capability under limiting plant voltage conditions.
- The team observed a simulator scenario to align containment sump recirculation
- The team observed a simulator scenario to trip two of three reactor coolant pumps following a loss of normal feed water.
- The team observed a simulator scenario to unblock a power-operated relief valve (PORV) to ensure one pressurizer PORV is available following an inadvertent safety injection.

b. Findings

.1 Inadequate Operability Evaluation of the CCW Miscellaneous User Isolation Valves

Introduction: The team identified an Green non-cited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” for the licensee’s failure to perform an adequate operability evaluation following the discovery that CCW miscellaneous user isolation valves would not isolate the safety-related CCW piping from the non-safety related (NSR) portion. Specifically, the licensee incorrectly determined that CCW remained operable.

Description: The team reviewed TS section 3.7.7, “Component Cooling Water System,” which states, “Two CCW trains shall be operable.” Section 9.2.2 of the UFSAR, “Cooling System for Reactor Auxiliaries,” states, “All portions of the component cooling system that are safety-related are Seismic Category I design.” In addition, Section 9.2.2 states, in part, “…valves in the supply and return lines for NSR equipment will be automatically closed by a low-low level signal in the surge tank or remote manually from the control room, by closing air-operated valves Q1/2P17HV3096A & B (HV3096A/B), which are the CCW nonessential user isolation vales." NMP-AD-012, "Operability Determinations and Functionality Assessments," Step 4.10 states, in part, that "…when system capability is degraded to a point where it cannot perform with reasonable expectation or reliability, the system should be judged inoperable, even if at this instantaneous point in time the system could provide the specified safety function."

In April of 2006, the licensee changed the cushion regulator setpoint pressure for HV3096A/B on Unit 1, based on new air operated valve software (Kalsi Engineering Valve and Actuator Program (KVAP)) to determine the setpoints, thrust, and maximum air supply to the valves. Similarly, the licensee implemented the new setpoint in May and August of 2006, for HV3096A and HV3096B on Unit 2, respectively. On February 5, 2008, the licensee discovered the cushion regulator was incorrectly modeled in the KVAP software. As a result, the cushion regulator setpoint pressure was too low to ensure full closure of the valve disc, resulting in a leak rate of 30 gallons per minute (gpm) per valve for a total of 60 gpm. The station corrected the issue by increasing the cushion regulator setpoint on March 6, 2008. The inadequate setpoint change issue was previously dispositoned as a licensee-identified violation in the Farley integrated inspection report 2008002.

The licensee captured this issue in their Corrective Action Program (CAP) as condition report (CR) 2008101055 and performed an operability determination. This evaluation concluded CCW remained operable, by crediting the operators' ability to detect lowering surge tank level and align makeup to CCW prior to CCW surge tank drainage. As a result, the licensee determined that compensatory measures were not required, the issue did not represent a reportable condition per 10 CFR 50.73, "Licensee Event Report System," and the system remained in compliance with TS. In an effort to minimize leakage, the licensee immediately closed the valves with a jacking device.

The team noted the licensee failed to consider the following issues: 1) the operators' response time to establish makeup to the surge tank; 2) the surge tank low level (30 inches) alarms were not seismically qualified; and 3) the valves would be open at the beginning of a seismic event resulting in an initial loss of inventory an order of magnitude greater than the anticipated 60 gpm leak. The licensee validated this activity and determined the operators would require 10 to 11 minutes to establish makeup once lowering surge tank level was detected. The team determined the tank and associated piping would drain prior to establishing make-up to the CCW surge tank (2000 gallon capacity tank, 60 inches), causing the on-service (train connected to NSR piping) CCW pump to cavitate. Although the indication of low surge tank level alarms is at 30 inches CCW surge tank level, there is also a divider plate that begins at 30 inches. So, 500 gallons of approximately 1000 gallons of the tank volume is available and the valves would not begin to isolate until CCW surge tank level reached 17 inches (approximately 285 gallons). In addition, the HV3096A/B allowed a total of 60 gpm leak by and would not fully isolate. The CCW flow rate to the NSR section of piping is around 800 gpm. Based on the flow rate to the NSR piping, the HV3096A/B leak rate, a stroke time for HV3096 of 10 seconds, and an available tank volume of 285 gallons, the team determined that make-up would not be established fast enough to ensure the on-service CCW pump supplying the NSR piping would remain operable during a seismic event. As a result, the team concluded the licensee did not perform an adequate operability evaluation and, since TS compliance was impacted, a reportable condition existed per 10 CFR 50.73, and compensatory actions were required to ensure operability at the time of discovery. Because the licensee did not recognize the condition, per NRC enforcement policy section 2.2.1.d, a 50.73 violation is not being issued. The licensee entered the issue into their CAP as CR 823056. The valve actuators were previously modified from air to open and close, to a spring to close design so this is not a current operability issue.

Analysis: The team determined that the failure to perform an adequate operability evaluation as required by NMP-AD-012, "Operability Determinations and Functionality Assessments," was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of Equipment Performance and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the team had reasonable doubt on the past operability of component cooling water because the operability evaluation relied on assumptions that were not correct, regarding the ability to establish make-up water to the on-service component cooling water train. In addition, the performance deficiency is similar to example 3k of IMC 0612, Appendix E, "Examples of Minor Issues," because inspectors had reasonable doubt on the past operability of the CCW system.

The team performed a significance screening of this finding using the guidance provided in IMC 0609, "Significance Determination Process," Attachment 0609.04, "Initial Characterization of Findings." The team determined the finding required a detailed risk evaluation in accordance with Exhibit 2, "Mitigating Systems Screening Questions," and Exhibit 4, "External Event Screening Questions." A risk analysis was completed by a regional senior reactor analyst in accordance with the guidance of NRC IMC 0609 Appendix A. A bounding analysis was performed using Farley site specific seismic data and a conditional core damage probability determined using the NRC Farley SPAR PRA model. In addition, "NUREG/CR6544 and NUREG/CR4550 show SSC fragility data for generic component types. From Table 1 Generic Seismic Fragilities the data shows that offsite power would be affected at 0.3G, electrical equipment and large flat bottomed storage tanks at approx. 1G, heat exchangers at 1.9 G with motor driven pumps at 2.0 G and piping at 3.8G. The major analysis assumptions included: a one year exposure period, no credit for the reactor coolant pump (RCP) shutdown seals, the performance deficiency was assumed to result in lowering surge tank level and subsequent common cause failure of all three CCW pumps with no recovery, and the miscellaneous header piping and components were assumed to fail from a seismic event of magnitude 0.3 – 0.5 G. The dominant sequence was a loss of RCP seal cooling resulting in an RCP seal LOCA caused by loss of CCW. The risk was mitigated by the low frequency of the seismic initiating event. The analysis determined that the risk increase due to the performance deficiency was an increase in core damage frequency of < 1E-6/year, a GREEN finding of very low safety significance.

The team did not identify a cross-cutting aspect associated with this finding because this performance deficiency is not indicative of present licensee performance. Specifically, the licensee's operability evaluation was performed more than three years ago.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality be prescribed by documented procedures of a type appropriate to the circumstances and be accomplished in accordance with these procedures. The licensee established NMP-AD-

012, “Operability Determinations and Functionality Assessments,” as the implementing procedure for determining and documenting SSC operability, an activity affecting quality. Step 4.10 states, in part, that “…when system capability is degraded to a point where it cannot perform with reasonable expectation or reliability, the system should be judged inoperable, even if at this instantaneous point in time the system could provide the specified safety function.”

Contrary to the above, on February 6, 2008, the licensee failed to follow step 4.10 of procedure NMP-AD-012. Specifically, the licensee did not declare the component cooling water on service train inoperable, because the on-service CCW train would not have remained reliable. This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. The violation was entered into the licensee’s CAP as CR 823056. (NCV 05000348, 364/2014007-01, Inadequate Operability Evaluation of the CCW Miscellaneous User Isolation Valves).

.2 Failure to Comply with IEEE 308-1971 for the Required Independence of 120 Volt Vital AC Distribution System Channels

Introduction: The team identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the licensee’s failure to demonstrate compliance with IEEE 308-1971 for the required independence of 120V vital AC distribution system channels.

Description: Section 8.3.1.2(E) of the UFSAR requires compliance with IEEE 308-1971, which states that, “All components of the Class 1E electric system (discussed in UFSAR paragraph 8.3.1.1) are designed to meet their functional requirements under conditions produced by the design basis events.” UFSAR section 8.3.1.1.4 described the 120V vital instrument power system, in part, as follows, “Four redundant channelized 120 V-ac vital instrumentation distribution panels are provided for each unit to supply power for essential instrumentation and control loads under all operating conditions (see drawings D-177024, D-207024, D-177025, and D-207025). Each distribution panel is supplied separately from a static inverter.”

The team determined the licensee failed to evaluate the impact of NSR circuit failures on the safety-related vital instrumentation and control systems. These failures could adversely impact the B and D (Class 1E) inverters in vital AC distribution channels 2 and 4. These safety-related inverters could go into current limiting conditions due to seismically induced faults on NSR circuits. The team found that only vital AC distribution channels 2 and 4 contained NSR circuits and, therefore, were the only channels that were susceptible to seismically induced circuit failures.

Specifically, the vital distribution channels 2 and 4 are susceptible to a common failure mode, as defined in IEEE-308-1971, due to a lack of sufficient output voltage from the inverter during current limiting conditions. The inverter could experience current limiting conditions during a seismic event that could cause the failure of non-seismically qualified NSR circuits that are part of the design. Additionally, concurrent with a seismic event, a loss of offsite power/loss of station power (LOOP/LOSP) can be postulated, which eliminates the backup constant voltage transformer (CVT) for a short period of time and removes the CVT’s ability to interrupt the circuit faults. The CVT that provides backup power to the inverter system, is being powered from the station emergency diesel generator (EDG), and is not available to provide the backup power to interrupt electrical fault conditions until approximately 12 seconds into the event. During the 12 seconds without power to the CVT, the safety-related inverter must provide power for circuit protective devices (i.e. fuses or circuit breakers) to operate and clear the NSR circuit faults. The licensee determined that during the 12 seconds without power, the inverter could go into a current limiting condition that could result in a very low inverter output voltage and the potential for inadvertent instrumentation actuation.

Prior to the inspection, Farley had not evaluated this condition to determine whether the safety-related vital instrumentation and control could perform as designed during a seismic event. The licensee entered this issue into their CAP as CR 820528 and performed an immediate determination of operability, which concluded the 120V vital AC system remained operable but non-conforming.

<u>Analysis</u>: The team determined that the failure to conform to the independence requirements of IEEE 308-1971, to which the licensee was committed, was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of Equipment Performance and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the finding results in a condition where there was a reasonable doubt of the operability of the 120V vital AC distribution system channels. Additionally, the performance deficiency is similar to example 3j of IMC 0612, Appendix E, "Examples of Minor Issues." The team used IMC 0609, Att. 4, "Initial Characterization of Findings," issued June 19, 2012, for Mitigating Systems cornerstone, and IMC 0609, App. A, "The Significance Determination Process (SDP) for Findings At-Power," issued June 19, 2012, and the team determined that the finding was of very low safety significance (Green) because it was not a design deficiency resulting in the loss of functionality or operability. The licensee determined the 120V vital AC inverters were operable but non-conforming. The team did not identify a cross-cutting aspect associated with this finding because this performance deficiency is not indicative of present licensee performance. Specifically, non-safety related loads were added to the inverters more than three years.

<u>Enforcement</u>: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," states, in part, that design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. In addition, IEEE 308-1971, subsection 5.4.2, Design Requirements, states, in part, for Vital Instrumentation and Control Power Systems, that power must be supplied to these systems in such a manner as to preserve their reliability, independence, and redundancy. IEEE 308-1971, section 3, "Definitions," defines *independence* as "no common mode failure for any design basis event."

Contrary to the above, as of May 5, 2014, the licensee failed to verify the adequacy of the safety-related vital B and D (Class 1E) inverters in Channels 2 and 4 by either an analysis or testing. Specifically, the licensee failed to verify that the safety-related vital B and D (Class 1E) inverters in Channels 2 and 4 would continue to operate during a seismic event concurrent with a LOOP/LOSP, when the CVT was not available during the period before the EDG is supplying standby power to station busses. This was due to a lack of independence between NSR circuits and the safety-related inverters required by IEEE 308-1971. This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. The violation was entered into the licensee's CAP as

CR 820528. (NCV 05000348, 364/2014007-02, Failure to Comply with IEEE 308-1971 for the Required Independence of 120 Volt Vital AC Distribution System Channels)

.3 Failure to Correct Lack of Validated Time Critical Operator Actions Analyses

Introduction: The team identified a Green NCV of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” for the licensee’s failure to promptly correct a lack of documented verification and validation for time critical operator actions, which are inputs into design basis plant safety analyses.

Description: The Farley UFSAR describes several time critical operator actions for accident mitigation. These time critical operator actions should be validated in order to ensure the plant is able to operate within the design basis assumptions used in various safety analyses. The Southern Nuclear Company (SNC) Quality Assurance Topical Report, Paragraph II.3-2, “Design Records,” requires that SNC maintain records sufficient to provide evidence that the design was properly accomplished. These records include the final design output and any revisions thereto, as well as records of the important design steps (e.g., calculations, analyses, and computer programs) and the sources of input that support the final output. The team requested records of time validations for a sample of three time critical operator actions described in the UFSAR and the licensee was unable to provide any documentation to demonstrate the ability to accomplish these actions in the assumed times.

A review of CAP documents requested by the team revealed multiple CRs written by SNC Corporate as well as the site licensee, dating back to September 2012. Specifically, Farley CR 521579 (September 2012) identified that some time critical operator actions had not been validated to ensure the times assumed in the licensing and design bases documents and analysis could be met. No immediate actions were taken at that time to verify the existence of documents, which would support reasonable assurance of operability. Later, SNC Corporate issued CR 733032 (November 2013) documenting elevated concerns to Farley for a failure to make continued forward progress on a Time Critical Operator Actions program. Again, no immediate actions were taken to verify current ability to accomplish these design basis actions. Following a delay in the issuance of SNC Fleet guidance for a Time Critical Operator Action program, Farley began implementing procedure NMP-OS-014, “Time Critical Operator Action Program,” in 2013. However, a majority of operator actions listed in the UFSAR and other design and licensing documents were not yet validated or incorporated into the program.

In order to validate the current time assumptions contained in the UFSAR, the team observed simulator scenarios for a loss of normal feed water event, an inadvertent safety injection/pressurizer filling event, and the transfer to cold leg recirculation following a large break LOCA. The licensed operator crew was able to demonstrate that the times assumed in the safety analyses could be accomplished in accordance with current procedures. Additional time critical operator actions specified in the UFSAR were identified by the licensee and scheduled to be validated prior to June 30, 2014.

The team determined the licensee had failed to promptly correct a condition adverse to quality in that the licensee had not conducted an adequate immediate operability evaluation to determine whether the time critical operator actions could be accomplished within the time assumptions. Licensee procedure NMP-AD-012, “Operability

Determinations and Functionality Assessments," requires an initial determination of operability after confirming the existence of a degraded, nonconforming or unanalyzed condition. The determination should be made without delay and the information should be sufficient to conclude that there is a reasonable expectation that an SSC is operable. From August 2012 when this issue was identified, until the team arrived onsite, an adequate immediate determination of operability was not performed to provide reasonable assurance the time critical operator actions described in the UFSAR could be accomplished despite the lack of documentation or validations. No tabletop reviews of the procedures or simulator walkthroughs were performed to provide a basis of reasonable assurance. Only six of 28 UFSAR time critical actions were validated to date and simulator scenarios observed during the inspection were completed successfully. When fully implemented, this program will provide future assurance that time critical actions can be accomplished within the assumed times listed in the UFSAR and safety analyses. The licensee entered this issue into their CAP as CR 823401.

<u>Analysis</u>: The team determined the licensee's failure to promptly correct a lack of documented verification and validation for time critical operator actions, which are inputs into design basis plant safety analysis was a performance deficiency. The performance deficiency was more than minor because it was associated with the Mitigating Systems cornerstone attribute of Design Control and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the programmatic failure to ensure design basis operator actions could be accomplished within required time limits could impact the availability and capability of systems that respond to initiating events and could result in unanalyzed plant conditions. In addition, the performance deficiency is similar to example 3j of IMC 0612, Appendix E, "Examples of Minor Issues." The team used IMC 0609, Att. 4, "Initial Characterization of Findings," issued June 19, 2012, for Mitigating Systems cornerstone, and IMC 0609, App. A, "The Significance Determination Process (SDP) for Findings At-Power," issued June 19, 2012, and determined that the finding was of very low safety significance (Green) because it was not a design deficiency resulting in the loss of functionality or operability. The team determined this finding was associated with the cross-cutting aspect of Evaluation in the area of Problem Identification and Resolution because following the identification of this deficiency in 2012, the licensee did not adequately evaluate the current operability for mitigating SSCs reliant upon these time critical operator actions described in the UFSAR. [P.2]

<u>Enforcement</u>: Title 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," states in part that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected.

Contrary to the above, since September 2012, the licensee failed to assure that a condition adverse to quality is promptly corrected. Specifically, the licensee did not correct the lack of documentation or validation of time critical operator actions described in the UFSAR. Initial time validations of the more limiting time critical operator actions were completed and the remaining UFSAR-described time critical operator actions have been identified and scheduled for validation. This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. The violation was entered into the licensee's CAP as CR 823401. (NCV 05000348, 364/2014007-03, Failure to Correct Lack of Validated Time Critical Operator Actions Analyses)

.4 Acceptance Criterion for Ultimate Heat Sink Temperature Did Not Consider Instrument Uncertainty

Introduction: The team identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” for the licensee’s failure to include an appropriate acceptance criterion for ultimate heat sink (UHS) temperature in surveillance procedures. Specifically, the acceptance criterion did not account for instrument uncertainty.

Description: As required by TS 3.7.9, “Ultimate Heat Sink,” the UHS shall be operable in MODEs 1, 2, 3, and 4. In order to ensure UHS operability, surveillance requirement 3.7.9.2 required the licensee to verify UHS water temperature was less than or equal to 95 degrees Fahrenheit (F). The licensee implemented this surveillance requirement via procedure FNP-1-STP-1.0, “Operations Daily and Shift Surveillance Requirements.”

While reviewing CCW heat exchanger performance calculations, the team noted they did not account for UHS temperature instrument uncertainty. Calculation SJ-C071863601-005, “Total Loop Uncertainty Calculation for the UHS Temperature Indicators,” determined the instrument uncertainty was 4.71 degrees F. The team noted procedure FNP-1-STP-1.0 used the associated TS limit value of 95 degrees F as the acceptance criterion and did not consider instrument uncertainty.

The licensee entered this issue into their CAP as CR 810638. As an immediate corrective action, the licensee established an action tracking item for control room operators to declare UHS inoperable if indicated temperature exceeded 90 degrees F to ensure the 95 degree F, including instrument uncertainty, is not exceeded. In addition, the licensee performed a historic review and did not find an example where TS temperature limits of 95 degrees F were exceeded when accounting for instrument uncertainty. The corrective action to restore compliance was to revise the associated acceptance criterion included in surveillance procedure FNP-1-STP-1.0 to account for instrument uncertainty.

Analysis: The team determined the failure to include appropriate acceptance criterion for UHS temperature in surveillance procedures was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of Equipment Performance and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of the UHS system to respond to initiating events to prevent undesirable consequences. Specifically, the failure to account for UHS temperature instrument uncertainty was significant enough to require revision of the associated surveillance procedure to ensure the validity of heat exchanger performance calculations and compliance with TS limits.

The team used IMC 0609, Att. 4,“Initial Characterization of Findings,” issued June 19, 2012, for Mitigating Systems cornerstone, and IMC 0609, App. A, “The Significance Determination Process (SDP) for Findings At-Power,” issued June 19, 2012, and the team determined that the finding was of very low safety significance (Green) because it was not a design deficiency resulting in the loss of functionality or operability. Specifically, a historic review did not find an example where TS temperature limits were exceeded. The team did not identify a cross-cutting aspect associated with this finding because it was not indicative of present licensee performance. Specifically, the affected procedure was developed more than three years ago.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” states in part, “…instructions, procedures, or drawings. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.”

Contrary to the above, as of May 5, 2014, the licensee failed to include appropriate acceptance criteria in surveillance procedures for the UHS. Specifically, the UHS temperature acceptance criterion included in surveillance procedure FNP-1-STP-1.0 did not account for instrument uncertainty to ensure compliance with the TS temperature limit.

The licensee is still evaluating its planned corrective actions; however, the team determined that the continued non-compliance does not present an immediate safety concern because the licensee established an action tracking item for control room operators to declare UHS inoperable if indicated temperature exceeded 90 degrees F. This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. The violation was entered into the licensee’s CAP as CR 820528. (NCV 05000348, 364/2014007-04, Acceptance Criterion for UHS Temperature Did Not Consider Instrument Uncertainty)

.5 Inadequate Acceptance Criteria for Testing of Check Valves

Introduction: The team identified a Green NCV of 10 CFR 50.55a(f), "Inservice testing requirements," subsection (4), American Society of Mechanical Engineers (ASME) Operation and Maintenance of Nuclear Power Plants (OM) code Subsection ISTC-5221, “Check Valves” with two examples for the licensee’s failure to incorporate adequate acceptance criteria for testing safety-related check valves into the procedures.

Description: The team reviewed surveillance procedure FNP-1/2-STP-4.10, “Reverse Flow Test of RWST to Charging Pump Check Valve and CVCS Emergency Borate Filter to Charging Pump Suction Check Valve.” Section 2.0 of the surveillance test procedure included acceptance criterion for demonstrating the check valve from the Units 1 & 2 RWST to the charging pump suction (Q1/2E21V026) is closed by aligning the boric acid transfer pump to the system downstream of check valve (QV026); and verifying the discharge pressure remains greater than 80 psig and no detectable level drop in the associated boric acid tank. The team determined the acceptance criterion was inadequate because the boric acid tank pump’s head-capacity curve indicated the pump discharge pressure would always be greater than 80 psig, thereby satisfying this acceptance criterion, even if check valve QV026 failed open. The team determined based on the pump curve, the lowest achievable boric acid tank pump discharge pressure is 103 psig. In addition, the boric acid tank level indication is measured in increments of 2% level. The team determined that 1% of water level height is equivalent to190 gallons of water. According to the procedure, the readings are taken at 10 minute intervals. If the highest level of accuracy is 1% tank level deviation, a reverse flow rate of more than 19 gpm through this check valve could occur while still indicating that the check valve is fully seated. This would not satisfy the ASME OM code requirement of ensuring the safety-related function to prevent back flow into the RWST is met.

The second example was identified while reviewing surveillance procedures FNP-1/2-STP-4.1, 4.2, 4.3, “1/2 Charging Pump Quarterly Inservice Test” for trains A, B and C, respectively. The acceptance criterion in section 2.7 used identification of reverse rotation for the charging pump impeller to indicate whether the check valve failed to close. This criterion was used for the Units 1 & 2 charging pump miniflow line check valves (Q1/2E21V0121A/B/C), and the charging pump discharge check valves (Q1/2E21V0122A/B/C). Step 4.6 of the surveillance test procedures required, in part, that reverse flow testing be performed on check valves Q1/2E21V0121A/B/C (QV121) and Q1/2E21V0122A/B/C (QV122). Step 5.24 of the surveillance test procedure required the operators to check the stopped pump for [reverse] rotation, and to declare check valves QV121 and QV122 inoperable if the pump was reverse rotating. The licensee identified in 2004, as documented in CR 2004100022, that the reverse flow rate through the eleven-stage high-head charging pump would have to be significant to cause reverse rotation and the acceptance criteria was inadequate. The CR stated, “The IST plan currently credits STP-4.1, 4.2, 4.3 which use pump reverse rotation as the acceptance criteria. However, this method was not adequate to detect the leakage which led to gas buildup on 2B.” The team determined the boric acid tank pump flow could never be high enough to cause the HHSI pump impeller to move in the reverse direction.

The team determined the acceptance criteria in both examples would not have ensured the check valves’ operation in the closed direction by ensuring the obturator seated. The licensee entered both issues into their CAP as CRs 816150 and 816303. The team determined testing performed on QV026 indicated the valve caused no reduction in pump discharge pressure when introducing test flow to the valve in the reverse flow direction, indicating the valve held against system pressure. Therefore, no immediate operability concern exists for QV026. In addition, the comprehensive flow test for the HHSI system indicates adequate flow downstream of the miniflow check valves, and discharge check valves ensuring operability and functionality of the HHSI system. Therefore, no operability concerns exist regarding QV121 and QV122.

Analysis: The team determined failure to establish acceptance criteria that demonstrates closure of safety-related check valves was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of Design Control and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, using acceptance criteria of greater than 80 psig boric acid tank pump discharge pressure and no change in boric acid tank level may result in undetected significant reverse flow to the RWST. In addition, using an acceptance criterion of no reverse rotation on an eleven-stage pump for QV121 and QV122 may have resulted in reduced HHSI flow of the operating pump through the idle pump and allow gas intrusion into the system.

The team used IMC 0609, Att. 4, “Initial Characterization of Findings,” issued June 19, 2012, for Mitigating Systems cornerstone, and IMC 0609, App. A, “The Significance Determination Process (SDP) for Findings At-Power,” issued June 19, 2012, and determined that the finding was of very low safety significance (Green) because it was not a design deficiency resulting in the loss of functionality or operability. Specifically, the pump data during QV026 testing indicated the valve caused no reduction in boric acid tank pump discharge pressure when introducing test flow to the valve in the reverse flow direction; indicating the valve held against system pressure and limited leakage would be allowed in the reverse direction. Therefore, no immediate operability concern exists for QV026. In addition, the comprehensive flow test for the high-head safety injection system indicated adequate flow downstream of the QV121, and QV122 ensuring operability and functionality of the high-head safety injection system. Therefore, no operability concerns exist regarding QV121, and QV122.

The team did not identify a cross-cutting aspect with this finding because it is not indicative of present licensee performance. The inadequate acceptance criterion were established more than three years ago.

Enforcement: Title 10 CFR 50.55a(f), "Inservice testing requirements," subsection (4) states in part, that pumps and valves which are classified as ASME Class 1, Class 2, and Class 3 must meet the inservice test requirements set forth in the ASME OM Code and addenda that become effective subsequent to editions and addenda specified in paragraphs (f)(2) of this section. Further, subsection (4)(ii) requires, "inservice tests to verify operational readiness of pumps and valves, whose function is required for safety, conducted during successive 120-month intervals must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section 12 months before the start of the 120-month interval." The ASME OM Code Code of record for Farley, is the 2001 Edition with Addenda through OMb-2003. Subsection ISTC-5221, "Check Valves," section a(1) states in part "Check valves that have a safety function in both the open and close directions shall be exercised by initiating flow and observing that the obturator has traveled to either the full open position or to the position required to perform its intended function, and verify that on cessation or reversal of flow, the obturator has traveled to the seat."

Contrary to the above, as of May 5, 2014, the licensee failed to assure that during reversal of flow testing, the check valves' obturator traveled to its seat and fully closed. Specifically, the reverse flow criteria for FNP-1/2-STP-4.1,4.2,4.3 and FNP-1/2-STP-4.10 would have allowed significant reverse flow that could have challenged HHSI system, allowed reverse flow to the RWST, and would not have ensured the safety-related check valve's obturator traveled to its seat. This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. The violation was entered into the licensee's CAP as CRs 816150 and 816303. (NCV 05000348, 364/2014007-05, Inadequate Acceptance Criteria for Testing of Check Valves)

.6 Inadequate Characterization of IST Program Valves

Introduction: The team identified a Green NCV of 10 CFR 50.55a(f), "Inservice testing requirements," subsection (4), ASME OM code Subsection ISTC-1300, "Valve Categories," for the licensee's failure to categorize Unit 1 & 2 charging pump suction isolation valves (LCV115 B &D), and Unit 1 & 2 RWST supply to charging header check valves (Q1/2E21V026 (QV026)) as Class "A" for which seat leakage is limited to a specific maximum amount in the closed position. Specifically, the licensee's inservice testing program did not test safety-related valves to ensure they could perform their safety function in the closed direction and meet seat leakage requirements.

Description: The team reviewed the requirements for the RWST to charging pump suction check valve QV026, which is located on the suction line of the emergency core cooling system (ECCS) pumps from the RWST. The team also reviewed motor operated valves (MOVs) LCV115 B & D (charging pump suction isolation valves) which are located downstream of QV026, on parallel lines leading to the suction header of all three charging pumps (HHSI pumps). The result of this arrangement is that each path includes QV026 and either one of the two MOVs.

The NRC issued Information Notice (IN) 91-56, “Potential Radioactive Leakage To Tank Vented To Atmosphere,” alerting licensees to potential problems resulting from the leakage of isolation valves in ECCS recirculation lines to the RWST, which is vented to the atmosphere. The NRC informed licensees of corrective actions that could address the leak path vulnerabilities, by incorporating the valves identified in the leak path into the inservice testing program, and categorizing them as Category “A” valves. The NRC noted 10 CFR 50.55a, which references the ASME OM Code, and subsection ISTC-1300 “Valve Categories” stipulates that Category “A” valves are valves for which seat leakage is limited to a specific maximum amount in the closed position for fulfillment of their required functions. The team determined the licensee did not leak test QV026, or LCV115 B & D, to assure that following a Title 10 CFR Part 100 accident, during ECCS recirculation, the dose resulting from reverse leakage of containment sump water to the RWST is maintained within site boundary dose limits. The team reviewed calculation SM-03-0018-004, “Offsite and Control Room LOCA Doses,” Rev. 1, for compliance with Title 10 CFR Part 100. The calculation’s computer model failed to recognize the path specified above (sump water to RWST through either LCV115B or D and QV026) as a release path despite the operating experience discussed in IN 91-56.

The licensee entered the issues into their CAP as CRs 823022 and 815699. The team determined boric acid pump discharge pressure data available during testing of QV026 indicated the valve caused no reduction in pump discharge pressure when introducing test flow to the valve in the reverse flow direction, thereby indicating the valve held against system pressure and would not allow a significant reverse flow rate. All of the motor operated valve traces indicate the LCV 115 B & D seat. Therefore, no immediate operability concern exists.

Analysis: The team determined that failure of the licensee to properly categorize LCV115 B & D and QV026 in their inservice testing program to ensure they could perform their safety function was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of Design Control and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to properly categorize valves as Category “A” resulting in failure to leak test the valves to ensure reverse flow path from containment sump to RWST did not exceed the plant’s dose rate limits. The team used IMC 0609, Att. 4, “Initial Characterization of Findings,” issued June 19, 2012, for Mitigating Systems cornerstone, and IMC 0609, App. A, “The Significance Determination Process (SDP) for Findings At-Power,” issued June 19, 2012, and the team determined the finding was of very low safety significance (Green) because it was not a design deficiency resulting in the loss of functionality or operability. Specifically, pump data during Q1/2E21V026 reverse flow testing indicated the valve held against system pressure and would not allow a significant reverse flow to the RWST during containment sump recirculation. In addition, all of the motor operated valve traces for the LCV 115 B & D indicate the valves seat when closed. Therefore, no operability concerns exist regarding the ability of Q1/2E21V026 to prevent backflow from the containment sump to the RWST. The team did not identify a cross-cutting aspect with this finding because it is not indicative of present licensee performance. The incorrect valve scoping occurred more than three years ago.

Enforcement: Title 10 CFR 50.55a(f), "Inservice testing requirements," subsection (4) requires in part, that pumps and valves which are classified as ASME Class 1, Class 2, and Class 3 must meet the inservice test requirements set forth in the ASME OM Code and addenda that become effective subsequent to editions and addenda specified in paragraphs (f)(2) of this section. Furthermore, subsection (4)(ii) requires, "inservice tests to verify operational readiness of pumps and valves, whose function is required for safety, conducted during successive 120-month intervals must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section 12 months before the start of the 120-month interval." The ASME Code of record for Farley for Operation and Maintenance of Nuclear Power Plants (OM) is the 2001 Edition with Addenda through OMb-2003. Subsection ISTC-1300, "Valve Categories," requires in part, that valves within this subsection shall be placed in one or more of the following categories. Category "A" is for valves for which the seat leakage is limited to a specific maximum amount in the closed position for fulfillment of their required function(s), as specified in ISTA-1100.

Contrary to the above, since 1991, the licensee did not categorize MOVs LCV115 B & D, or check valve QV026 as Category "A" valves to ensure the ASME OM test requirements were met by leak testing the valves to ensure the dose remains within site boundary limits during a Title 10 CFR Part 100 accident. This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. The violation was entered into the licensee's CAP as CRs 823022 and 815699. (NCV 05000348, 364/2014007-06, Inadequate Characterization of IST Program Valves)

.7 (Opened) Unresolved Item (URI) 05000348, 364/2014007-07, Potential Effects of an RCP Thermal Barrier Break to the CCW System Were Not Evaluated

Introduction: The team identified an unresolved item (URI) regarding the lack of evaluations to ensure the CCW system was capable of withstanding the forces resulting from a reactor coolant pump (RCP) thermal barrier break hydraulic transient. Specifically, this event can lead to a thermal-hydraulic transient and its effects may not have been evaluated.

Description: The team reviewed section 9.2.2 of the UFSAR, "Cooling System for Reactor Auxiliaries," which states "High pressure switches on the return line from each reactor coolant pump thermal barrier cooling coil and a high flow switch on the common return from all three pumps will initiate the rapid closure of isolation valves to isolate the reactor coolant pumps in the event of a leak in the thermal barrier." Calculation 36.7, "Setpoint Basis for CCW Thermal Barrier Isolation Pressure Switches PSH-3184A,B,C," concluded that the pressure drop in the CCW line is sufficient to prevent any pressurization to the point of over stressing the lower design pressure piping and that the pipe schedule is sufficient to protect against erosion in the pipe due to high flow. The calculation assumed the closure time of these isolation valves (i.e., HV-3184 and HV-3046) was 10 seconds. Section 13.6.2 of the Technical Requirements Manual, "Containment Isolation Valves," included a requirement to verify these isolation valves close within 10 seconds. Recent test results showed these valves close in about 2.5 to 5 seconds. In addition, Calculation 36.7 determined that the maximum flow from a thermal barrier rupture to the 2.5 inch CCW piping would be 275 gpm of RCS water at

2235 psig and 550 degrees F. Normal service conditions at the CCW piping would be about 40 psig.

However, the team noted these conditions had the potential to result in significant steam void formation at the CCW piping. Specifically, a simplified thermodynamic analysis performed by the inspection team, and assumed initial and final saturation conditions of 550 and 200 degrees F, respectively, determined that approximately 36% of the leaked water mass would evaporate. However, the resulting steam volume would be about 547 times greater than the initial water volume at 550 degrees F. The team noted this condition and its potential effects were not evaluated. The transient could be caused by a steam void collapse upstream of the valve following the rapid valve closure of HV-3184 and downstream of the valve due to potential condensation-induced water hammer when the steam void collapses while combining with colder CCW water at a larger process line.

Section 9.2.2 of the UFSAR states “System components are designed to the codes given in Table 9.2-10.” Table 9.2-10 identified ASME Section III as the applicable code for the CCW piping. The licensee indicated the specific code of record is the 1971 Edition through the summer 1971 Addenda. Impact loading for class 3 piping, such as CCW piping, is addressed in Section ND-3600, which referred to the requirements of NB-3600. Section NB-3622.1, “Impact,” stated “Impact forces caused by either external or internal conditions shall be considered in the piping design.”

Based on this information, the team determined additional information is required to assess the potential for a thermal-hydraulic transient and its effects to CCW. The licensee captured the team concerns in their CAP as CR 834119. This issue is unresolved pending further licensee analysis to resolve the issue and to determine if a performance deficiency exists (URI 05000348, 364/2014007-07, Potential Effects of an RCP Thermal Barrier Break to the CCW System Were Not Evaluated).

.3 Operating Experience

a. Inspection Scope

The team reviewed five operating experience issues for applicability at Joseph M. Farley Nuclear Plant, Units 1 and 2. The team performed an independent review of these issues and, where applicable, assessed the licensee’s evaluation and dispositioning of each item. The issues that received a detailed review by the team included:

- IN 2013-18 - Refueling Water Storage Tank Degradation
- IN 2013-12 - Improperly Sloped Instrument Sensing Lines
- IN 2012-14 - Motor-Operated Valve Inoperable Due To Stem-Disc Separation
- Regulatory Issue Summary (RIS) 13-09 - NRC Endorsement of Nuclear Energy Institute (NEI) 09-10, Revision 1a-A, Guidelines for Effective Prevention and Management of System Gas Accumulation
- IN 2012-03 - Design Vulnerability In Electric Power System

b. Findings

.1 Failure to Update the UFSAR with the Safety Analysis Performed in Response to GL 2008-01

Introduction: The team identified a Severity Level (SL) IV NCV of 10 CFR 50.71, "Maintenance of Records, Making of Reports," for the failure to update the UFSAR. Specifically, the UFSAR was not updated to reflect the analysis requested by the NRC in GL 2008-01, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems."

Description: On January 11, 2008, the NRC requested each addressee of GL 2008-01 to evaluate its ECCS, decay heat removal (DHR), and containment spray (CS) systems licensing basis, design basis, testing, and corrective actions to ensure that gas accumulation was maintained less than the amount that would challenge the operability of these systems, and take appropriate actions when conditions adverse to quality were identified. The licensee performed analyses that resulted in the development of void acceptance criteria, identification of gas susceptible locations in piping, and development of periodic gas monitoring procedures for these newly identified locations. However, on May 13, 2014, the team noted the licensee had not updated the UFSAR to reflect the analyses.

Regulatory Guide 1.181, "Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)," stated that Revision 1 of NEI 98-03, "Guidance for Updating Final Safety Analysis Reports," provided methods that were acceptable to the NRC staff for complying with the provisions of 10 CFR 50.71(e). NEI 98-03 defined safety analyses, in part, as those performed pursuant to Commission requirement to demonstrate the capability to shut down the reactor and maintain it in a safe shutdown condition, or the capability to prevent or mitigate the consequences of accidents. In addition, Section 6.1.3 of NEI 98-03 stated, in part, that the effects of analyses and evaluations performed in response to NRC GLs must be reflected in UFSAR updates if, on the basis of the results of the requested analyses or evaluations, the existing design bases or UFSAR description are either not accurate or not bounding or both.

The team, in consultation with members of the Office of Nuclear Regulations (NRR), determined that the UFSAR description at the time of the inspection was not bounding with respect to the subject of gas accumulation management for the GL 2008-01 scoped systems. Specifically, the UFSAR did not contain a description associated with gas accumulation management in the GL 2008-01 scoped systems. As such, the allowance of gas in GL 2008-01 scoped systems is not considered part of the design bases as described in the UFSAR. However, the licensee, in response to GL 2008-01, developed analyses that established allowable gas sizes for all sections of piping that did not challenge operability of these systems. This information is used to create acceptance criteria for procedures used for monitoring a number of gas susceptible locations. This represented a change from the UFSAR de facto allowance for gas accumulation (i.e., no gas allowed).

The licensee entered this issue into their CAP as CR 823270. The corrective action that was considered at the time of this inspection was to update the UFSAR to incorporate the analyses performed during the GL 2008-01 reviews.

Analysis: The team determined the failure to update the UFSAR with the analyses performed for GL 2008-01 was a performance deficiency. Failures to update the UFSAR are dispositioned using the traditional enforcement process instead of the SDP in accordance with IMC 0612, Appendix B, Block TE2, because they potentially impede or impact the regulatory process. Specifically, failure to update the UFSAR challenges the regulatory process because it serves as a reference document used, in part, for recurring safety analyses, evaluating license amendment requests, and in preparation for and conduct of inspection activities. As a result, the team compared the performance deficiency against the examples in Section 6.1 of the NRC Enforcement Policy and determined it constituted a more than minor traditional enforcement violation because it rose to a SL-IV violation. Specifically, SL-IV violation example d.3 stated "A licensee fails to update the UFSAR as required by 10 CFR 50.71(e) but the lack of up-to-date information has not resulted in any unacceptable change to the facility or procedures." The team determined an evaluation for cross-cutting aspect was not applicable because this is a traditional enforcement violation.

Enforcement: Title 10 CFR 50.71(e) requires, in part, that each person licensed to operate a nuclear power reactor shall update periodically the UFSAR originally submitted as a part of the application for the license to assure that the information included in the report contains the latest information developed. It also states that this submittal shall include the effects of all analyses of new safety issues performed by or on behalf of the licensee at Commission request.

Contrary to the above, as of May 5, 2014, the licensee failed to update the UFSAR to assure that the information included in the report contains the effects of all analyses of new safety issues performed by or on behalf of the licensee at Commission request. Specifically, the licensee had not updated the UFSAR to reflect the analyses performed at the Commission's request contained in GL 2008-01.

The licensee is still evaluating its planned corrective actions. However, the team determined that the continued non-compliance does not present an immediate safety concern because the lack of up-to-date information had not resulted in any known unacceptable change to the facility. Because this was a SL-IV violation and was entered into the licensee's CAP CR 823270, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. (NCV 05000348, 364/2014007-08, Failure to Update the UFSAR with the Safety Analysis Performed in Response to GL 2008-01)

.2 Failure to Ensure that the RHR System Would Be Capable to Mitigate a MODE 4 LOCA

Introduction: The team identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to ensure that residual heat removal (RHR) would be capable to respond to a MODE 4 loss of coolant accident (LOCA).

Description: The team reviewed section 3.5.3 of TS, "ECCS - Shutdown," which was applicable in MODE 4 and states "One ECCS train shall be operable." The associated TS Bases states, "The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the RWST can be injected into the RCS following the accidents described in Bases 3.5.2." Section 3.5.2 of TS Bases described a large break LOCA as coolant leakage greater than the capability of the normal charging system as one of the applicable accidents. TS section 3.5.3 was modified by a note that allowed a

RHR train to be considered operable during alignment and operation for DHR mode, if it was capable of being manually realigned to the ECCS mode of operation. The DHR mode of operation is when RHR suction is aligned to the hot leg of the RCS system, and cooled through the RHR heat exchangers and returned to the system via the RCS cold leg. The ECCS mode of operation is when RHR suction is aligned to the RWST, and injected into the system via the RCS cold leg.

The team reviewed procedure FNP-1/2-SOP-7.0, "Residual Heat Removal System," which allowed operation of both trains of RHR in its DHR mode of operation up to an RCS indicated temperature of 225 degrees F. Due to instrument uncertainty, this temperature could be as high as 232 degrees F. The basis for this temperature limit was provided by evaluation C101206101, "RHR Evaluation for NSAL [Nuclear Safety Advisory Letter] 09-08." The NSAL discussed the potential for water flashing to steam in the RHR piping upon switching from DHR to ECCS mode of operation. High temperature water in the RHR system has the potential to flash to steam during a LOCA scenario while in MODE 4. Specifically, the RHR system operating in its DHR mode of operation would be at RCS temperature and pressure. Following a LOCA, the trapped fluid in the RHR lines would flash because it would suddenly be exposed to lower pressures resulting from swapping the suction of RHR over to the RWST following system realignment to its ECCS mode of operation. Evaluation C101206101 determined a maximum limit value of 232 degrees F would preclude adverse void formation by crediting the static head of the RWST.

The team noted, however, that both RHR systems would experience steam void formation at temperatures below 232 degrees F if a large break LOCA occurs that quickly depressurizes the RCS before the system suction was aligned to the RWST. Specifically, during a MODE 4 large break LOCA, the RCS would depressurize to containment pressure. This results in the flash evaporation of water inside the RHR system because its temperature will be above the saturation temperature of water at containment pressure. Although the energy of the water volume is not enough to evaporate the entire volume, flashing would occur at all locations where saturation conditions are not met and the resulting steam volume would be significantly greater than the water volume prior to the flash evaporation. The team performed a simplified thermodynamic analysis that assumed initial and final saturation conditions of 232 and 212 degrees F, respectively, and determined that approximately 2% of the water mass could evaporate. However, the resulting steam volume would be about 33 times greater than the initial 2% water volume at 232 degrees F. The team questioned if the RHR system could be significantly voided before its suction is swapped over to the RWST, and potentially vapor lock the RHR pump or create a thermal-hydraulic transient at the discharge of the pump.

The licensee entered this issue into their CAP as CR 826059. As an immediate corrective action, the licensee performed an extent of condition to identify other deficient procedures. In addition, the licensee performed a historic review and determined that in the last two years Unit 1 was operated with both trains of DHR above 212 degrees F for an accumulative time of 4.51 minutes. Unit 2 did not operate with both trains above this temperature during the last two years. The corrective action to restore compliance was to revise evaluation C101206101 and the affected procedures to protect one train of RHR for ECCS operation above 212 degrees F. In addition, the licensee implemented action tracking items in the control room to limit one train of DHR operation while above 212 degrees F.

Analysis: The team determined that the failure to ensure that RHR would be capable to respond to a LOCA that initiates in MODE 4 as required by TS 3.5.3., “ECCS - Shutdown,” was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Mitigating System cornerstone attribute of Equipment Performance and adversely affected the cornerstone objective of ensuring the capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the current procedures and design of RHR did not ensure the capability of its ECCS mode of operation to perform its mitigating function while in MODE 4 because the potential exists to allow significant steam void formation and this condition was not evaluated.

The team used IMC 0609, Att. 4, “Initial Characterization of Findings,” issued June 19, 2012, for the Mitigating Systems cornerstone and subsequently transitioned to IMC 0609 Appendix G “Shutdown Operations Significance Determination Process,” issued May 9, 2014, as the finding represented a degraded condition which could occur only during shutdown conditions. NRC IMC 0609 Appendix G Attachment 1, “Shutdown Operations Significance Determination Process Phase 1 Initial Screening and Characterization of Findings,” issued May 9, 2014, screening determined that the finding represented a potential loss of system safety function and required a phase 2 shutdown risk assessment. A bounding phase 2 shutdown risk assessment was performed by a regional senior reactor analyst in accordance with NRC IMC 0609 Attachment 2. The major assumptions in the analysis included an exposure interval of 5 minutes for Unit 1 only and a bounding conditional core damage probability of 1.0 given a LOCA. The risk was mitigated by the short exposure period and the low probability of a LOCA during shutdown conditions. The result of the analysis was an increase in core damage frequency of < 1E-6/year, a Green finding of very low safety significance.

The team did not identify a cross-cutting aspect associated with this finding because this performance deficiency is not indicative of present licensee performance. Specifically, the licensee’s evaluation regarding steam voids in MODE 4 was completed more than three years ago.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” requires, in part, design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews.

Contrary to the above, as of June 12, 2014, the licensee had not verified the adequacy of the RHR system design. Specifically, the licensee did not verify the RHR ECCS mode of operation would remain operable in MODE 4 as required by TS 3.5.3. The procedures and design would not prevent steam void formation during a MODE 4 LOCA and this condition was not evaluated.

The licensee is still evaluating its planned corrective actions; however, the team determined that the continued non-compliance does not present an immediate safety concern because TS 3.5.3 was not applicable for the conditions during the inspection and the licensee established an interim action to ensure one RHR train is not operated in its DHR mode above 212 degrees F. This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. The violation was entered into the licensee’s CAP as CR 826059. (NCV 05000348, 364/2014007-09, Failure to Ensure that the RHR System Would Be Capable to Mitigate a MODE 4 LOCA)

.3 <u>(Opened) URI 05000348, 364/2014007-10, Potential Misapplication of Void Transport Empirical Correlations</u>

Introduction: The team identified a URI regarding the potential misapplication of void transport empirical correlations. Specifically, the licensee developed suction void acceptance criteria using questionable empirical correlations.

Description: On January 11, 2008, the NRC requested each addressee of GL 2008-01 to evaluate its ECCS, DHR, and CS systems licensing basis, design, testing, and corrective actions to ensure that gas accumulation is maintained less than the amount that would challenge the operability of these systems, and take appropriate actions when conditions adverse to quality were identified. Part of the licensee's actions to address these requests was to develop acceptance criteria for void volumes that are identified during periodic monitoring.

The team noted the acceptance criteria methodology used for assessing suction voids found at suction piping potentially misapplied empirical correlations for void transport. Specifically, these empirical correlations were developed from a test known as "Purdue test" and were a set of correlations that preceded the ones reviewed by NEI 09-10, "Guidelines for Effective Prevention and Management of System Gas Accumulation." The NRC endorsed this industry guidance in RIS 2013-09. Section 3.16 of NEI 09-10 discussed the Purdue test report and stated "Empirical correlation predictions are limited to estimating uncertainties." It also discussed test data weaknesses and states, "The NRC staff concludes that the modeling of two-phase two-component transient flow must be conducted with allowance for these data weaknesses." The associated NRC safety evaluation (SE) (i.e., ML12342A368) included the same statements. In addition, it states "The NRC staff finds the methodologies and correlations described in the references have not always been sufficiently compared to experimental data to establish that they are acceptable for determining operability under all conditions." It also states "Licensee use of the references must address the weaknesses that are addressed in this SE." Based on this information, the team questioned the technical adequacy of the empirical correlations used by the licensee when developing suction void acceptance criteria. In addition, since these empirical correlations were the predecessors of the correlations reviewed by NEI and NRC, the team questioned if additional weaknesses existed. The licensee captured the team questions in their CAP as CR 834115. The team, in consultation with NRR staff, determined this issue is unresolved pending further NRR review of the acceptability of the licensee's methodology for determining suction void acceptance criteria and determination of further NRC actions to resolve the issue to determine if a performance deficiency exists. (URI 05000348, 364/2014007-10, Potential Misapplication of Void Transport Empirical Correlations)

4OA6 <u>Meetings, Including Exit</u>

On July 13, 2014, the team exited with Todd Youngblood and members of the licensee's staff. In addition, on July 23, 2014, the team re-exited via telephone with members of the licensee's staff. The inspectors verified that no proprietary information was documented in this report.

ATTACHMENT: SUPPLEMENTARY INFORMATION

SUPPLEMENTARY INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:
J. Andrews, Maintenance Director
B. Arens, Licensing Supervisor
R. Bethea, Design Engineering
B. Cates, Site Design Engineering
J. Collier, Licensing Engineer
H. Cooper, Engineering Programs Supervisor
P. Cooper, Accreditation Supt Manager
D. Enfinger, Corrective Action Program Supervisor
C. Gayheart, Site Vice President
A. Gray, Eng Programs Manager
B. Griner, Fleet Programs Director
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D. Hobson, Shift Operations Manager
J. Holton, Components Eng Supervisor
J. Hutto, Plant Manager
D. Lambert, Fleet Design Manager
R. Lyons, Design Engineering
L. Mansfield, Fleet Design Director
R. Martin, Regulatory Affairs Manager
D. McKinney, Licensing Manager
J. McLean, Licensing Engineer
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W. Nobles, Site Design Engineer
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D. Reed, Operations Support Manager
L. Riley, Performance Improvement
R. Smith, Site Design Manager
B. Taylor, Nuclear Oversight Supervisor
C. Thornell, Operations Director
C. Westberry, Engineering Programs Supervisor
E. Williford, Lead Engineer - Licensing
T. Youngblood, Engineering Director

NRC personnel
F. Ehrhardt, Chief, Projects Branch 3, Division of Reactor Projects
P. Niebaum, Senior Resident Inspector, Division of Reactor Projects
J. Sowa, Resident Inspector, Division of Reactor Projects
G. MacDonald, Senior Reactor Analyst, Division of Reactor Projects

LIST OF ITEMS OPENED, CLOSED, DISCUSSED, AND UPDATED

Opened and Closed

05000348, 364/2014007-01	NCV	Inadequate Operability Evaluation of the CCW Miscellaneous User Isolation Valves (Section 1R21.2b.1)
05000348, 364/2014007-02	NCV	Failure to Comply with IEEE 308-1971 for the Required Independence of 120 Volt Vital AC Distribution System Channels (Section 1R21.2b.2)
05000348, 364/2014007-03	NCV	Failure to Correct Lack of Validated Time Critical Operator Actions Analyses (Section 1R21.2b.3)
05000348, 364/2014007-04	NCV	Acceptance Criterion for UHS Temperature Did Not Consider Instrument Uncertainty (Section 1R21.2b.4)
05000348, 364/2014007-05	NCV	Inadequate Acceptance Criterion for Testing of Check Valves (Section 1R21.2b.5)
05000348, 364/2014007-06	NCV	Inadequate Characterization of IST Program Valves (Section 1R21.2b.6)
05000348, 364/2014007-08	SLIV	Failure to Update the UFSAR with the Safety Analysis Performed in Response to GL 2008-01 (Section 1R21.3b.1)
05000348, 364/2014007-09	NCV	Failure to Ensure that the RHR System Would Be Capable to Mitigate a MODE 4 LOCA (Section 1R21.3b.2)

Opened

05000348, 364/2014007-07	URI	Potential Effects of an RCP Thermal Barrier Break to the CCW System Were Not Evaluated (Section 1R21.2b.7)
05000348, 364/2014007-10	URI	Potential Misapplication of Void Transport Empirical Correlations (Section 1R21.3b.3)

LIST OF DOCUMENTS REVIEWED

Procedures

FNP-0-EMP-1335.01, Inspection, Maintenance and Testing of 7.5 kVA Inverter, Ver. 22
FNP-0-ARP-2.5, Annunciator Response Procedure Emergency Power Board Annunciator Panel Z, Ver. 35.0
FNP-0-ARP-2.4, Annunciator Response Procedure Emergency Power Board Annunciator Panel Y, Ver. 26.2
FNP-0-EMP-1313.19, Inspection and Adjustment of Cutler Hammer 4.16kV Circuit Breakers Type MA-VR350, Ver. 10.0
FNP-0-EMP-1313.20, Enhanced Inspection and Adjustment of Cutler Hammer 4.16kV Circuit Breakers Type MA-VR350, Ver. 10.0
FNP-0-EMP-7542.01, Electrical Maintenance Procedure General Electric Time Overcurrent Relays Type IAC 53B, 54A 66B, Ver. 10.0
FNP-1-AOP-4.1, Abnormal Reactor Coolant Pump Seal Leakage, Rev. 11.0
FNP-1-AOP-28.1, Fire or Inadvertent Fire Protection System Actuation in the Cable Spreading Room, Ver. 34.1
FNP-1-ARP-1.4, Main Control Board Annunciator Panel D, Ver. 54.1
FNP-2-AOP-9.0, Loss of Component Cooling Water, Ver. 24.0
FNP-2-EEP-0, Reactor Trip or Safety Injection, Rev. 42.0
FNP-2-EEP-1, Loss of Reactor or Secondary Coolant, Rev. 28
FNP-2-ESP-1.2, Post LOCA Cooldown and Depressurization, Rev. 23
FNP-2-ESP-1.3, Transfer to Cold Leg Recirculation, Rev. 22
FNP-2-SOP-54.0, Spent Fuel Pit Cooling and Purification System, Ver. 67.1
FNP-2-SOP-54.4, RWST Silica Removal by the Boric Acid Recovery System, Ver. 24.0
NMP-OS-014, Time Critical Operator Action Program, Ver. 2
NMP-OS-014-001, FNP Time Critical Operator Action Program, Ver. 2
FNP-2-AOP-9.0, Loss of Component Cooling Water, Rev. 24
FNP-2-SOP-23.0, Component Cooling Water System, Rev.91.4
FNP-2-STP-1.0, Operations Daily and Shift Surveillance Requirements, Rev. 98.0
FNP-2-ARP-1.1, Annunciator Response Procedure, Rev. 35.3
FNP-0-MP-94.0, Inspection of CCW Pumps, Rev. 16.0
FNP-1-STP-23.8, CCW Valve IST, Rev. 50.0
FNP-0-ETP-4574.0, Gas Accumulation Monitoring and Trending, Rev. 11.0
NMP-ES-024-515, UT Examination Procedure for Liquid Level Measurement, Rev. 3.1
FNP-0M-82, Service Water Plan, Rev. 12.0
FNP-1-SOP-2.1, CVCS Plant Startup and Operation, Rev. 134.0
FNP-1-STP-23.1, 1A CCW Pump Quarterly IST, Rev. 38.3
FNP-1-ARP-1.4, Main Control Board Annunciator Panel D, Rev. 54.1
FNP-1-SOP-7.0, RHR System, Rev. 103.0
FNP-0-EMP-2300.07, Thermometer Calibration Main Power, Startup, And Unit Auxiliary Transformers
FNP-1-STP-40.0A, Safety Injection With Loss Of Off-Site Power Test – A Train, Ver. 2
FNP-0-EMP-2300.12, Start-Up Transformer Insulation Power Factor And Ancillary Testing, Ver. 7
FNP-0-EMP-2570.05, Transformer And Tower Insulator Exterior Preservation And Inspection, Ver. 7
FNP-0-EMP-2570.06, Transformer Automatic Gas Seal Equipment Test, Ver. 6
FNP-0-EMP-2570.09, Westinghouse Sudden Pressure Relay Test, Ver. 4
FNP-0-EMP-2570.12, Control And Alarm Testing Units 1 And 2 Start-Up Transformers, Ver. 8

FNP-1-STP-205.1, Refueling Water Storage Tank Level Q1F16LT0501 Loop Calibration, Ver. 18
FNP-1-STP-205.2, Refueling Water Storage Tank Level Q1F16LT0502 Loop Calibration, Ver. 19
FNP-2-STP-205.1, Refueling Water Storage Tank Level Q2F16LT0501 Loop Calibration, Ver. 22
FNP-2-STP-205.2, Refueling Water Storage Tank Level Q2F16LT0502 Loop Calibration, Ver. 20
FNP-1-IMP-221.1, Refueling Water Storage Tank Eccs Valve Low Level Switches Loop Calibration And Operational Check, Ver. 4
FNP-2-IMP-221.1, Refueling Water Storage Tank ECCS Valve Low Level Switches Loop Calibration And Operational Test, Ver.5
FNP-0-EMP-2541.03, HFA65D Relay Inspection, Ver. 3
FNP-0-EMP-2541.01, GE Time Overcurrent Relays Type 1A, Ver. 12
FNP-0-EMP-2541.02, GE Differential Lockout Relay Functional Test, Ver. 9
FNP-1-EMP-2541.03, SUT 1A Differential Lookout Relay Functional Test, Ver. 6
FNP-1-EMP-2541.04, SUT 1B Differential Lookout Relay Functional Test, Ver. 4
FNP-2-AOP-16.0, Abnormal Operating Procedure, CVCS Malfunction, Rev. 20.0
FNP-2-SOP-54.4, RWST Silica Removal By The Boric Acid Recovery System, Rev. 24
FNP-2-SOP-2.1, Chemical and Volume Control System Plant Startup and Operation, Rev. 131.0
FNP-0-M-115, Check Valve Condition Monitoring Plan, Rev. 7
NMP-ES-017-004, MOV Diagnostic Procedure for Gate and Globe Valves, Rev. 3.0
FNP-0SOP-0.6, System Operating Procedure, Limitorque MOV Lubrication,
FNP-1-STP-4.10, Reverse Flow Test of RWST to Charging Pump Check Valve and CVCS Emergency Borate Filter to Charging Pump Suction Check Valve,
FNP-1-STP-4.10 Reverse Flow Test of RWST to Charging Pump Check Valve and CVCS Emergency Borate Filter to Charging Pump Suction Check Valve,
FNP-1-STP-4.1, 1A Charging Pump Quarterly Inservice Test, Rev. 68.0,
FNP-1-STP-4.2, 1B Charging Pump Quarterly Inservice Test, Rev. 69.0,
FNP-1-STP-4.3, 1C Charging Pump Quarterly Inservice Test, Rev. 63.0,
FNP-0-GMP-27.2, Disassembly, Inspection, Repair and Reassembly of Safety Related and Non safety Related Check Valves, Rev. 20.0,
NMP-ES-017, Motor Operated Valve Program, Version 8.0,
NMP-ES-017-01, MOV Regulatory Scoping Process, Version 5.0,
NMP-ES-017-01-F-V1, Version 1.0, FNP MOV Program Scope
NMP-ES-017-02, MOV Design Basis Setpoint Determination, Version 5.0,
NMP-ES-017-03, MOV Trending and Margin Management, Version 3.0,
NMP-ES-017-04, MOV Diagnostic Procedure for Gate, Globe Valves, Version 7.0,
NMP-ES-017-08, MOV Mechanical and Electrical Inspections Version 8.1,
NMP-ES-017-021, MOV Diagnostic Testing Votes Infinity Version 1.0,
NMP-ES-017-GL01, MOV Problem Solving and Troubleshoot Version 1.1,
FNP-0-PMP-303, Guidelines for Site-Routing and Supporting of Tubing Version 8,
FNP-0-PMP-303, Guidelines for Site-Routing and Supporting of Tubing Rev. 0,
NMP-ES-017-001-F, Farley Nuclear Plant MOV Regulatory Scope Version 1,
NMP-ES-017-04, MOV Diagnostic Procedure for Gate, Globe Valves, Version 6.0,
NMP-ES-017-04, MOV Diagnostic Procedure for Gate, Globe Valves, Version 7.0,
FNP-0-EMP-1501.17, Rising Stem Test Field Data Sheet,
NMP-ES-044, Preparation of Design Change Packages, Version 13.0,

Drawings
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D-203096, Unit No. 2 Loads Diagram (Emergency), Ver. 10.0
D-207024, Single Line 120 VAC Vital & Regulated System Train A, Ver. 26.0
D-207025, Single Line 120 VAC Vital & Regulated System Train B, Ver. 26.0
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D-207180, Elementary Diagram Charging/High Head Safety Injection Pump 2A, Ver. 12.0
D-172674, Elementary Diagram – Service Water to Turbine Building Isolation MOV's, Ver. 10.0
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D-177143, Elementary Diagram – 4160V Bus 1F Incoming Breaker From Diesel Gen. 1-2A, Ver. 24
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D-177659, Elementary Diagram Loading Sequencer B1H Load Shedding Circuit, Ver. 21
D-177660, Elementary Diagram Loading Sequencer B1J Bus 1J Load Shedding Circuit, Ver. 16
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D-207646, Elementary Diagram Loading Sequencer B2G ESS Sequencer, Ver. 17
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D-207650, Elementary Diagram Loading Sequencer B2G LOSP Sequencer Bus 2G, Ver. 10
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D-207654, Elementary Diagram Loading Sequencer B2G Load Shedding Circuit, Ver. 13
D-207659, Elementary Diagram Loading Sequencer B2H Load Shedding Circuit, Ver. 21

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D-175039, P&ID Chemical and Volume Control System, Sheet 7, Rev. 9.0
D-175039, P&ID Chemical and Volume Control System, Sheet 2, Rev. 30.0
D-175039, P&ID Chemical and Volume Control System, Sheet 6, Rev. 10.0
D-175039, P&ID Chemical and Volume Control System, Sheet 1, Rev. 25.0
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D175038, P&ID Safety Injection System, Sheet 2, Rev. 23.0
D175038, P&ID Safety Injection System, Sheet 3, Rev. 27.0
D-205291, CVCS Boric Acid Tank, Sheet 1, Rev.1.0
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D-177631, Sheet 1, Elementary Diagrams – Refueling Water Storage Tank to Charging Pump 575V Motor Operated Valve, Rev. 14.0
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D-207631, Sheet 1, Elementary Diagrams – Refueling Water Storage Tank to Charging Pump 575V Motor Operated Valve, Rev. 10.0
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D-205002, Sheet 2, P&ID – Component Cooling Water System, Rev. 21.0
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E-095, Auxiliary Building Battery Capacity and Voltage Evaluation, Ver. 12.0
SE-91-1976-1, Motor Starter Control Circuit Study, Ver. 6
SE-94-0470-004, Unit 1 Load Study Summary, Ver. 4
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SE-94-0-0378-001, Unit 1 MOV MCCB Instantaneous Trip Setting/Thermal Overload Relay Settings and Heater Sizes, Rev. 3
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SE-94-0470-001, Unit 1 As-built Load Study, Ver. 7
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SM-C063346201-003, Development of Farley U1 and 2 CCW Heat Exchanger Model with PROTO-HX, Rev. 1
SM-C080146901-001, U1 and 2 CCW Heat Exchanger PROTO-HX Computer Model, Rev. 1
CN-96-0047, CCW Evaluation–Power Uprate and Replacement Steam Generator, Rev.8
34.5, Component Cooling Water System NPSH, Rev. 1
SM-C063346201-001, U1 CCW Flow Model, Rev. 4
39.3, CCW Surge Tank Analytical Limit for Level Setpoint, Rev. 0

CN-SEE-III-08-42, Evaluation of Suction Side Gas Void Volumes for J.M. Farley Units 1 & 2 to Address GL-2008-01, Rev. 1
SM-96-9012-002, Effects of Plastocor Coating on CCW HX's Thermal Performance, Rev. 0
SM-92-1-7927-001, Turbine Building Service Water Supply Isolation Valve Throttling, Rev. 0
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REA 93-0155, CCW Check Valve Q1/2P17V288 Flow Criteria, 7/22/93
37.7, CCW Flow Balance, Rev. 1.0
SM-C101142901-001, SW System Flow Balancing Model, Rev. 1
SM-SNC82835-001, AOV Setpoints Review for Q2P17HV3096A&B, 8/22/12
SM-ES-89-1499-06, SW Flow Isolation at Turbine Building, Rev. 0
RER C101206101, RHR Evaluation for NSAL 09-08, 5/5/10
TE694116, Review of RIS 2013-09, 8/27/13
E-35, Setting of Protective Relays for FNP Units 1 & 2 Auxiliary Power Systems, Rev. 1
SE-94-0470-001, Unit 1 As- Built Load Study, Ver. 7
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SE-98-1845-2-PE, Large, Small and SBO Diesel Dynamic Study, Ver. 1
A508666, Scaling For The RWST Level Loops Q1/2f16lt501 And Q1/2f16lt502 And For The RWST Level Switches Q1/2f16ls507, Q1/2f16ls508, Q1/2f16ls515,
Q1/2f16ls516 Scaling Document, Ver. 3
SJ-98-1693-001, Calculation Establish the Setpoint for the RWST Level Transmitters
Q1/2F16LT501 and LT502, Ver. 4
SM-94-0452-001, RWST Maximum Drain Down Rate, Ver. 4
SJ-98-1693-003, Calculation Establish the Setpoint for the RWST Level Switches
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23.08, RWST Check Valves, Rev. 3
X4C1204T05, RWST and BAST, Rev. 1.0
SM-94-0452-001, RWST Volume, Rev. 5.0
BM-97-1547-001, RWST Vortex Analysis, Rev. 2.0
SJ-00-2290-005, RWST Tank Curve, Rev. 0
MC-V-07-0251, RWST Volume, Setpoints, and Operator Time Requirements for ECCS Pump Suction, Rev. 3.0
7.3, RWST Volume Required, Rev. 0
SM-90-1653-001, MOV Thrust Requirements for Gate and Globe Valves, Rev. 16
SM-90-1653-003, Design Basis Differential Pressure for the MOV Program, Rev. 15
SM-90-1653-012, MOV Nominal Stroke Time for Gate, Globe and Butterfly Valves, Rev. 13
SM-94-0452-001, RWST Maximum Draindown rate, Rev. 5.0
2.8, Chemical and Volume Control System, Rev. 2
CN-SEE-IV-08-12, Recommendation on Vogtle Units 1 and 2 and Farley Units 1 and 2 Volume
Control Tank Initial Level to Prevent Air Ingestion Due to Vortex During Tank Draindown, Rev. 0
CN-SEE-III-08-42, Evaluation of Suction Side Gas Void Volumes for JM Farley Units 1&2 to Address GL-2008-01, Rev. 3
SM-03-0018-004, Offsite and Control Room LOCA Doses, Rev. 1
SM-90-1653-002 Reduced Voltage Torque/Thrust Capability for Gate & Glove Valves in the FNP MOV Program, Ver. 23,
ER- 5.0, Equipment Inaccuracy Summary for Motor Operated Valves, Rev 26,
SE-94-0470-001, Unit 1 As-Built Load Study, Ver. 7.0,
SE-94-0470-007, Unit 2 As-Built Load Study, Ver. 6.0,
03.14, Outdoor Storage Tanks Yearly Dose Rate Rev 1,
SE-94-0-0378-001, MOV Combination Starter Component Sizes and Settings, Rev 3,

SE-94-0-0378-002 MOV Combination Starter Component Sizes and Settings, Rev 3,
MC-F-13-0004, Auxiliary Building Capacity and Voltage Evaluation Version 1,
39.3, CCW Surge Tank Analytical Limit for Level Setpoint Rev. 0,
SE-94-0470-004, Unit 1 Load Study Summary, Version 4,
SE-94-0470-005, Unit 2 Load Study Summary, Version 5,
SM-90-1653-012, MOV Nominal Stroke Times for Gate, Globe, and Butterfly Valves, Version 13,

<u>CRs/TEs/CARs</u>

CR458319	CR505403	CR603276	CR604937	CR625699	CR454020
CR513694	CR580831	CR581247	CR659064	CR739061	CR789899
CR517201	CR677033	CR544619	CR774303	CR557863	CR390674
CR515413	CR573144	CR573146	CR576106	CR580465	CR581169
CR687866	CR071357	CR627056	CR626824	CR498687	CR2004100022
CR058612	CR393860	CR400342	CR401257	CR436172	CR438288
CR496336	CR505906	CR563396	CR613013	CR622844	CR713134
CR717968	CR724453	CR733284	CR521579	CR546397	CR671914
CR723815	CR733032	CR745197	CR618306	CR420910	CR2009112150
CR347322	CR746021	CR759442	CR815709	CR675720	CR2008101055
CR811703	CR2009112150		TE676528	CAR 207147	

<u>Corrective Action Documents Written Due to this Inspection</u>

CR850528, CDBI Question 362 on Vital Inverter Non-Safety Related Loads
CR817096, NRC questioned validity of previous Operability Determination
CR816316, Omission of fuse and relay contact resistances in base calc SE-91-1976-1
CR815709, CDBI Related-Loss of Normal Feedwater Operator Actions
CR821018, FSAR documentation deficiency
CR823056, Operability Determination Inadequacies
CR823270, FSAR Did Not Incorporate GL2008-01 Analyses
CR819467, One Channel of UHS Temperature Instrument Exceeded Allowable Value,
CR810638, Uncertainty Analysis Assume Incorrect Calibration Frequency
CR817148, NRC Question No. 164 noted issues with Calc 34.05, “CCW NPSH”, during the 2014 Farley
CR810018, U1 CCW Surge Tank Hold Down Bolt Nuts Less Than Fully Engaged
CR810199, U2 CCW Surge Tank Hold Down Bolt Nuts Less Than Fully Engaged
CR826059, Potential Steam Formation in RHR
CR 820553, Unit 2 RWST Level Switches Operability Due to Calibration Data
CR810995, E21LCV115B/D Stroke Time Impact from EDG Frequency and Voltage
CR816303, Charging Pump Mini Flow Check Valves
CR820350, Evaluate CDBI Q12 and Determine Actions Needed for VCT Outlet Isolation Valves Positive Interlock
CR823022, NRC CDBI Issue 21, Valve Leakage
CR823270, NRC CDBI Identified that FNP Did Not Update the FSAR to Incorporate Analyses Performed to Address Generic Letter 2008-01
CR815699, CDBI Identified Need for Review of E21 Valves IST Program Categories
CR816150, Q1/2E21V0026 Acceptance Criteria Needs to be Modified
CR826581, Charging Pump Data Review
CR834115, ECCS Gas Void Criteria
CR834119, RCP Thermal Barrier Breach
CR811703, Calculation editorial error found for LCV115B D
CR811708, Gap with MOV Margin Calculations

CR823401, FNP could not provide documentation of previous validations of time critical actions of several design basis accident analysis
CR821999, No corrective action taken for NRC minor violation Identified by NRC CDBI inspector

Work Orders
SNC75650, 10 year Component Replacement PM 7.5 kVA Inverter 2A, dated 11/1/11
SNC81176, 7.5 kVA Inverter 2A Inspect, Clean and Test per PM, dated 10/21/11
SNC75652, 7.5 kVA Inverter 2B 10YR Component Replacement PM, dated 10/31/11
SNC81175, 7.5 kVA Inverter 2B Inspect, Clean and Test per PM, dated 10/29/11
2091479201, Replace the Components which have a 10 Year Design Life 7.5 kVA Inverter 2C, dated 4/11/10
SNC81023, 7.5 kVA Inverter 2C Inspect, Clean and Test per PM, dated 4/12/13
SNC75653, Replace the Components which have a 10 Year Design Life, dated 4/23/13
SNC81021, 7.5 kVA Inverter 2D Inspect, Clean and Test per PM, dated 4/12/13
1061956301, SW to Turb Bldg Iso A Trn Q1P16V515, dated 10/12/07
107166301, SW to Turb Bldg Iso A Trn Q1P16V515, dated 4/25/09
1072600901, SW to Turb Bldg Iso A Trn Q1P16V515, dated 4/5/09
2090552301, SW to Turb Bldg Iso A Trn Q1P16V515, dated 4/25/10
1071663501, SW to Turb Bldg Iso B Trn Q1P16V517, dated 4/23/09
1072600801, SW to Turn Bldg Iso B Trn Q1P16V517, dated 4/23/09
2080298301, SW to Turn Bldg Iso B Trn Q2P16V517, dated 4/26/10
1090925901, Bkr DF01 O/C Relay Ph 1 Unit 1, dated 10/19/10
2071652601, Bkr DG01 O/C Relay Ph 1 Unit 2, dated 4/18/10
SNC55381, Bkr DG01 O/C Relay Ph 2 Unit 1, dated 10/8/13
2071627901, Bkr DG01 O/C Relay Ph 3 Unit 2, dated 11/16/09
1072880401, Incoming Startup Transformer 1A 52-DF01, dated 4/21/09
1072881801, Incoming Startup Transformer 1A 52-DG01, dated 4/21/09
1071331201, Overcurrent Relay Ph 1, dated 8/7/09
1071331101, Overcurrent Relay Ph 2, dated 8/7/09
1071362901, Overcurrent Relay Ph 3, dated 8/7/09
2050926401, Overcurrent Relay Ph 1, dated 3/16/07
2050926701, Overcurrent Relay Ph 2, dated 3/19/07
2050926301, Overcurrent Relay Ph 3, dated 3/21/07
SNC389440, Perform FNP-1-STP-10.3RF (Emergency Core Cooling Valves Inservice Test and PORV Block Valve Stroke Test).
SNC480560, Perform FNP-1-STP-10.3 (Emergency Core Cooling Valves Inservice Test and PORV Block Valve Stroke Test) on A-train components
SNC485592, Perform FNP-2-STP-10.3 (Emergency Core Cooling Valves Inservice Test and PORV Block Valve Stroke Test) on B-train components ***stroke MOV8131B after SNC420756 is complete***
SNC490288, Perform FNP-2-STP-10.3 (Emergency Core Cooling Valves Inservice Test and PORV Block Valve Stroke Test) on A-train components
SNC490760, Perform FNP-2-STP-10.3 (Emergency Core Cooling Valves Inservice Test and PORV Block Valve Stroke Test) on B-train components
SNC490803, Perform FNP-2-STP-10.3 (Emergency Core Cooling Valves Inservice Test and PROV Block Valve Stroke Test) on A-train components
SNC491115, Perform FNP-1-STP-10.3 (Emergency Core Cooling Valves Inservice Test and PORV Block Valve Stroke Test) on A-train components
SNC491117, Perform FNP-1-STP-10.3 (Emergency Core Cooling Valves Inservice Test and PORV Block Valve Stroke Test) on B-train components

1090767401, 1A CCW HX ECT, dated 8/20/10
1090767501, 1A CCW HX TC, dated 8/20/10
SNC463653, 1A CCW HX ECT, dated 8/21/13
SNC464689, 1B CCW HX ECT, dated 11/12/13
SNC453168, 1C CCW HX ECT, dated 9/26/13
SNC417826, 2B CCW HX ECT, dated 4/24/13
SNC416500, 2A CCW HX ECT, dated 3/26/13
SNC416974, 2C CCW HX ECT, dated 4/24/13
SNC57378, Calibrate CCW Pump Room Flood Annunciation, dated 7/9/12
SNC490723, FNP-1-STP-23.8 A TRN and Misc CCW Valve IST, dated 3/11/14
SNC448950, Blow Down Instrument Sensing and Calibrate Q1P16PDS0565, dated 1/14/14
SNC63692, Perform CCW Surge Tank LT Per FNP-1-IMP-210.6, dated 4/13/13
SNC76935, Perform FNP-2-IMP-210.7, dated 4/4/12
SNC80401, 2C CCW HX Clean/Inspect, dated 9/26/12
SNC468739, CCW HX 2B Jacket Relief Valve, dated 4/10/13
SNC353715, CCW Refueling Outage Valves IST, dated 4/30/13
SNC482385, 1A CCW Pump Quarterly, dated 11/12/13
SNC473308, CCW Surge Tank Relief Valve, dated 11/26/13
SNC63390, CCW Surge Tank Vacuum Breaker, dated 3/25/13
SNC77822, JOG MOV Butterfly Valve Mods, dated 10/18/11
1071663301, Design Basis Diagnostic Test V515, dated 4/25/09
SNC400338, Perform FNP-1-STP-45.6, dated 10/16/13
SNC424826, Perform Diagnostic testing of Q2P17HV3096A after replacement on DCP82835, dated 5/1/13
SNC433014, Perform Diagnostic testing of Q2P17HV3096B after replacement on DCP82835, dated 5/1/13
1092311601, Perform FNP-0-ETP-4574, dated 11/10/10
SNC358817, Unit 1 Q1E21MOV8107, Perform the full test (diagnostic) per NMP-ES-017-004
SNC400906, Unit 1 Q1E21MOV8107, Task 004, Lube and Inspect
SNC62878, Q1E21MOV8108 – Charging Pumps to regenerative heat exchanger perform a design basis diagnostic (full) test
1041878001, Unit 1, Perform a design basis diagnostic (full) test per FNP-0-EMP-1501.17
1070672301, Perform FNP-1-STP-4.10 (reverse flow test of RWST to charging pump check valve
2082284501, Perform FNP-1-STP-4.10 (reverse flow test of RWST to charging pump check valve
SNC353704, Unit 2 CVCS Cold Shutdown Valves Inservice Test
SNC402450, Unit 2 Perform SNP-1-STP-45.1, CVCS Cold Shutdown Valves Inservice Test
SNC62073, Unit 1 Perform FNP-1-STP-40.0, Safety Injection with Loss of Offsite Power Test
SNC453817, Unit 1 Perform LOOP Calibration per FNP-1-IMP-202.11
SNC402450, Unit 1 Perform FNP-1-STP-45.1 (CVCS Cold Shutdown Valves Inservice Test
SNC478897, Unit 1 Perform FNP-1-STP-4.1, 1A Charging Pump Quarterly Inservice Test
1090915601, VCT to Charging Pump Suction Header Check Valve
1090703801, RWST to CHG Pump, Rev 1, dated 02/12/2010
2090693101, RWST to CHG Pump, Rev 1, dated 08/05/2009
1080551001, RWST to CHG Pump, Rev 0, dated 03/19/2008
SNC62870, LCV115D Full Test Per NMP-ES-017-004, dated 04/18/2012
2090693102, RWST to CHG Pump, Rev 0, dated 04/17/2010
2090693103, RWST to CHG Pump, Rev 0, dated 04/15/2010
SNC68680, LCV115B Full Test Per NMP-ES-017-004, dated 04/21/2012

SNC540344, Q1P17HV3096A CCW from Eva PKGS & H2 Recomb Stroke time greater than Max., dated 03/10/2014
1050855401, Test and Inspect SU XFMR 1A O/C Relay Phase 1, dated 2/22/06
1050856101, Test and Inspect SU XFMR 1A O/C Relay Phase 2, dated 2/22/06
1050856201, Test and Inspect SU XFMR 1A O/C Relay Phase 3, dated 2/22/06
1052846301, Test and Inspect SU XFMR 1A Differential Relay Phase 1, dated 10/14/07
1070088201, Test and Inspect SU XFMR 1A Differential Relay Phase 2, dated 10/14/07
1072777601, Calibration of RWST Level Switch LS507 and LS508, dated 8/16/09
1072880401, Test and Inspect SU XFMR 1A Differential Relay Phase 3, dated 10/14/07
1081618501, Calibration of RWST LT501, dated 3/19/10
1082315701, BF1 Sequencer Operability Test, dated 5/19/10
1082451301, B1G Sequencer Operability Test, dated 6//23/10
1090770201, Test and Inspect SU XFMR 1A O/C Relay Phase 3, dated 10/15/10
1090770301, Test and Inspect SU XFMR 1A O/C Relay Phase 2, dated 10/15/10
1090770401, Test and Inspect SU XFMR 1A O/C Relay Phase 1, dated 10/15/10
1092193701, Calibration of RWST Level Switch LS515 and LS516, dated 3/4/11
2041473501, Calibration of RWST Level Switch LS515 and LS516, dated 11/7/05
2061392701, Calibration of RWST Level Switch LS515 and LS516, dated 5/1/07
2071344801, Calibration of RWST Level Switch LS515 and LS516, dated 3/14/09
2072886401, Perform Maintenance and Inspection of Breaker 2A DF15, dated 11/18/08
2072887101, Perform Maintenance and Inspection of Breaker DG07, dated 10/15/11
2072887201, Perform Maintenance and Inspection of Breaker DG08, dated 9/15/08
2082323601, Calibration of RWST LT501, dated 5/21/10
2090200501, Calibration of RWST Level Switch LS507 and LS508, dated 3/19/09
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SNC346831, Calibration of RWST LT501, dated 7/31/13
SNC360313, FNP-1STP-226.1 Sequencer Operability Test, dated 7/1/13
SNC363627, FNP-1-STP-934.1, Degraded Voltage Test, dated 5/14/13
SNC368273, FNP-1-STP-934.1, Degraded Voltage Test, dated 6/25/13
SNC375918, Calibration of RWST LT502, dated 5/30/13
SNC376341, Loading Sequencer Agastat Relay 2-15F Replacement, dated 4/19/12
SNC376344, Loading Sequencer Agastat Relay 2-12F Replacement, dated 4/19/12
SNC376346, Loading Sequencer Agastat Relay 2F Replacement, dated 4/19/12
SNC376347, Loading Sequencer Agastat Relay YFX Replacement, dated 4/19/12
SNC376353, Loading Sequencer Agastat Relay 2-13F Replacement, dated 4/19/12
SNC376355, Loading Sequencer Agastat Relay 2-1F Replacement, dated 4/19/12
SNC381107, FNP-1STP-226.2 Sequencer Operability Test, dated 8/15/13
SNC381998, FNP-1-STP-933.1, Degraded Voltage Test, dated 7/29/13
SNC381999, FNP-1-STP-933.1, Degraded Voltage Test, dated 9/3/13
SNC401406, 1B SU XFMR Gas Seal Equipment Test, dated 10/13/13
SNC401407, 1B SU XFMR Pressure Relay Test, dated 10/7/13
SNC401413, 1B SU XFMR Control and Alarm Test, dated 10/3/13
SNC401419, 1B SU XFMR Exterior Preservation Inspection, dated 10/7/13
SNC410131, Perform Maintenance and Inspection of Breaker DG08, dated 3/27/14
SNC410290, Calibration of RWST Level Switch LS507 and LS508, dated 6/5/13
SNC432659, Perform Ultrasonic1B SU XFMR, dated 7/23/13
SNC432661, Perform Ultrasonic1A SU XFMR, dated 3/20/13
SNC443376, FNP-0-STP-80.6 DG 1-2A 24 Hour Load Test, dated 12/20/13
SNC451810, RWST Lo-Lo LVL To ECCS Switchover CH3, dated 12/11/12

SNC490739, FNP-1-STP-80.16 (Train A) Degraded Grid Voltage and Loss of Voltage Protection Relays Operability Test, dated 1/17/14
SNC490740, FNP-1-STP-80.16 (Train A) Degraded Grid Voltage and Loss of Voltage Protection Relays Operability Test, dated 2/14/14
SNC490741, FNP-1-STP-80.16 (Train A) Degraded Grid Voltage and Loss of Voltage Protection Relays Operability Test, dated 3/14/14
SNC52411, Calibration of RWST LT502, dated 1/7/09
SNC52918, Perform Maintenance and Inspection of Breaker DF08, dated 10/26/12
SNC57240,1A SU XFMR Exterior Preservation Inspection, dated 4/9/12
SNC57956, Calibration of RWST LT501, dated 7/8/11
SNC57971, Calibration of RWST LT502, dated 2/1/12
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SNC58051, Perform Ultrasonic1A SU XFMR, dated 9/20/11
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SNC60348, Test HEA Relay 87T1C-X1 per FNP-2541.03, dated 4/9/12
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SNC60353, Test and Inspect SU XFMR 1A Neutral O/C Relay 51N-O, dated 4/9/12
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SNC60355, Test and Inspect SU XFMR 1A Neutral O/C Relay 51N-M, dated 4/9/12
SNC60356, Test and Inspect SU XFMR 1A O/C Relay Phase 3, dated 4/9/12
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SNC63165, Control and Alarm Checks 1A SUT, dated 4/9/12
SNC63166, Calibrate High Voltage Winding Temperature Indicator SUT 1A, dated 4/9/12
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