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

May 8, 2015

Louis P. Cortopassi, Site Vice President
Omaha Public Power District
Fort Calhoun Station
P.O. Box 550
Fort Calhoun, NE 68023-0550

Subject: FORT CALHOUN - NRC INTEGRATED INSPECTION REPORT NUMBER 05000285/2015001

Dear Mr. Cortopassi:

On March 31, 2015, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Fort Calhoun Station. On April 13, 2015, the NRC inspectors discussed the results of this inspection with you, and other members of your staff. The inspectors documented the results of this inspection in the enclosed inspection report.

NRC inspectors documented two findings of very low safety significance (Green) in this report. One of these findings involved violations of NRC requirements. Further, inspectors documented licensee-identified violations which were determined to be Severity Level IV in this report. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2.a of the NRC Enforcement Policy.

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

If you disagree with a cross-cutting aspect assignment or a finding not associated with a regulatory requirement in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region IV; and the NRC resident inspector at the Fort Calhoun Station.

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

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Sincerely,

/RA/

Geoffrey B. Miller
Chief, Project Branch D
Division of Reactor Projects

Docket: 50-285
License: DPR-40

Enclosure:
NRC Inspection Report 05000285/2015001
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-285
License:	DPR-40
Report:	05000285/2015001
Licensee:	Omaha Public Power District
Facility:	Fort Calhoun Station
Location:	9610 Power Lane Blair, NE 68008
Dates:	January 1 through March 31, 2015
Inspectors:	S. Schneider, Senior Resident Inspector B. Cummings, Resident Inspector P. Elkmann, Senior Emergency Preparedness Inspector T. Buchanan, Operations Engineer S. Hedger, Operations Engineer B. Hagar, Senior Project Engineer
Approved By:	Geoffrey B. Miller Chief, Projects Branch D Division of Reactor Projects

Enclosure

SUMMARY

IR 05000285/2015001; 1/01/2015 – 3/31/2015; Fort Calhoun Station, Licensed Operator Requalification, Maintenance Risk Assessments.

The inspection activities described in this report were performed between January 1 and March 31, 2015, by the resident inspectors at FCS, inspectors from the NRC's Region IV office, and a senior project engineer from the NRC's Region IV office. Two findings of very low safety significance (Green) are documented in this report. One of these findings involved a violation of NRC requirements. Additionally, NRC inspectors documented in this report two licensee-identified Severity Level IV violations of very low safety significance. The significance of inspection findings is indicated by their color (Green, White, Yellow, or Red), which is determined using Inspection Manual Chapter 0609, "Significance Determination Process." Their cross-cutting aspects are determined using Inspection Manual Chapter 0310, "Aspects within the Cross-Cutting Areas." Violations of NRC requirements are dispositioned in accordance with the NRC Enforcement Policy. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process."

Cornerstone: Mitigating Systems

Green. The inspectors identified a Green finding with four examples for failing to conduct and evaluate simulator performance testing in accordance with the standards of ANSI/ANS-3.5-2009. Specifically, the licensee failed to do the following:

- Set initial reactor power at 15 percent in accordance with plant design for all performances between 1990 and 2014 of Transient (6), "Main Turbine Trip from Maximum Power Level That Does Not Result in Immediate Reactor Trip"
- Set the instantaneous main turbine load reduction to 10 percent as supported by design basis data in the 2014 performance of Transient (11), "Maximum Design Load Rejection"
- Evaluate the results of the 100 percent power Steady-State Performance Test using the correct acceptance criteria in accordance with the standard, Appendix B, Section B.1.1
- Evaluate all transient test results versus acceptance criteria 4.1.4(1) in accordance with the standard, Appendix B, Section B.1.2

After NRC identification of the transient test issues, licensee evaluation revealed that the initial conditions for Transients (5) and (10) were in error as well. The licensee initiated corrective action documented in condition reports 2014-14190, 2014-14208, and 2015-02547.

The licensee's failure to conduct and evaluate performance testing in accordance with the ANSI/ANS-3.5-2009 standard as endorsed by Regulatory Guide 1.149, Revision 4, was the performance deficiency. Per licensee Procedure TQ-AA-306, "Simulator Management," the licensee uses ANSI/ANS-3.5-2009 as the standard for their simulator testing. The performance deficiency is more than minor because if left uncorrected, the performance deficiency could have become more significant in that not completing the required simulator testing correctly can lead to not detecting and correcting errors in the simulator so it actually models the plant correctly. This can both leave the potential for negative training of licensed operators and call into question the ability to conduct valid licensing examinations with the simulator. Using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," Attachment 4,

Tables 1 and 2 worksheets, and the corresponding Appendix I, "Licensed Operator Requalification Significance Determination Process (SDP)," Flowchart Block No.14, the finding was determined to have very low safety significance (Green) because it dealt with deficiencies associated with simulator testing, modification, and maintenance and there was no evidence that the plant-referenced simulator does not demonstrate the expected plant response or have uncorrected modeling and hardware deficiencies.

This finding has a cross-cutting aspect in the change management area of human performance, associated with leaders using a systematic process for evaluating and implementing change so that nuclear safety remains their overriding priority. There were efforts on-site to change to the 2009 version of the standard as early as 2011, but the efforts were rescinded by plant management in December 2011 for unknown reasons. When they officially switched from the 1985 to the 2009 version of the standard (on March 1, 2013), there is no evidence an effective change management plan was implemented. Efforts to transition between the testing and maintenance requirement differences were complicated by lack of allocating necessary resources to support this effort. There was minimal simulator staffing during the extended plant outage (April 2011 to December 2013), and no effective plan to deal with knowledge management to compensate for simulator employee turnover. Internal audits in May 2014 and October 2014 found numerous issues with their simulator testing and configuration management program, many of which could have been averted or addressed earlier with an effective transition plan in place (H.3). (Section 1R11)

<u>Green</u>. The inspectors identified a non-cited violation of very low safety significance of 10 CFR 50.65 paragraph (a)(4) "Requirements for Monitoring the Effectiveness of Maintenance of Nuclear Power Plants," because the licensee did not effectively manage the increase in risk that resulted from maintenance activities. Specifically, the licensee failed to implement key risk management actions outlined in site risk assessment and management guidance for diesel driven auxiliary feedwater (AFW) pump maintenance that resulted in a "Yellow" risk configuration. This violation was entered into the licensee's corrective action program and actions taken for this violation included verifying that all remaining online work prior to the scheduled refueling outage was properly screened and assessed in accordance with site risk management procedures. In addition, the licensee conducted training on risk management guidance that had been recently implemented during corporate alignment for personnel involved with scheduling and operations.

The inspectors determined that the licensee's failure to implement key risk management actions outlined in site risk assessment and management guidance for diesel driven AFW pump maintenance was a performance deficiency within the licensee's ability to foresee and correct and should have been prevented. The finding is more than minor because it is associated with the Equipment Performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to perform maintenance on a continuous work schedule as required by site procedures resulted in a longer unavailability time of the equipment and an extended "Yellow" risk condition. Using NRC IMC 0609, Appendix K, "Maintenance Risk Assessment and Risk Management Significance Determination Process", dated May 19, 2005, Flowchart 2, "Assessment of [Risk Management Actions]", the inspectors determined the incremental core damage probability (ICDP) associated with the maintenance activity to be approximately 1E-7, and therefore was determined to have a very low safety significance (Green), since the calculated ICDP was less than 1E-6. Because the licensee did

not use a systematic process to ensure that nuclear safety remained the overriding priority while they implemented a corporate alignment, the finding has a cross-cutting aspect in the area of Human Performance, Change Management (H.3). (Section 1R13)

Licensee-Identified Violations

Violations of very low safety significance Severity Level IV that were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. These violations and associated corrective action tracking numbers are listed in Section 4OA7 of this report.

PLANT STATUS

The unit began the inspection period at approximately 100 percent power. On January 23, 2015, the licensee reduced unit power to 98 percent power for moderator temperature coefficient testing. The unit achieved approximately 100 percent power on January 27, 2015. On March 29, 2015, the licensee reduced unit power to 85 percent power to support main condenser cleaning. The unit achieved approximately 100 percent power on March 31, 2015, and operated at that power level for the remainder of the inspection period.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

1R01 Adverse Weather Protection (71111.01)

Readiness to Cope with External Flooding

a. Inspection Scope

On January 9, the inspectors completed an inspection of the station's readiness to cope with external flooding. After reviewing the licensee's flooding analysis, the inspectors chose two plant areas that were susceptible to flooding:

- Intake structure
- Auxiliary building

The inspectors reviewed plant design features and licensee procedures for coping with flooding. The inspectors walked down the selected areas to inspect the design features, including the material condition of seals, drains, and flood barriers. The inspectors evaluated whether credited operator actions could be successfully accomplished.

This activity constituted one sample of readiness to cope with external flooding, as defined in Inspection Procedure 71111.01.

b. Findings

No findings were identified.

1R04 Equipment Alignment (71111.04)

Partial Walkdown

a. Inspection Scope

The inspectors performed partial system walk-downs of the following risk-significant systems:

- January 29, motor driven auxiliary feedwater pump during diesel driven auxiliary feedwater pump testing

- February 2, chemical and volume control system

- February 11, diesel generator number 1 during maintenance on diesel generator number 2

- March 15, component cooling water following installation of a permanent modification

The inspectors reviewed the licensee's procedures and system design information to determine the correct lineup for the systems. They visually verified that critical portions of the systems were correctly aligned for the existing plant configuration.

These activities constituted four partial system walk-down samples as defined in Inspection Procedure 71111.04.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

Quarterly Inspection

a. Inspection Scope

The inspectors evaluated the licensee's fire protection program for operational status and material condition. The inspectors focused their inspection on six plant areas important to safety:

- February 2, Room 7, chemical volume and control system valve area 1, Fire Area 9

- February 2, Room 5, charging pump area, Fire Area 10

- February 6, Room 56W, west switchgear area, Fire Area 36B

- February 6, Room 56E, east switchgear area, Fire Area 36A

- February 26, Room 19, compressor bay area, Fire Area 32

- March 3, Room 21, safety injection and containment spray pump area, Fire Area 1

For each area, the inspectors evaluated the fire plan against defined hazards and defense-in-depth features in the licensee's fire protection program. The inspectors evaluated control of transient combustibles and ignition sources, fire detection and

suppression systems, manual firefighting equipment and capability, passive fire protection features, and compensatory measures for degraded conditions.

These activities constituted six quarterly inspection samples, as defined in Inspection Procedure 71111.05.

b. Findings

No findings were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

On March 26, the inspectors completed an inspection of the station's ability to mitigate flooding due to internal causes. After reviewing the licensee's flooding analysis, the inspectors chose one plant area containing risk-significant structures, systems, and components that were susceptible to flooding:

- Room 21, safety injection pump room

The inspectors reviewed plant design features and licensee procedures for coping with internal flooding. The inspectors walked down the selected areas to inspect the design features, including the material condition of seals, drains, and flood barriers. The inspectors evaluated whether operator actions credited for flood mitigation could be successfully accomplished.

This activity constitutes completion of one flood protection measures sample, as defined in Inspection Procedure 71111.06.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program and Licensed Operator Performance (71111.11)

.1 Review of Licensed Operator Requalification

a. Inspection Scope

On February 4, the inspectors observed an evaluated simulator scenario performed by an operating crew. The inspectors assessed the performance of the operators and the evaluators' critique of their performance. The inspectors also assessed the modeling and performance of the simulator during the scenario.

This activity constitutes completion of one quarterly licensed operator requalification program sample, as defined in Inspection Procedure 71111.11.

b. Findings

No findings were identified.

.2 Review of Licensed Operator Performance

a. Inspection Scope

On March 8, the inspectors observed the performance of on-shift licensed operators in the plant's main control room. At the time of the observations, the plant was in a period of heightened activity due to the discovery of an inoperable snubber related to the auxiliary feedwater system that had placed the licensee in a 12 hour technical specification action statement. The inspectors observed the operators' performance related to the assessment of operability, TS applicability, plant risk, oversight of repairs to correct the adverse condition, and adherence to operations department procedures related to the performance of emergent work.

This activity constitutes completion of one quarterly licensed operator performance sample, as defined in Inspection Procedure 71111.11.

b. Findings

No findings were identified.

.3 Biennial Review of Requalification Program

a. Inspection Scope

The licensed operator requalification program involves two training cycles that are conducted over a 2-year period. In the first cycle, the annual cycle, the operators are administered an operating test consisting of job performance measures and simulator scenarios. In the second part of the training cycle, the biennial cycle, operators are administered an operating test and a comprehensive written examination. This inspection is a continuation of the requalification program inspection that was documented in Inspection Report 2014005. This continuation was necessary due to the timing of the facility examinations at the end of the calendar year.

To assess the performance effectiveness of the licensed operator requalification program, the inspectors reviewed both the operating tests and written examinations, and observed ongoing operating test activities.

The inspectors reviewed operator performance on the written examinations and operating tests. These reviews included observations of portions of the operating tests by the inspectors. The operating tests observed included twelve job performance measures and one scenario that were used in the current biennial requalification cycle. These observations allowed the inspectors to assess the licensee's effectiveness in conducting the operating test to ensure operator mastery of the training program content. The inspectors also reviewed medical records of eight licensed operators for conformance to license conditions and the licensee's system for tracking qualifications and records of license reactivation for three operators.

The results of these examinations were reviewed to determine the effectiveness of the licensee's appraisal of operator performance and to determine if feedback of performance analyses into the requalification training program was being accomplished. The inspectors interviewed six members of the training department and reviewed the minutes of two training review group meetings to assess the responsiveness of the licensed operator requalification program to incorporate the lessons learned from both plant and industry events. Examination results were also assessed to determine if they were consistent with the guidance contained in NUREG 1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9, Supplement 1, and NRC Manual Chapter 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process."

In addition to the above, the inspectors reviewed examination security measures, simulator fidelity, and existing logs of simulator deficiencies.

Upon completion of the exam process the licensee informed the lead inspector of the results of the written examinations and operating tests for the Licensed Operator Requalification Program. The inspectors compared these results to the Appendix I, "Licensed Operator Requalification Significance Determination Process," values and determined that there were no findings based on these results and because all of the individuals that failed the applicable portions of their examinations and/or operating tests were remediated, retested, and passed their retake examinations prior to returning to shift.

The inspectors completed one inspection sample of the biennial licensed operator requalification program.

b. <u>Findings</u>

<u>Failure to Conduct and Evaluate Simulator Testing In Accordance With ANSI/ANS-3.5-2009</u>

<u>Introduction.</u> The inspectors identified a Green finding with four examples for failing to conduct and evaluate simulator performance testing in accordance with the standards of ANSI/ANS-3.5-2009. Specifically, the licensee failed to do the following:

- Set initial reactor power at 15 percent in accordance with plant design for all performances between 1990 and 2014 of Transient (6), "Main Turbine Trip from Maximum Power Level That Does Not Result in Immediate Reactor Trip"
- Set the instantaneous main turbine load reduction to 10 percent as supported by design basis data in the 2014 performance of Transient (11), "Maximum Design Load Rejection"
- Evaluate the results of the 100 percent power Steady-State Performance Test using the correct acceptance criteria in accordance with the standard, Appendix B, Section B.1.1
- Evaluate all transient test results versus acceptance criteria 4.1.4(1) in accordance with the standard, Appendix B, Section B.1.2

Description. In order to maintain an NRC-approved simulation facility, the licensee is required to conduct performance testing throughout the life of the simulator to ensure that it can be used to model control manipulations consistent with the actual plant. An acceptable method for conducting this testing is by using industry standard ANSI/ANS-3.5, "Nuclear Power Plant Simulators for Use in Operator Training and Examination." This industry standard has been endorsed by the NRC as an acceptable method to completing required simulator testing to meet the requirements of 10 CFR 55.46 per Regulatory Guide 1.149, "Nuclear Power Plant Simulation Facilities for Use in Operator Training, License Examinations, and Applicant Experience Requirements." Per licensee procedure TQ-AA-306, "Simulator Management," the licensee uses ANSI/ANS-3.5-2009 as the standard for their simulator testing. The licensee made a decision to use the 2009 version of the standard on March 1, 2013. Prior to that, the licensee implemented the 1985 version of the same standard.

Two documented assessments of their progress in implementing the newer version of the standard were conducted. The Nuclear Oversight (NOS) audit, completed on May 16, 2014, identified in part that:

- There was no proof of comparison of reference baseline data to the results of simulator performance testing.
- The test for Transient (11), "Maximum Design Load Rejection," was not completed at all. This test was not part of the 1985 version of the standard, but is included in the 2009 version.
- There was no documented evidence that simulator noticeable differences and deviations were evaluated using training needs assessments. Training needs assessments, described in Section 4.2.1.4 of the standard, were not included in the 1985 version.
- Plant data or other sources of baseline data were not available for comparison to the simulator's transient tests results, notably for Transients (4), (5), (6), (7), (8), (10), and (11).

A licensee self-assessment completed on October 31, 2014, echoed the point made in the NOS audit indicating that the licensee was in the process of developing reference baseline data needed to complete their simulator performance testing.

The licensee initiated corrective action to address the issues identified in these assessments. One corrective action, which was to develop and complete the Transient (11) test, was completed on November 17, 2014.

Starting the week of November 17, 2014, NRC inspectors reviewed an initial sample of five transient tests. In addition, the issues identified by the assessments, the corrective actions identified, as well as the completed corrective actions were reviewed in detail. The tests for Transients (6) and (11) were evaluated, taking into account that corrective action on the test for Transient (11) was recently completed.

Example 1:

Transient (6), "Main Turbine Trip from Maximum Power Level that Does Not Result in An Immediate Reactor Trip," involves setting up the simulator to respond to a main turbine trip at the maximum reactor power level which will not cause a reactor trip by design. Test Procedure TQ-FC-302-0111, Revision 01, described the licensee's version of this test. The initial conditions, described in Section 2.0, indicated that the reactor was set up at 100 percent power. Test results further indicated that the test was set up at 100 percent reactor power, which resulted in a reactor trip following a main turbine trip. The NRC inspectors questioned the licensee about the test's initial conditions. The licensee responded that the correct initial condition should have been established with reactor power set at 15 percent. This error had been part of the initial conditions since the installation of the simulator in 1990. The licensee had documented that they had documented an exception to this test, with no basis provided.

The licensee is documenting corrective actions addressing this issue in CR 2014-14190.

Example 2:

Transient (11), "Maximum Design Load Rejection," involves initiating the maximum design step load reduction that will not result in a reactor trip. Test Procedure TQ-FC-302-0116, Revision 01, defined the amount of main turbine load reduced in Section 4.5. This basis for this plant-specific parameter is typically detailed in the Updated Safety Analysis Report (USAR), but there was no reference to the USAR document in the procedure. When the NRC inspectors asked what the basis behind the amount of step load reduction was, the licensee had to review the documentation about how the test was written. The licensee found that the test was designed by incrementally reducing the main turbine load in the simulator until the reactor tripped, instead of by finding the design step load change the reactor plant can withstand without a reactor trip. In USAR-14-9, Revision 16, Section 14.9.1.2, it states, in part, that, "The plant is designed...to accept a 10 percent step reduction in load without actuating a reactor trip."

The licensee is documenting corrective actions addressing this issue in CR 2014-14208.

Example 3:

Based on identification of Examples 1 and 2, the NRC inspectors continued the performance testing records review including all transient tests, 1 steady-state test, and 3 scenario-based test (SBT) packages. Additional feedback was provided to the licensee at the end of this review on January 22. In terms of completed steady-state tests, the licensee had completed the 100 percent power test in 2014. This test, described in Appendix B, Section B.1.1; and Section 4.1.3.1 of the standard; involves operating the simulator at a fixed reactor power level, monitoring the output of defined plant parameters, and comparison to specific tolerance bands specified in the acceptance criteria. It may be an acceptable result if it varies by less than or equal to 1 percent, 2 percent, or 10 percent, depending on the parameter. Some of the allowed tolerance bands changed between the 1985 and 2009 versions of the standard, in most cases the allow band reducing in size. Therefore, the NRC inspectors looked to see if the licensee had adjusted the tolerance bands in the test. Based on the review, it was confirmed that the licensee failed to reduce the tolerance bands for at least 14 parameters.

The licensee is documenting corrective actions addressing this issue in CR 2015-02547.

Example 4:

Part of the required testing detailed in this standard includes transient performance tests, where the licensee conducts simulator tests on eleven specific transients specified in Appendix B, Section B3.2, of the standard. Appendix B, Section B.1.2, states that the acceptance criteria for these tests are stated in Section 4.1.4. This section states that simulator response during transient testing will meet the following acceptance criteria:

(1) The simulator allows the use of applicable reference unit procedures

(2) Any observable change in simulated parameters corresponds in direction to the change expected from actual or best estimate response of the reference unit to the transient test

(3) The simulator shall not fail to cause an alarm or automatic action if the reference unit would have caused an alarm or automatic action under identical circumstances

(4) The simulator shall not cause an alarm or automatic action if the reference unit would not cause an alarm or automatic action under identical circumstances.

The licensee has a transient test procedure for each transient stated in the standard. Based on review of the acceptance criteria stated in each procedure, the NRC inspectors identified that acceptance criterion (1) as stated above was omitted from these procedures. This was communicated to the licensee on January 22. The licensee is documenting corrective actions addressing this issue in CR 2015-02547.

Licensee corrective action since identification of the NRC-identified issues included a full review of the simulator's transient testing practices. In an apparent cause evaluation associated with CR 2014-14190, completed January 8, the licensee identified that three additional transient tests had errors in their initial conditions or how the test was conducted. Subsequently, corrective actions involved rewriting test procedures for Transients (7), (9) and (10), followed by scheduled re-performance of the tests. Actions to establish baseline data for transient test evaluation are in process. Corrective actions are documented in CR 2015-02573 and CR 2015-02575.

Analysis. The licensee's failure to conduct and evaluate performance testing in accordance with the ANSI/ANS-3.5-2009 standard as endorsed by Regulatory Guide 1.149, Revision 4, was the performance deficiency. Per licensee Procedure TQ-AA-306, "Simulator Management," the licensee uses ANSI/ANS-3.5-2009 as the standard for their simulator testing. The performance deficiency is more than minor because if left uncorrected, the performance deficiency could have become more significant in that not completing the required simulator testing correctly can lead to not detecting and correcting errors in the simulator, so it actually models the plant correctly. This can both leave the potential for negative training of licensed operators and call into question the ability to conduct valid licensing examinations with the simulator. Using Manual Chapter 0609, "Significance Determination Process," Attachment 4, Tables 1 and 2 worksheets (issue date June 19, 2012), and the corresponding Appendix I, "Licensed Operator Requalification Significance Determination Process (SDP)," Flowchart Block

No.14 (issue date December 6, 2011), the finding was determined to have very low safety significance (Green) because the issue was associated with simulator testing, modifications, and maintenance, and there was no evidence that the plant-referenced simulator does not demonstrate the expected plant response or have uncorrected modeling and hardware deficiencies.

This finding has a cross-cutting aspect in the change management area of human performance, associated with leaders using a systematic process for evaluating and implementing change so that nuclear safety remains their overriding priority. Per the 2014-14190 apparent cause evaluation, there were efforts on-site to change to the 2009 version of the standard as early as 2011, but the efforts were rescinded by plant management in December 2011 for unknown reasons. Although the licensee adopted the 2009 version of the standard on March 1, 2013, there is no evidence an effective change management plan was implemented. Efforts to transition between the testing and maintenance requirement differences were complicated by lack of allocating necessary resources to support this effort. There was minimal simulator staffing during the extended plant outage (April 2011 to December 2013), and no effective plan to deal with knowledge management to compensate for simulator employee turnover. Internal audits in May 2014 and October 2014 found numerous issues with their simulator testing and configuration management program, many of which could have been averted or addressed earlier with an effective transition plan in place [H.3].

Enforcement. This finding does not involve enforcement action because no violation of regulatory requirements was identified. Because this finding does not involve a violation and has very low safety significance, it is identified as FIN 05000285/2015001-01, "Failure to Conduct and Evaluate Simulator Testing In Accordance With ANSI/ANS-3.5-2009."

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors reviewed one instance of degraded performance or condition of safety-related structures, systems, and components (SSCs):

- March 2, CH-1B charging pump (a)(1) action plan

The inspectors reviewed the extent of condition of possible common cause SSC failures and evaluated the adequacy of the licensee's corrective actions. The inspectors reviewed the licensee's work practices to evaluate whether these may have played a role in the degradation of the SSCs. The inspectors assessed the licensee's characterization of the degradation in accordance with 10 CFR 50.65 (the Maintenance Rule), and verified that the licensee was appropriately tracking degraded performance and conditions in accordance with the Maintenance Rule.

These activities constitute completion of one maintenance effectiveness sample, as defined in Inspection Procedure 71111.12

b. Findings

No findings were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

a. Inspection Scope

The inspectors reviewed six risk assessments performed by the licensee prior to changes in plant configuration and the risk management actions taken by the licensee in response to elevated risk:

- January 7, yellow risk during channel "A" safety injection, containment spray, and recirculation actuation signal test

- January 29, yellow risk during performance of the diesel driven auxiliary feedwater pump performance test

- February 11, yellow risk during raw water strainer "B" maintenance and steam driven auxiliary feedwater system testing

- February 25, yellow risk during diesel driven auxiliary feedwater pump maintenance

- March 4, yellow risk during emergency diesel generator number 1 maintenance

- March 16, yellow risk during emergency diesel generator number 2 and safety injection system snubber maintenance

The inspectors verified that these risk assessments were performed timely and in accordance with the requirements of 10 CFR 50.65 (the Maintenance Rule) and plant procedures. The inspectors reviewed the accuracy and completeness of the licensee's risk assessments and verified that the licensee implemented appropriate risk management actions based on the result of the assessments.

These activities constitute completion of six maintenance risk assessment inspection samples, as defined in Inspection Procedure 71111.13.

b. Findings

Failure to Implement Risk Management Actions for Planned Maintenance Activities

Introduction. The inspectors identified a non-cited violation of very low safety significance (Green) of 10 CFR 50.65 paragraph (a)(4) "Requirements for Monitoring the Effectiveness of Maintenance of Nuclear Power Plants," because the licensee failed to implement key risk management actions (RMAs) outlined in site risk assessment and management guidance for diesel-driven auxiliary feedwater (AFW) pump maintenance that resulted in a "Yellow" risk configuration.

Description. The inspectors reviewed the risk assessment and RMAs associated with diesel-driven AFW pump maintenance activities conducted from February 24th to February 26th. The licensee's risk assessment had identified a "Yellow" online risk configuration per licensee procedure FCSG-19, "Risk Assessments", that would require implementation of RMAs per licensee procedure WC-AA-101, "On-Line Work Control

Processes". WC-AA-101 requires that a continuous work schedule be established for maintenance activities that place the plant in a "Yellow" risk configuration to limit the duration of equipment unavailability due to the maintenance. Otherwise, a specific Plant Manager or designee exemption is required.

On February 25 and at the site of a maintenance activity that affected the diesel-driven AFW pump, the inspectors questioned the licensee about the implementation of RMAs required by WC-AA-101. Maintenance workers told the inspectors that a continuous work schedule for the maintenance had not been established, and that no diesel-driven AFW pump maintenance would be performed on the evening shift. The inspectors asked the licensee whether a specific exemption per WC-AA-101 had been granted by the Plant Manager. The following morning, the licensee told the inspectors that a specific exemption had not been granted by the Plant Manager, and that they had initiated Condition Report 2015-02302 to document that deficiency. During this discussion, the licensee told the inspectors that they had implemented this specific requirement only recently as part of a corporate alignment, and that they had not ensured that all work schedulers were familiar with it. The inspectors considered that this information indicated that the licensee had not used a systematic process to ensure that the changes they made during the corporate realignment did not impact nuclear safety.

Furthermore, the licensee then told the inspectors that they had not implemented another RMA required by SO-G-123, "Protected Equipment Program" prior to commencement of the diesel drivel AFW pump maintenance. That RMA requires the use of administrative controls to physically protect the steam-driven and motor-driven AFW pumps to ensure that backup equipment remained functional to prevent an unacceptable risk configuration. The licensee discovered this issue shortly after they established the clearance for the diesel-driven AFW pump, and they took immediate corrective action to protect the backup equipment.

Analysis. The inspectors determined that the licensee's failure to implement the station's RMAs in accordance with WC-AA-101 for the diesel-driven AFW pump maintenance was a performance deficiency within the licensee's ability to foresee and correct and therefore should have been prevented. That performance deficiency is more than minor and therefore is a finding because it is associated with the Equipment Performance attribute of the Mitigating Systems cornerstone and adversely affected the Cornerstone objective to ensure the availability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to perform maintenance on a continuous work schedule as required by site procedures resulted in a longer unavailability time of the equipment and an extended "Yellow" risk condition.

NRC IMC 0612, Appendix K, "Maintenance Risk Assessment and Risk Management Significance Determination Process (SDP)", provides guidance for SDP screening related to Maintenance Rule (a)(4) inspection findings. NRC IMC 0609, Appendix K, Attachment 1, states that implementation of RMAs associated with maintenance activities are a key element of the licensee's risk management program and that measures to minimize the duration of the risk associated with the maintenance activity are the principal RMA. NRC IMC 0612, Appendix K was utilized in this case for significance determination since the licensee failed to implement RMAs that would reduce the duration of the maintenance activity as required by site procedure WC-AA-104.

Using NRC IMC 0609, Appendix K, "Maintenance Risk Assessment and Risk Management Significance Determination Process", dated May 19, 2005, Flowchart 2, "Assessment of RMAs", the inspectors determined the incremental core damage probability (ICDP) associated with the maintenance activity to be approximately 1 x 10-7, and therefore determined that the finding has a very low safety significance (Green), since the calculated ICDP was less than 1 x 10-6. Because the licensee did not use a systematic process to ensure that nuclear safety remained the overriding priority while they implemented a corporate alignment, the finding has a cross-cutting aspect in the area of Human Performance, Change Management (H.3).

Enforcement. 10 CFR 50.65(a)(4) requires, in part, that before performing maintenance activities, the licensee shall assess and manage the increase in risk that may result from proposed maintenance activities. Contrary to the above, before performing certain maintenance activities, the licensee assessed but failed to manage the increase in risk that resulted from those maintenance activities. Specifically, between February 24th, 2015, and February 26th, 2015, and before performing maintenance that affected the diesel-driven AFW pump, the licensee failed to implement RMAs to support maintenance on a continuous work schedule and to protect redundant plant equipment in the presence of an elevated risk condition. Following the discovery of this performance deficiency, the licensee took corrective actions to ensure that all remaining online work prior to the scheduled refueling outage was properly screened and assessed in accordance with site risk management procedures. In addition, the licensee conducted training on risk-management guidance that had been recently implemented during corporate alignment, for personnel involved with scheduling and operations. Because this violation was of very low safety significance and was entered into the licensee's corrective action program (CR 2015-02302), this violation is being treated as an NCV, consistent with section 2.3.2 of the NRC's Enforcement Policy.
(NCV 05000285/2015001-02, Failure to Implement Risk Management Actions for Planned Maintenance Activities.)

1R15 Operability Determinations and Functionality Assessments (71111.15)

a. Inspection Scope

The inspectors reviewed five operability determinations that the licensee performed for degraded or nonconforming structures, systems, or components (SSCs):

- January 15, operability determination following 10 CFR 21 determination of potentially defective control switches installed in 480V busses

- February 5, operability determination of emergency diesel generator number 1 following discovery of an air supply damper malfunction

- February 26, operability determination upon discovery of oil leaks on a control room air conditioner

- March 11, operability determination of snubber FWS-5A following quality control visual inspection rejection

- March 24, operability determination following discovery that low pressure safety injection pumps may not deliver expected flow when nearing shutoff head of the pump curve

The inspectors reviewed the timeliness and technical adequacy of the licensee's evaluations. Where the licensee determined the degraded SSC to be operable, the inspectors verified that the licensee's compensatory measures were appropriate to provide reasonable assurance of operability. The inspectors verified that the licensee had considered the effect of other degraded conditions on the operability of the degraded SSC.

These activities constitute completion of five operability review samples, as defined in Inspection Procedure 71111.15.

b. Findings

No findings were identified. The inspectors identified one unresolved item associated with required in-service testing for steam generator auxiliary feedwater inlet valves which is related to an inspection sample documented in Inspection Report 05000285/2014005.

Unresolved Item associated with Required In-Service Testing for Steam Generator Auxiliary Feedwater Inlet Valves

Introduction. The inspectors identified an unresolved item associated with required in-service testing (IST) for steam generator auxiliary feedwater (AFW) inlet valves. The licensee is updating a design analysis for these valves to establish a formal technical position with regard to the "active" safety functions that these valves serve. The inspectors will review the licensee's updated design analysis upon its completion to determine if the IST is being performed per the IST ASME OM Code.

Description. On December 22, 2014, the inspectors reviewed an operability determination following the licensee's discovery of a degraded condition related to a leak identified on AFW valve HCV-1108A. The inspectors reviewed surveillance testing required by Technical Specification (TS) 3.9.3 which requires that the operability of HCV-1108A be confirmed in accordance with the licensee's IST program. The inspectors identified that these AFW valves had both open and close "active" safety functions per design basis document EA11-026, "CQE Valves w/Essential Accumulators". In addition, the valve is classified as a category "B" valve per their IST program.

The licensee committed to ASME OM Code 1998 Edition w/1999 through 2000 addenda for their IST program. Applicable to this Code, an "active" safety function requires the valve to change position to accomplish its specific safety function. Category "B" valves with these "active" functions are required to be stroke time tested in the direction of valve repositioning. The inspectors discussed these requirements with the licensee and they subsequently identified that they were not performing ASME IST for HCV-1107A/B and HCV-1108A/B for the "active" close safety function of the valves. The licensee is performing a review of the "active" close safety function to determine whether the "active" safety function described in EA11-026 is appropriate.

The inspectors will review the licensee determination and assess whether the testing being performed on the steam generator AFW inlet valves meets the applicable ASME

OM Code requirements. This issue will be tracked as unresolved item (URI) 05000285/2015001-03.

1R18 Plant Modifications (71111.18)

a. Inspection Scope

On March 13, the inspectors reviewed a permanent modification involving the installation of a gas accumulation tank on the suction side of the "A" component cooling water pump AC-3A.

The inspectors reviewed the design and implementation of the modification. The inspectors verified that work activities involved in implementing the modification did not adversely impact operator actions that may be required in response to an emergency or other unplanned event. The inspectors verified that post-modification testing was adequate to establish the operability of the SSC as modified.

These activities constitute completion of one sample of permanent modifications, as defined in Inspection Procedure 71111.18.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed six post-maintenance testing activities that affected risk-significant SSCs:

- February 3, emergency diesel generator number 2 following replacement of switch relays
- February 10, emergency diesel generator number 1 lube oil pump leak repair and protective relay clean and inspect
- February 20, emergency diesel generator number 2 following replacement of primary starting air receivers
- February 20, high pressure safety injection pump SI-2B following repair of leaking weld
- February 23, emergency diesel generator number 2 following replacement of secondary starting air receivers
- February 24, chemical volume and control system, charging pump CH-1C, following pump rebuild

The inspectors reviewed licensing and design basis documents for the SSCs and the maintenance and post-maintenance test procedures. The inspectors observed the performance of the post-maintenance tests and/or reviewed the test results to verify that the licensee performed the tests in accordance with approved procedures, satisfied the established acceptance criteria, and restored the operability of the affected SSCs.

These activities constitute completion of six post-maintenance testing inspection samples, as defined in Inspection Procedure 71111.19.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed five risk-significant surveillance tests and/or reviewed the test results to verify that these tests adequately demonstrated that the SSCs were capable of performing their safety functions:

In-service tests:

- January 16, steam driven auxiliary feedwater pump isolation valve and check valve tests

- January 23, emergency diesel generator number 2 starting air compressors discharge check valves exercise test

- March 23, AC-3B component cooling water pump in-service test

Other surveillance tests:

- January 21, emergency diesel generator number 2 monthly load testing

- March 23, visual inspection of Bergen Paterson Hydraulic Shock Suppressor units in the auxiliary building

The inspectors verified that these tests met technical specification requirements, that the licensee performed the tests in accordance with their procedures, and that the results of the test satisfied appropriate acceptance criteria. The inspectors verified that the licensee restored the operability of the affected SSCs following testing.

These activities constitute completion of five surveillance testing inspection samples, as defined in Inspection Procedure 71111.22.

b. Findings

No findings were identified.

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspector performed an in-office review of:

- Radiological Emergency Response Plan, Section B, "Organizational Control of Emergencies," Revision 32;
- Radiological Emergency Response Plan, Section E, "Notification Methods and Procedures," Revision 29;
- Radiological Emergency Response Plan, Section H, "Emergency Facilities and Equipment," Revision 40;
- Radiological Emergency Response Plan, Section J, "Protective Response," Revision 23;
- Emergency Plan Implementing Procedure EOF-6, "Offsite Dose Assessment using the Unified RASCAL Interface," Revision 48; and
- Emergency Plan Implementing Procedure EOF-7, "Protective Action Guidelines," Revision 26.

These changes to the site emergency plan and emergency plan implementing procedures were submitted by the licensee on December 16 and December 17, 2014. These revisions:

- Implemented the Unified RASCAL Interface dose assessment model as a replacement for the licensee's previous model, EAGLE;
- Incorporates EP-FC-1001, Addendum 1, "On Shift Staffing Analysis Report," into the emergency plan;
- Identified that the alternate Technical Support Center and alternate Operations Support Center are located at the Emergency Operations Facility building in North Omaha;
- Defined the back-up method for notifying the public in the emergency planning zone of an emergency condition at the licensee's plant;
- Defined that testing of the Alert and Notification System is performed as specified in the Federal Emergency Management Agency-approved system design report; and
- Updated other references in the Radiological Emergency Response Plan.

These revisions were compared to their previous revisions, to the criteria of NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, and to the standards in 10 CFR 50.47(b) to determine if the revisions adequately

implemented the requirements of 10 CFR 50.54(q)(3) and 50.54(q)(4). The inspector verified that the revisions did not decrease the effectiveness of the emergency plan. These reviews were not documented in a safety evaluation report and did not constitute approval of licensee-generated changes; therefore, these revisions are subject to future inspection.

These activities constitute completion of six emergency action level and emergency plan change samples as defined in Inspection Procedure 71114.04.

b. Findings

No findings were identified.

1EP6 Drill Evaluation (71114.06)

Training Evolution Observation

a. Inspection Scope

On February 4, the inspectors observed simulator-based licensed operator requalification training that included implementation of the licensee's emergency plan. The inspectors verified that the licensee's emergency classifications and off-site notifications were appropriate and timely. The inspectors verified that any emergency preparedness weaknesses were appropriately identified by the evaluators and entered into the corrective action program for resolution.

This activity constituted completion of one training observation sample, as defined in Inspection Procedure 71114.06

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Public Radiation Safety, Occupational Radiation Safety, and Security

4OA1 Performance Indicator Verification (71151)

.1 Safety System Functional Failures (MS05)

a. Inspection Scope

For the period of January 1st, 2014 through December 31st 2014, the inspectors reviewed licensee event reports (LERs), maintenance rule evaluations, and other records that could indicate whether safety system functional failures had occurred. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, and NUREG-1022, "Event Reporting Guidelines: 10 CFR 50.72 and 50.73," Revision 3, to determine the accuracy of the data reported.

These activities constituted verification of the safety system functional failures performance indicator, as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

.2 Reactor Coolant System Identified Leakage (BI02)

a. Inspection Scope

The inspectors reviewed the licensee's records of reactor coolant system leakage for the period of January 1st, 2014 through December 31st 2014 to verify the accuracy and completeness of the reported data. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, to determine the accuracy of the reported data.

These activities constituted verification of the reactor coolant system leakage performance indicator, as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution (71152)

.1 Routine Review

a. Inspection Scope

Throughout the inspection period, the inspectors performed daily reviews of items entered into the licensee's corrective action program and periodically attended the licensee's condition report screening meetings. The inspectors verified that licensee personnel were identifying problems at an appropriate threshold and entering these problems into the corrective action program for resolution. The inspectors verified that the licensee developed and implemented corrective actions commensurate with the significance of the problems identified. The inspectors also reviewed the licensee's problem identification and resolution activities during the performance of the other inspection activities documented in this report.

b. Findings

No findings were identified.

.2 Annual Follow-up of Selected Issues

a. Inspection Scope

The inspectors selected one issue for an in-depth follow-up:

- On March 31, systemic introduction of air into the component cooling water system (CR 2014-08639)

The inspectors assessed the licensee's problem identification threshold, apparent cause analysis, and extent of condition reviews, proposed and completed corrective actions, and the prioritization and timeliness of the corrective actions to evaluate whether the licensee was appropriately identifying, characterizing, and correcting problems associated with air introduction into the CCW system and whether the planned and/or completed corrective actions were appropriate. The inspectors compared the actions taken to the requirements of the licensee's corrective action program, 10 CFR 50, Appendix B, and the licensee's procedures. In addition, the inspectors interviewed licensee personnel to understand the licensee's causal analysis approach and to assess the effectiveness of the implemented corrective actions.

These activities constitute completion of one annual follow-up sample as defined in Inspection Procedure 71152.

b. Findings

No findings were identified.

4OA3 Follow-up of Events and Notices of Enforcement Discretion (71153)

.1 Plant Events

a. Inspection Scope

For the plant event listed below, the inspectors reviewed and observed plant parameters, reviewed personnel performance, and evaluated performance of mitigating systems as applicable. The inspectors communicated the plant event to appropriate regional personnel, and compared the event details with criteria contained in Inspection Manual Chapter 0309, "Reactive Inspection Decision Basis for Reactors," for consideration of potential reactive inspection activities. As applicable, the inspectors verified that the licensee made appropriate emergency classification assessments and properly reported the event in accordance with 10 CFR Parts 50.72 and 50.73. The inspectors reviewed the licensee's follow-up actions related to the event to assure that the licensee implemented appropriate corrective actions commensurate with their safety significance.

- Operator response to an unplanned Technical Specification (TS) action statement entry when a snubber affecting both trains of auxiliary feedwater was declared inoperable on March 8.

These activities constitute completion of one event follow-up sample, as defined in Inspection Procedure 71153.

b. Findings

No findings were identified.

.2 (Closed) Licensee Event Report (LER) 05000285/2013008-1, "Previously Installed General Electric IVA Relays Failed Seismic Testing"

The licensee submitted this report because on April 11, 2013, three previously installed General Electric (GE) IVA relays did not pass seismic testing. Those relays were safety-related and had been seismically qualified. After the licensee entered this condition into their corrective action program as condition report 2013-08095, their causal analysis determined that the cause of the failure was the control spring in the relay contacting either the disk or the drag magnet during seismic testing, resulting in a short. Their causal analysis also determined that a wire used to support the spring was not installed in the relays that failed the testing, thus allowing the control spring to sag and make electrical contact. To address this condition, the licensee replaced the relays that had failed the seismic testing. They also completed an extent-of-condition review that identified 32 similar relays. Of those 32, 14 either had the support wire installed or had already been replaced with a device that had the support wire. The licensee replaced the remaining 18. The inspectors considered the licensee's actions to be adequate and appropriate, and did not identify any related performance deficiency. Therefore, this LER is closed. Documents reviewed are listed in the Attachment.

.3 (Closed) Licensee Event Report (LER) 05000285/2013-018-0, "Postulated Fire Event Could Result in Shorts Impacting Safe Shutdown"

The licensee submitted this report after they discovered that in the event of a fire in the control room, unfused Direct Current (DC) ammeter circuits could become shorted to ground and develop ground loops through the unprotected ammeter wiring. The licensee had made this discovery when they assessed whether the condition described in NRC Event Report 49422 was applicable to their site. To address this vulnerability, the licensee initiated and processed CR 2013-19107. They subsequently developed Engineering Change 62826 to include fuses in the design of the affected ammeter circuits, and work orders to install the subject fuses. At the time of this inspection, the licensee had installed fuses into the affected circuits that were accessible during normal operation, and had developed plans to install fuses into the remaining affected circuits during the upcoming refueling outage. The inspectors considered these responses to be appropriate and adequate, and did not identify any associated performance deficiency. Therefore, this LER is closed. Documents reviewed are listed in the Attachment.

.4 (Closed) Unresolved Item (URI) 05000285/2013008-07, "Administrative Controls for a Technical Specification for Low River Level"

The NRC opened this URI because an inspection team had assessed licensee TS 2.16 as apparently deficient, because:

- TS 2.16 contained a limiting condition for operation for the Missouri River water

level to remain greater than or equal to 976 feet 9 inches mean sea level (MSL) as measured at the intake structure;

- plant procedures have operators secure the raw water pumps at an intake cell level of 976 feet 9 inches;
- with raw water pumps secured, those pumps would not be available for heat removal;
- because of trash racks and travelling screens between the intake cell and the river, cell level will be lower than river level; and
- the team questioned how the technical-specification-required action to place the plant in cold shutdown at a river level 976 feet 9 inches could be carried out with the raw water pumps secured by procedure.

The team therefore questioned whether the river level of 976 feet 9 inches in TS 2.16 was conservative, and whether a higher river level would be more appropriate to assure plant safety. Also, because the licensee had evaluated this condition in Condition Report (CR) 2006-03381 and had not initiated a change to the river level of 976 feet 9 inches in TS 2.16, the team considered that these circumstances might constitute untimely corrective action for a deficient technical specification.

The inspectors reviewed CR 2006-03381, and noted that in action item 2006-03381-011, the licensee stated that a change to the TS 2.16 minimum river level of 976'-9" was not necessary, and that they had prepared Technical Specification Basis Change (TSBC) 07-002-0 to add to the bases for TS 2.16 a statement that the circulating water pumps have a much higher MSL requirement (983 feet 0 inches) and would become unstable and trip or be manually shutdown well before intake cell levels decrease to the raw water pump MSL. The licensee submitted that statement as part of Licensing Amendment Request (LAR) 12-003, and the NRC subsequently approved that LAR in a letter dated January 28, 2014. In the Safety Evaluation that accompanied that letter, the NRC said,

> *"In addition, the Updated Safety Analysis Report (USAR) Section 9.8 indicates that although intake cell levels are also adversely affected by the flows associated with the non-safety related [circulating water (CW)] pumps (since the large flow rates associated with the CW pumps create significant head losses even with relatively clean intake cell conditions), the CW pumps have a much higher [mean sea level (MSL)] requirement (983 feet 0 inches) and would become unstable and trip or be manually shutdown well before intake cell levels decrease to the [raw water (RW)] pump MSL of 976 feet 9 inches. The head loss associated with CW pump flow would then be recovered and intake cell levels would rise."*

In the same Safety Evaluation, the NRC staff also stated that TS 2.16 was acceptable. Thus, if river water level were to steadily decrease toward 976 feet 9 inches, before level decreased to 983 feet, the licensee would shutdown the plant and initiate shutdown cooling before they secured the circulating water pumps, well before level decreased to 976 feet 9 inches. Therefore, the licensee could not reasonably encounter the scenario of concern to the inspection team, in which they would be required to initiate a technical-specification-required action to place the plant in cold shutdown at a river level 976 feet 9 inches with the raw water pumps secured by procedure. Furthermore, in the associated circumstances, the inspectors did not

identify any performance deficiency. Therefore, this URI is closed. The documents reviewed are listed in the Attachment.

.5 <u>(Closed) VIO 05000285/2014002-05, "Untimely Submittal of Required Licensee Event Reports."</u>

The NRC issued this violation because the licensee had failed to submit Licensee Event Reports (LERs) for two events that met the requirements specified in 10 CFR 50.73 for reporting within 60 days after the discovery of each event. Specifically, the licensee had submitted LERs 2013-101-0 and 2013-017-0 more than 60 days after they had discovered the corresponding events.

The licensee addressed this violation in condition report 2014-03669. That CR reported that the licensee's process analysis had revealed a weak LER reporting program. To address this weakness, the licensee reinforced reportability determination standards during licensed operator requalification training, and established a weekly review by the operations department of recently completed immediate reportability determinations in accordance with station procedure OP-FC-108-1003. Furthermore, the licensee implemented the Exelon Reportability Reference Manual to ensure timely event reporting.

The NRC considers the licensee to have restored compliance and to have taken appropriate corrective actions to address the cause of this violation. Therefore, this violation is closed.

4OA6 Meetings, Including Exit

<u>Exit Meeting Summary</u>

O March 5, the inspectors conducted a telephonic exit meeting with Mr. S. Dean, Plant Manager, and other members of the licensee's staff of the results of the licensed operator requalification program inspection. The licensee representatives acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

On March 12, the inspector conducted a telephonic exit meeting to present the results of the in-office inspection of changes to the licensee's emergency plan and emergency action levels to Mr. E. Plautz, Manager, Emergency Preparedness, and other members of the licensee staff. The licensee acknowledged the results.

On April 13, the inspectors presented the integrated inspection results to Mr. Lou Cortopassi and other members of the licensee's staff. The licensee acknowledged the issues presented. The licensee confirmed that any proprietary information reviewed by the inspectors had been returned or destroyed.

4OA7 Licensee-Identified Violations

The following Severity Level IV violations were identified by the licensee and are violations of NRC requirements which meet the criteria of the NRC Enforcement Policy, for being dispositioned as non-cited violations.

.1 Title 10 CFR 50.74, "Notification of Change in Operator or Senior Operator Status," requires that "each licensee shall notify the appropriate Regional Administrator within 30 days in regard to a licensed operator or senior operator: permanent disability or illness as described in Part 55.25." Contrary to the above, from January 27, 2014, to April 23, 2014, the licensee failed to notify the Regional Administrator within 30 days of a permanent disability or illness of a licensed operator. Specifically, a licensed operator was prescribed medication for hypertension; however, this condition was not reported to the NRC as required by 10 CFR 55.25.

The licensee documented the deficiency in Condition Report 2014-05185. The failure to report required information to the NRC is a violation. The violation was evaluated using the traditional enforcement process because it impacted the NRC's ability to perform its regulatory function. The violation was determined to be Severity Level IV because it fits the example of Enforcement Policy Section 6.4.d.1(d), "Violation Examples: Licensed Reactor Operators." This section states, "SL IV violations involve, for example: ... an individual operator who met ANSI/ANS 3.4, Section 5, as certified on NRC Form 396, required by 10 CFR 55.23, but failed to report a condition that would have required a license restriction to establish or maintain medical qualification based on having the undisclosed medical condition."

.2 Title 10 CFR 50.74, "Notification of Change in Operator or Senior Operator Status," requires that "each licensee shall notify the appropriate Regional Administrator within 30 days in regard to a licensed operator or senior operator: permanent disability or illness as described in Part 55.25." Contrary to the above, from February 17, 2013, to April 16, 2014, the licensee failed to notify the Regional Administrator within 30 days of a permanent disability or illness of a licensed operator. Specifically, a licensed operator was prescribed medication for hypertension; however, this condition was not reported to the NRC as required by 10 CFR 55.25.

The licensee documented the deficiency in Condition Report 2014-05185. The failure to report required information to the NRC is a violation. The violation was evaluated using the traditional enforcement process because it impacted the NRC's ability to perform its regulatory function. The violation was determined to be Severity Level IV because it fits the example of Enforcement Policy Section 6.4.d.1(d), "Violation Examples: Licensed Reactor Operators." This section states, "SL IV violations involve, for example: ... an individual operator who met ANSI/ANS 3.4, Section 5, as certified on NRC Form 396, required by 10 CFR 55.23, but failed to report a condition that would have required a license restriction to establish or maintain medical qualification based on having the undisclosed medical condition." In this case, the licensed operator would have met the ANSI/ANS 3.4 certification; but would have required a "must take medication as required" license restriction.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

J. Wilson, Manager, Design Engineering
D. Bakalar, Manager, Security
D. Bonwell, Licensed Operator Requalification Training Supervisor
C. Cameron, Regulatory Compliance
L. Cortopassi, Site Vice President
S. Dean, Plant Manager
S. Fatora, Director, Site Work Management
M. Ferm, Manager, System Engineering
H. Goodman, Site Engineering Director
M. Joe, Manager, Operations Training
R. Hugenroth, Supervisor Nuclear Oversight
K. Ihnen, Manager, Site Nuclear Oversight
R. Lowery, Senior Operations Training Instructor
E. Matzke, Senior Licensing Engineer
J. McManis, Manager Engineering Programs
E. Plautz, Manager, Site Emergency Planning
J. Short, Manager, Instrument and Control Maintenance
T. Simpkin, Manager, Site Regulatory Assurance
S. Swanson, Director, Operations
R. Thomas, Supervisor, Nuclear Engineering
T. Uehling, Manager, Site Training

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000285/2015001-03	URI	Required In-Service Testing for Steam Generator Auxiliary Feedwater Inlet Valves (Section 1R15)

Opened and Closed

05000285/2015001-01	FIN	Failure to Conduct and Evaluate Simulator Testing In Accordance with ANSI/ANS-3.5-2009 (Section 1R11)
05000285/2015001-02	NCV	Failure to Implement Risk Management Actions for Planned Maintenance Activities (Section 1R13)

Closed

05000285/2013-008-1	LER	Previously Installed General Electric IVA Relays Failed Seismic Testing (Section 4OA3)
05000285/2013-018-0	LER	Postulated Fire Event Could Result in Shorts Impacting Safe Shutdown (Section 4OA3)
05000285/2013008-07	URI	Administrative Controls for a Technical Specification for Low River Level (Section 4OA3)
05000285/2014002-05	VIO	Untimely Submittal of Required Licensee Event Reports (Section 4OA3)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Procedures

Number	Title	Revision
AOP-01	Acts of Nature	42
PE-RR-AE-1001	Flood Barrier and Sandbag Staging and Installation	22
USAR 2.7	Hydrology	14

Section 1R04: Equipment Alignment

Procedures

Number	Title	Revision
ER-AP-331	Boric Acid Corrosion Control Program	7
OI-AFW-1	Auxiliary Feedwater Actuation System Normal Operation	83
OI-CC-1	Component cooling system normal operation	82
OI-DG-1	Diesel generator #1	63
OP-PM-AFW-001	Auxiliary Feedwater System Flow verification using FW-10	0
OP-ST-AFW-0001	Auxiliary Feedwater System valve alignment check	24
SO-G-20A	Equipment Tagging Program	42
SO-G-123	Protected Equipment Program	8

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2013-20452	2013-20615	2013-20841	2013-21146	2015-00393
2015-00841	2015-3175	2015-21417		

Miscellaneous Documents

Number	Title	Revision
DBD-CH-08	Chemical and Volume Control System Design Basis Document	29
USAR 9.2	Chemical and Volume Control System	27

Drawings

Number	Title	Revision
33287	Composite Flow Diagram Chemical and Volume Control System P&ID	52

Section 1R05: Fire Protection

Procedures

Number	Title	Revision
AOP-06-02	Fire Emergency, Uncontrolled Access of Auxiliary Building	59

Procedures

Number	Title	Revision
FHA-EA-97-001	Fire Hazards Analysis (FHA) Manual	17
OP-AA-201-009	Control and Transportation of Combustible Material	16
SO-G-28	Station Fire Plan	87
SO-G-91	Control and Transportation of Combustible Material	30
USAR 9.11	Fire Protection System	25

Drawings

Number	Title	Revision
B-4701	Sh. 1, Penetration Layout Drawing Fire Area 1	0

Miscellaneous Documents

Number	Title	Revision
USAR 9.11	Fire Protection System	25

Section 1R06: Flood Protection Measures

Procedures

Number	Title	Revision
CC-FC-201-AD-Flood	Flood Barrier Supporting Documentation	0
CC-AA-201	Flood Barrier Control Program	10
K-CP-01-0565	Calibration of Safety Injection Pump Room 21, Sump Level Loop L-565	4
SO-M-26	Calibration Program	36

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2009-2691	2010-6512	2011-3255	2012-7543

Drawings

Number	Title	Revision
11405-M-7	Composite Flow Diagram Waste Disposal P&ID	36

Miscellaneous Documents

Number	Title	Revision
EA08-010	Internal Flooding	0
ALION-REP-OPPD—3173-027	Engineering Analysis for OPPD Internal Flooding Design Bases	0
LIC-93-0278	NRC Generic Letter 88-20 Submittal for Fort Calhoun Station "Individual Plant Examination for Severe Accident Vulnerabilities, 12-1-1993	
EA99-023	Fire Suppression Effects Analysis	0
USAR 11.1	Radioactive Waste Disposal System	20

Section 1R11: Licensed Operator Requalification Program and Licensed Operator Performance

Procedures

Number	Title	Revision
FC-1300A	Instructions for Completing Medical Examinations	6
OP-AA-105-102	NRC Active License Maintenance	11
OPD-3-11	License Activation and Watchstation Maintenance	19
SO-G-64	Medical Examination Program for Worker Qualification	35
SO-O-42	Notification to the NRC of Licensed Personnel Disability, Permanent Reassignment or Termination	8
TQ-FC-JA-201-01	Appendix B – Initial and Annual NRC Exam, In-Plant Security Checklist	0
TQ-AA-224	Exelon Nuclear Training – Implementation Phase	8
TQ-AA-150	Operator Training Programs	10
TQ-JA-150-03	JPM Briefing Job Aid	3
TQ-JA-150-20	LORT Annual Exam Development Job Aid	7
TQ-AA-155	Conduct of Simulator Training and Evaluation	3
TQ-AA-201	Exam Security and Administration	15
TQ-AA-201-F-02	Exam Proctor Checklist	4
TQ-AA-306	Simulator Management	7

	Week 1 SRO Written Examination	
	Week 2 Simulator Scenario	9/18/14
	Sniff Test Completion Dates/Results	
	Computer Usage Expectation for Operations Training Exams	6/23/14
	Licensed Operator Reactivation Packages	
AI 2014-14190	Check-In Self-Assessment: Determine Adequacy and Alignment of Transient Tests with Standard	1/30/15
Test No. 14.5.3.3.4	Steady-State Accuracy Test at 30% Power	12/16/12
Test No. 14.5.3.3.1	Steady State Accuracy Test at 100% Power	12/6/12
JPM Number P-0719	Startup Hydrogen Purge System	14
JPM Number P-450-F1	Emergency Start of the Diesel Fire Pump – Alternate Path	10
JPM Number S-0006-F5	SPTA With a Loss of All CCW (Faulted)	0 (2013 Crew B)
JPM Number S-0785	AOP -18, Raw Water Leak on East Header	0 (2013, Crew B)
JPM Number A-0362-15	No ATWS – Rx Trip From Clutch Power Supply Breakers	0 (2013, Crew B)
JPM Number P-0030F	Respond to Loss of Instrument Air	0 (2013, Crew B)
JPM Number P-0028	AOP-9, High Radioactivity Local Operation	0 (2013, Crew B)
TQ-FC-302-0110	Fort Calhoun Simulator Transient Testing (TR-5), Trip of Any Single Reactor Coolant Pump	11/13/14
TQ-FC-302-0115	Fort Calhoun Simulator Transient Testing (TR-10), Slow Primary System Depressurization to Saturated Condition Using Pressurizer Relief Stuck Open	11/13/14
TQ-FC-302-0107	Fort Calhoun Simulator Transient Testing (TR-2), Simultaneous Trip of All Main Feedwater Pumps	11/13/14
TQ-FC-302-0111	Fort Calhoun Simulator Transient Testing (TR-6), Turbine Trip From Maximum Power Level Does Not Result in Immediate Reactor Trip	11/13/14
TQ-FC-302-0116	Fort Calhoun Simulator Transient Testing (TR-11), Maximum Design Load Rejection	11/13/14

Test No.14.5.4.11	Loss of Load	12/19/12
RQCT-1423	Emergency Plan "High Intensity Training"	0
TQ-FC-302-0106	Fort Calhoun Simulator Transient Testing (TR-1), Manual Reactor Trip	11/13/14
TQ-FC-302-0108	Fort Calhoun Simulator Transient Testing (TR-3), Simultaneous Closure of Both Main Steam Isolation Valves	11/13/14
TQ-FC-302-0109	Fort Calhoun Simulator Transient Testing (TR-4), Simultaneous Trip of All Reactor Coolant Pumps	11/13/14
TQ-FC-302-0111	Fort Calhoun Simulator Transient Testing (TR-6), Turbine Trip from Maximum Power Level Does Not Result in Immediate Reactor Trip	11/13/14
TQ-FC-302-0112	Fort Calhoun Simulator Transient Testing (TR-7), Maximum Rate Power Ramp 100% to 75% to 100%	11/13/14
TQ-FC-302-0113	Fort Calhoun Simulator Transient Testing (TR-8), Maximum Size Reactor Coolant System Rupture Combined with a Loss of All Offsite Power	11/13/14
TQ-FC-302-0114	Fort Calhoun Simulator Transient Testing (TR-9), Maximum Size Unisolable Main Steam Line Rupture	11/13/14
TQ-FC-302-0116	Fort Calhoun Simulator Transient Testing (TR-11), Maximum Design Load Rejection	12/4/14
TQ-AA-306-F-20	Simulator Scenario Based Testing Checklist	1
USAR-14.9	Safety Analysis, Loss of Load	16

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2013-05070	2013-05570	2013-16631	2013-19712	2013-23048
2014-01221	2014-01671	2014-01672	2014-05108	2014-06086
2014-06806	2014-09826	2014-11190	2014-11379	2014-11493
2014-12933	2014-14104	2014-14105	2014-14148	2014-14190
2014-14208	2015-02547	2015-02548	2015-02573	2015-02575

Section 1R12: Maintenance Effectiveness

Procedures

Number	Title	Revision
ER-AA-310	Implementation of the Maintenance Rule	9
ER-AA-310-1002	Maintenance Rule Functions Safety Significance Classification	3
ER-AA-310-1003	Maintenance Rule-Performance Criteria Selection	4
ER-AA-310-1004	Maintenance Rule-Performance Monitoring	12
ER-AA-310-1005	Maintenance Rule-Dispositioning between (a)(1) and (a)(2)	7
ER-AA-310-1006	Maintenance Rule-Expert Panel Roles and Responsibilities	5
ER-AA-310-1001	Maintenance Rule Scoping	4

Miscellaneous Documents

Number	Title	Revision
	Charging System Health Report	
SDBD-CH-108	Chemical and Volume Control System Design Basis Document	29

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2014-13249	2014-4232	2014-9782	2015-2834

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures

Number	Title	Revision
ER-AA-600-1011	Risk Management Program	13
FCSG-19	Performing Risk Assessments	17
OP-PM-AFW-004	Third Auxiliary Feedwater Pump Operability Verification	39
OP-ST-DG-0001	Diesel Generator 1 Check	86
OP-ST-ESF-0009	Channel "A" Safety Injection, Containment Spray and Activation Signal Test	59a

Procedures

Number	Title	Revision
SO-G-23	Protected Equipment Program	8
SO-G-96	Planned LCO Entry Criteria and Equipment Reliability Control	15
WC-AA-101	On-line Work Control Process	23
WC-AA-104	Integrated Risk Management	21

Miscellaneous Documents

Number	Title	Revision
RG 1.160	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	39
NUMARC 93-01	Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	4A
	Work Schedule of Week Commencing 1/26/15	
	Equipment Out of Service Quantitative Risk Assessment Tool	
	Work Schedule of Week Commencing 2/9/15	
	Work Schedule of Week Commencing 2/23/15	
	Work Schedule of Week Commencing 3/2/15	
TAC No. MC8094	Fort Calhoun Station, Unit No. 1 – Issuance of Amendment RE: Inoperability of Snubbers	3/1/2006

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2015-02302	2015-3069	2015-3073

Section 1R15: Operability Determinations and Functionality Assessments

Procedures

Number	Title	Revision
OM-AA-101-1007	Outage Scope Control	9
OP-FC-108-115	Operability Determinations	0A
OP-FC-108-115-1002	Supplemental Consideration for On-shift immediate operability determinations	
QC-ST-HSS-0003	Visual Inspection of Bergen-Peterson Hydraulic Shock Suppressor (HSS) units in the Auxiliary Building	
SO-G-44	Snubber List and Operability Requirements	31

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2015-01489	2014-02197	2014-09179	2014-09036	2015-01766
2015-02235	2014-06387	2014-01659	2015-2033	2015-02760
2015-02762				

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540193	540791	545745-01	455157	526415

Miscellaneous Documents

Number	Title	Revision Date
USAR-8.4	Electrical Systems, Emergency Power Sources	16
P21-01072015	Report of a Potential Failure of SBM type switches per 10CFR Part 21	12/8/14
USAR 8.4	Emergency Power Sources	18
DBD-DG-112	Emergency Diesel Generators	31
USAR-9.10	Heating, Ventilation, and Air Conditioning Systems	30
USAR	Appendix A, Quality Assurance Program	41
NO-FC-10	Quality Assurance Topical Report	2
PBD-G	Snubbers	24

Number	Title	Revision Date
USAR Appendix F	Classification of Structures and Equipment and Seismic Criteria	10
Calculation EA10-025	Cycle 27, Plant Database	0
Calculation FC07077	Safety Injection phase system performance for safety injection and containment spray systems	2

Drawings

Number	Title	Revision
185	Sh. 1 Simplified One Line Diagram Plant Electrical System P&ID	45
7668	Sh.1 Seismic Restraint FWS-5A	2
D-4621	Sh.1, CQE Piping Isometrics Sub SYS FW-342A	1
10459	Sh.1 Steam Generator Feedwater and Blowdown, Flow Diagram P&ID	99

Section 1R18: Plant Modifications

Procedures

Number	Title	Revision
PED-GEI-35	Preparation of Minor Configuration Changes	75
PED-OP-2	Configuration Change Control	63

Miscellaneous Documents

Number	Title	Revision
EC64929	Install Gas Accumulation Tank on CCW Pump AC-3A	
FC-1137	Acceptance for Operability for EC 64929	21

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Number	Title	Revision
13007.43-BZ-IB	Fire Protection System Pipe Supports	
13007.43-BZ-IN	Pipe Support Assembly for 5" Fire Support System	

Section 1R19: Post-Maintenance Testing

Procedures

Number	Title	Revision
EM-RR-EX-1606	Replacement of Potter and Brumfeld Relays and Sockets	3
IC-RR-VX-0505	Sumner Solenoid Valve Replacement	3
MA-AA-013	Post Maintenance Testing Selection Instructions	22
MA-AA-716-012	Post Maintenance Testing	19
MA-AA-716-100	Maintenance Alterations Process	12
MM-RR-CH-0001	Inspection and Repair of Charging Pump Hydraulic Section	10
OF-ST-CH-3003A	Chemical and Volume Control System Pump Check	4
OP-ST-DG-0001	Diesel Generator 1 check	86
OP-ST-DG-0002	Diesel Generator 2 check	77
PED-EWP-11.1	Two Wire butt Splice Installation	6
PED-GEI-55.7	Documentation for an ASME Section XI-1998/200A Repair/replacement Activity	613
SE-ST-SA-3002	DG-1 Air Start System Forty Month Inservice Pressure Test	7
SO-M-103	System Chemicals	15
SP-CP-08-D1-IAC	Calibration of the Time Overcurrent Relays for Diesel Generator number 1 (50/51/D1)	6
SP-CP-08-D1-GGP	Calibration of the Power Directional Relay for Diesel Generator Number 1 (67/D1)	7
SP-CP-08-D1-TC	Functional Checkout of the Protective Relays in Diesel Generator Number 1 Circuit	15a
SPP-7.1	Weld and Base Metal Repair	4

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2010-6372	2015-1518	2015-1489	2015-00685	2014-14624
2015-00330	2015-02010			

Drawings

Number	Title	Revision
11405-E-27	Sheet 8464, Wiring Diagram Diesel #1 Dampers YCV-871G, H, E	11

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316263-07	316263-09	442264-01	442264-03	519750
521958	00523409	00523410	525978-01	0541924-01
00543176	544054-01			

Drawings

Number	Title	Revision
B120F145015	Engine Control Panel and Wiring Diagram,	
09313133-SDFP-1	Schematic Diagram for Field Flashing Control for Replacement Diesel Generator Excitation/voltage Regulator Panels: AI-133A for DG-1and AI-133B for DG-2	2
B120F14501	Sh. 2, Schematic Engine Control	28
E-23866-210-130	Safety injection and Containment Spray System Flow Diagram	33
11405-M-5	Demineralized Water System Flow Diagram	24
FD 280337/9	File 876, High Pressure Safety Injection	6

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Number	Title	Revision Date
USAR 8.4	Emergency Power Sources	16
RG 1.9	Diesel Generator Testing	4
USAR-9.2	Chemical and Volume Control System	27
SDBD-DG-112	Emergency Diesel Generators	31
USAR Appendix N	Reclassification of Systems	10
EC42846	Replace DG-1 and DG-2 Starting Receivers	8/11/10

Section 1R22: Surveillance Testing

Procedures

Number	Title	Revision
IC-ST-SA-3001B	Diesel generator #2 Starting Air Compressors Discharge Check Valves Exercise Test	4
OP-ST-AFW-3011	Auxiliary Feedwater Pump FW-10,Steam Isolation Valve and Check Valve Tests	22
OP-ST-CCW-3012	AC-3B Component Cooling Water Pump In-Service Test (IST)	21
OP-ST-DG-0002	Emergency Diesel Generator #2 Check	77
SE-ST-HSS-0001	Hydraulic Shock Suppressor Life Monitoring	6
SO-G-23	Surveillance Test Program	66
SO-G-44	Snubber List and Operability Requirements	31

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Number	Title	Revision
USAR 8.4	Emergency Power Sources	16
TS 2.7.(2)j	Emergency Diesel Generators	
TS 2.3	Emergency Core Cooling	
TS 2.4	Containment Cooling	
TS 3.7	Emergency Power system periodic tests	264
USAR Appendix A	Quality Assurance Program	41
NO-FL-10	Quality Assurance Topical Report	2
PBD-6	Snubbers	24
FC05748	Outside containment FW piping class stress check and support reactions	0

Work Orders

316173-01	471074-01	523447-01

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2012-00803	2013-00803	2013-082728	2015-02760	2015-02762
2015-02837	2015-02808	2015-02857		

Drawings

Number	Title	Revision
7668	Sh. 1, Seismic Restraint FWS-5A	2
D-4621	Sh. 1, CQE Piping Isometrics Sub Sys FW-342A	1
10459	Sh. 1, Steam Generator Feedwater and Blowdown, Flow Diagram P & ID	9

Section 1EP6: Drill Evaluation

Procedures

Number	Title	Revision
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ARP-OCS-FW	Feedwater Annunciator Response Procedure	7
AOP-9	High Radioactivity	11
AOP-22	Reactor coolant leak	34
EPIP-EOP-7	Protective actions guidelines	26
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Number	Title	Revision Date
LIC-15-0001	Licensee Event Report 2014-007	2/13/15
LIC-15-0012	Licensee Event Report 2014-004	1/21/15

LIC-15-0019	Licensee Event Report 2013-0014	1/29/15
LIC-14-0006	Licensee Event Report 2013-019	1/31/15

Section 4OA2: Problem Identification and Resolution

Condition Reports

2014-07833	2014-08639	2014-14941	2014-15143

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Number	Title	Revision Date
	Enercon Project Report for Fort Calhoun Station Component Cooling Water System Gas Accumulation Evaluation	
	System engineering action Recommendations for Mitigating CCW Pump as Void Inoperability Omaha Public Power District, Fort Calhoun Station (FCS), Response to NRC Generic Letter 2008-01	10/14/08
LIC-10-0062	Response to NRC Request for Status of Corrective Actions Contained in OPD Response to 2008-01	5/23/11

Section 4OA3: Follow-up of Events and Notices of Enforcement Discretion

Procedures

Number	Title	Revision
QC-ST-HSS-0003	Visual inspection of Berger-Patterson Hydraulic Shock Suppressor Units in the Auxiliary Building	2
SO-G-44	Snubber List and Operability Requirements	31

Drawings

Number	Title	Revision
7668	Sheet 1 and 2, 120 Volt AC Instrument Buses	
10459	Sheet 1 Steam Generator Feedwater and Blowdown, Flow Diagram	99

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2013-19107	2013-20057	2013-21519	2013-08095	2013-03381
2013-15336				

Miscellaneous

Number	Title	Revision
Figure 8.1.1	Simplified Diagram Plant Electrical System P&ID	145
PBD-6	Snubbers	24
EC62826	Add Fuses to the DC Anmeter Circuitry for Anmeters	