UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
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April 10, 2015

Mr. Raymond Lieb
Site Vice President
FirstEnergy Nuclear Operating Co.
Davis-Besse Nuclear Power Station
5501 N. State Rte. 2, Mail Stop A–DB–3080
Oak Harbor, OH 43449-9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION, COMPONENT DESIGN BASES INSPECTION REPORT 05000346/2015008

Dear Mr. Lieb:

On February 27, 2015, the U.S. Nuclear Regulatory Commission (NRC) completed a Component Design Bases Inspection inspection at your Davis-Besse Nuclear Power Station. The enclosed report documents the results of this inspection, which were discussed on February 27, 2015, with yourself, and other members of your staff.

Based on the results of this inspection, five NRC-identified findings of very-low safety significance were identified. The findings involved violations of NRC requirements. However, because of their very-low safety significance, and because the issues were entered into your Corrective Action Program, the NRC is treating the issues as Non-Cited Violations (NCVs) in accordance with Section 2.3.2, of the NRC Enforcement Policy.

If you contest the subject or severity of the NCV, 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 III; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Davis-Besse Nuclear Power Station.

In addition, if you disagree with the cross-cutting aspect assigned to any finding 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 III, and the NRC Resident Inspector at the Davis-Besse Nuclear Power Station.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 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’s Public Document Room or from the Publicly Available Records (PARS) component of the NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Christine A. Lipa, Chief
Engineering Branch 2
Division of Reactor Safety

Docket No. 50–346
License No. NPF–3

Enclosure:
Inspection Report 05000346/2015008;
w/Attachment: Supplemental Information

cc w/encl: Distribution via LISTSERV®

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50–346
License No:	NPF–3
Report No:	05000346/2015008
Licensee:	FirstEnergy Nuclear Operating Company
Facility:	Davis-Besse Nuclear Power Station
Location:	Oak Harbor, OH
Dates:	January 26 through February 27, 2015
Inspectors:	A. Dunlop, Senior Engineering Inspector, Lead S. Sheldon, Senior Engineering Inspector, Electrical G. O'Dwyer, Engineering Inspector, Mechanical N. Valos, Senior Reactor Analyst S. Kobylarz, Electrical Contractor C. Baron, Mechanical Contractor
Approved by:	Christine A. Lipa, Chief Engineering Branch 2 Division of Reactor Safety

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SUMMARY OF FINDINGS

Inspection Report 05000346/20008, 01/26/2015 – 02/27/2015; Davis-Besse Nuclear Power Station; Component Design Bases Inspection.

The inspection was a 3-week onsite baseline inspection that focused on the design of components. The inspection was conducted by regional engineering inspectors and two consultants. Five Green findings were identified by the inspectors. The findings were also considered Non-Cited Violations (NCVs) of U.S. Nuclear Regulatory Commission (NRC) regulations. The significance of inspection findings is indicated by their color (i.e., greater than Green, Green, White, Yellow, or Red), and determined using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process (SDP)" dated June 2, 2011. Cross-cutting aspects are determined using IMC 0310, "Aspects Within the Cross-Cutting Areas" effective date January 1, 2014. 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 inspectors identified a finding of very-low safety significance (Green), and an associated NCV of Title 10, Code of Federal Regulations (CFR) Part 50, Appendix B, Criterion III, "Design Control," for the failure to assure that applicable regulatory requirements, and the design basis were correctly translated into specifications, drawings, procedures, and instructions, and verifying the adequacy of design. Specifically, the licensee failed to verify the adequacy of procedures controlling alignment of non-essential busses to the emergency diesel generators during a design basis event. The procedure's guidance could put the plant in an unanalyzed alignment with the potential to result in the failure of safety-related equipment. The licensee entered this finding into their Corrective Action Program (CAP), and initiated a Standing Order to preclude the unanalyzed alignment.

  The finding was determined to be more than minor because the finding was associated with the Mitigating Systems cornerstone attribute of Procedure Quality, and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding screened as very low-safety significance (Green) because the inspectors were able to answer "Yes" to screening question A1 because the finding represents a design deficiency confirmed not to result in loss of operability. The inspectors did not identify a cross-cutting aspect associated with this finding as it did not reflect current performance. (Section 1R21.3.b.(1))

- Green. The inspectors identified a finding of very-low safety significance (Green), and an associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to demonstrate compliance with Institute of Electrical and Electronics Engineers (IEEE) 308-1971, "IEEE Standard Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations," for the required independence of essential safety-related inverter distribution system channels. Specifically, a common mode failure due to inadequate fault protection on several outside distribution panels could cause the loss of redundant safety-related inverters. The licensee entered this finding into their CAP, and initiated a Standing Order that would identify the affected circuit breakers and require they be opened based on a tornado warning that could potentially affect the site.

The finding was determined to be more than minor because it was associated with the Mitigating System cornerstone attribute of Protection Against External Factors, and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding screened as of very-low safety significance (Green) based on a SDP Phase II analysis that determined a conservative delta core damage frequency (ΔCDF) of 1.29E-7/yr. The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance. (Section 1R21.3.b.(2))

- Green. The inspectors identified a finding of very-low safety significance (Green), and an associated NCV of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," for the licensee failing to incorporate the design requirements and acceptance limits into test procedures. Specifically, the design limit for the minimum voltage on essential safety-related inverter YV1 bus during Modes 5 and 6 was not correctly incorporated into surveillance test procedures. The licensee entered this finding into their CAP and re-analyzed for the 116 volts as-found value at panel Y1, and determined the loads that did not have adequate rated voltage at the surveillance procedure minimum acceptance criteria voltage, 114 volts, would have had sufficient voltage to operate at the as-found measured voltage to perform their intended safety function.

  The finding was determined to be more than minor because it was associated with the Mitigating System cornerstone attribute of Procedure Quality and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding screened as of very-low safety significance (Green) because the deficiency was confirmed not to result in a loss of safety function. The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance. (Section 1R21.3.b.(3))

- Green. The inspectors identified a finding of very-low safety significance (Green), and an associated NCV of Technical Specification (TS) Surveillance Requirement 3.7.7.1, for the licensee's failure to verify several component cooling water (CCW) system manual valves in the flow path servicing safety-related equipment that were not locked, sealed, or otherwise secured, were in the correct position every 31 days. Specifically, the unsecured CCW pump seal water flush isolation valves (two valves per pump) for the two required operable CCW pumps were not verified open every 31 days. The licensee entered this finding into their CAP, verified the correct position of the valves, and planned to revise the Locked Valve Program to include the requirement to have the valves in the locked open position.

  The finding was determined to be more than minor because it was similar to IMC 0612, Appendix E, Example 3.c, because more than one valve was in the required position, but not locked, sealed, or otherwise secured in the correct position, and because it was associated with the Mitigating Systems cornerstone's attribute of Configuration Control, and affected the cornerstone's objective of ensuring the availability, reliability, and capability of systems to respond to initiating events to prevent undesirable consequences. The finding screened as of very-low safety significance (Green) because the deficiency was confirmed not to result in a loss of safety function. The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance. (Section 1R21.3.b.(4))

- Green. The inspectors identified a finding of very-low safety significance (Green), and an associated NCV of TS 3.5.4, "Borated Water Storage Tank (BWST)," for the failure to comply with the limiting condition for operation (LCO) while the BWST was aligned to the non-seismic spent fuel pool purification system, causing the BWST to be inoperable based on no longer meeting the tank's seismic requirement. The licensee entered this finding into their CAP, and initiated an LCO Tracking Log entry to not place the BWST on spent fuel pool purification in Modes 1 through 4.

  The finding was determined to be more than minor because it was associated with the Mitigating Systems attribute of Protection Against External Factors, and affected the cornerstone's objective of ensuring the availability, reliability, and capability of systems to respond to initiating events to prevent undesirable consequences. The finding resulted in the loss of system safety function (i.e., the BWST). The finding screened as of very-low safety significance (Green) based on a SDP Phase II analysis that determined the ΔCDF for the finding was 1.17E-7/yr. This finding has a cross-cutting aspect in the area of Problem Identification and Resolution because recent operating experience was not adequately evaluated to prevent the identified finding. (Section 1R21.4.b.(1)) [P.5, Operating Experience]

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R21 Component Design Bases Inspection (71111.21)

.1 Introduction

The objective of the component design bases inspection is to verify that design bases have been correctly implemented for the selected risk-significant components, and that operating procedures and operator actions are consistent with design and licensing bases. As plants age, their design bases may be difficult to determine, and an important design feature may be altered or disabled during a modification. The Probabilistic Risk Assessment (PRA) model assumes the capability of safety systems and components to perform their intended safety function successfully. This inspectable area verifies aspects of the Initiating Events, Mitigating Systems, and Barrier Integrity cornerstones for which there are no indicators to measure performance.

Specific documents reviewed during the inspection are listed in the Attachment to the report.

.2 Inspection Sample Selection Process

The inspectors selected risk-significant components and operator actions for review using information contained in the licensee's PRA, and the Davis-Besse Nuclear Power Station Standardized Plant Analysis Risk Model. In general, the selection was based upon the components and operator actions having a risk achievement worth of greater than 1.3 and/or a risk reduction worth greater than 1.005; and components with large early release frequency (LERF) implications. The operator actions or operating procedures selected for review included actions taken by operators both inside and outside of the control room during postulated accident scenarios that were associated with the selected components. In addition, the inspectors selected operating experience issues associated with the selected components.

The inspectors performed a margin assessment and detailed review of the selected risk-significant components to verify that the design bases have been correctly implemented and maintained. This design margin assessment considered original design reductions caused by design modification, or power uprates, or reductions due to degraded material condition. Equipment reliability issues were also considered in the selection of components for detailed review. These included items such as performance test results, significant corrective action, repeated maintenance activities, Maintenance Rule (a)(1) status, components requiring an operability evaluation, U.S. Nuclear Regulatory Commission (NRC) resident inspector input of problem areas/equipment, and system health reports. Consideration was also given to the uniqueness and complexity of the design, operating experience, and the available defense in depth margins. A summary of the reviews performed and the specific inspection findings identified are included in the following sections of the report.

This inspection constituted 19 samples (15 components, 1 component with LERF implications, and 3 operating experience) as defined in Inspection Procedure 71111.21-05.

.3 Component Design

a. Inspection Scope

The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR), Technical Specifications (TSs), design basis documents, drawings, calculations and other available design basis information, to determine the performance requirements of the selected components. The inspectors used applicable industry standards, such as the American Society of Mechanical Engineers Code, Institute of Electrical and Electronics Engineers (IEEE) Standards, and the National Electric Code to evaluate acceptability of the systems' design. The NRC also evaluated licensee actions, if any, taken in response to NRC issued operating experience, such as Bulletins, Generic Letters, Regulatory Issue Summaries, and Information Notices (INs). The review was to verify that the selected components would function as designed when required and support proper operation of the associated systems. The attributes that were needed for a component to perform its required function included process medium, energy sources, control systems, operator actions, and heat removal. The attributes to verify that the component condition and tested capability was consistent with the design bases, and was appropriate may include installed configuration, system operation, detailed design, system testing, equipment and environmental qualification, equipment protection, component inputs and outputs, operating experience, and component degradation.

For each of the components selected, the inspectors reviewed the maintenance history, preventive maintenance activities, system health reports, operating experience-related information, vendor manuals, electrical and mechanical drawings, and licensee Corrective Action Program (CAP) documents. Field walkdowns were conducted for all accessible components to assess material condition, including age-related degradation, and to verify that the as-built condition was consistent with the design. Other attributes reviewed are included as part of the scope for each individual component.

The following 16 components (samples) were reviewed:

- Emergency Diesel Generator (1-1), including Air Start and Jacket Water Systems: The inspectors performed a limited review of the design of the emergency diesel generator (EDG) to verify the loading was within its rating. The inspectors reviewed calculations and surveillance records relating to EDG loading. Battery voltage calculations were reviewed to ensure that adequate voltage would be available to control the EDG output breaker. The inspectors reviewed the function of the air start and jacket water cooling systems during EDG operation, as well as the interface with other support systems. Specifically, the inspectors reviewed analyses related to the capability of the jacket water heat exchangers to remove the required heat load under all operating conditions, and within the temperature limits of the jacket water. The inspectors reviewed the pressure limits, and allowable leakage limits of the air start receivers.

- 4.16 kV Essential Switchgear Bus (C1): The inspectors reviewed load flow calculations to determine whether the 4.16 kV safety buses had sufficient capacity to support their required loads under worst case accident loading, and grid voltage conditions. Short circuit calculations were reviewed to ensure breakers were adequately sized. The inspectors reviewed elementary wiring diagrams for bus feeder and load breakers to determine whether system control logic was consistent with system design requirements stated in the UFSAR. Bus and load protective relaying were reviewed to determine whether it provided

adequate protection to the buses, and whether there would be any adverse interactions within the protection scheme that would reduce system reliability. The inspectors reviewed calibration procedures and records for undervoltage relays to determine whether the relays were maintained as required, and whether there were any adverse performance trends.

- 125 Vdc Distribution Panel (D1P): Design calculations and vendor documents were reviewed to verify adequate panel short circuit ratings, branch circuit fuse sizing, and fuse coordination. One-line diagrams and design basis documents for the electrical distribution system were reviewed to identify requirements and interfaces. Preventive maintenance thermography was reviewed for fuses and disconnect switches to verify component reliability was adequately maintained.

- 120 Vac Essential Inverter (YV1): Design calculations were reviewed to verify adequate inverter sizing, fuse coordination, and voltage available to loads. One-line diagrams and design basis documents for the inverter electrical distribution system were reviewed to identify design requirements and interfaces. Preventive maintenance activities were reviewed to verify the inverter system was maintained according to manufacturer recommendations. Alarm response procedures were reviewed for monitored conditions and operator response. Past modifications associated with the inverter were reviewed to verify adequacy for design basis considerations.

- 120 Vac Essential Instrumentation Distribution Panel (Y1): Design calculations and vendor documents were reviewed to verify adequate panel short circuit ratings and branch circuit fuse sizing and coordination. One-line diagrams and design basis documents for the electrical distribution system were reviewed to identify requirements and interfaces. Preventive maintenance was reviewed for fuses and disconnect switches to verify component reliability was adequately maintained.

- High Pressure Injection Pump (P58-2): The inspectors reviewed the design basis of the high pressure injection pump and motor including performance requirements, net positive suction head (NPSH) requirements, and electrical power requirements. The inspectors reviewed the function of the pump during postulated small and large break loss-of-coolant-accidents (LOCAs) including required minimum flow and runout limits. The inspectors also reviewed emergency operating procedures associated with aligning the pump suction for post-LOCA recirculation operation, including postulated single failures. The inspectors reviewed leakage limits associated with the pump minimum flow isolation valves. Surveillance test procedures and recent test results were reviewed to verify acceptance criteria were met and performance degradation would be identified. The inspectors reviewed elementary wiring diagrams for the supply breaker to determine whether system control logic was consistent with system design requirements stated in the UFSAR. The inspectors reviewed cable ampacity, voltage drop, and breaker protective relaying calculations to verify the pump would operate during anticipated conditions.

Decay Heat Removal Pump (P42-1): The inspectors reviewed the design basis of the decay heat pump and motor including performance requirements, NPSH requirements, and electrical power requirements. The inspectors reviewed the function of the pump during postulated small and large break LOCAs including required minimum flow and runout limits. The inspectors reviewed emergency operating procedures associated with aligning the pump suction for post-LOCA recirculation operation from the emergency containment sump, including postulated single failures. The inspectors also reviewed setpoints associated with the transfer of the pump suction from the borated water storage tank (BWST) to the emergency containment sump. Surveillance test procedures and recent test results were reviewed to verify acceptance criteria were met, and performance degradation would be identified. The inspectors reviewed elementary wiring diagrams for the supply breaker to determine whether system control logic was consistent with system design requirements stated in the UFSAR. The inspectors reviewed cable ampacity, voltage drop, and breaker protective relaying calculations to verify the pump would operate during anticipated conditions.

- Decay Heat Cooler 1-1 (E27-1): The inspectors reviewed various calculations related to the thermal performance of the heat exchanger under design basis accident and transient conditions, including conditions with maximum system heat load, maximum service water system supply temperature, and minimum service water flow to the heat exchanger. The inspectors also reviewed performance test results and analysis of the test result data to verify the analyzed performance would be bounded by the as-found conditions.

- Pressurizer Power-Operated Relief Valve (RC-2A): The inspectors reviewed the design basis of the pressurizer power-operated relief valve (PORV), including requirements for the valve to operate under postulated transient and accident conditions. This review included the capacity of the PORV to open and close under the most limiting design conditions. The inspectors reviewed vendor information and test procedures as well as the results of recent shop tests to verify acceptance criteria were met and performance degradation would be identified. Design drawings and vendor documents were reviewed to verify the installed configurations would support the design basis function under accident conditions and had been maintained to be consistent with design assumptions. Design calculations and vendor documents were reviewed to verify the solenoid valve had sufficient voltage to operate during limiting design basis conditions, and that circuit protective devices were adequately sized.

- Containment Emergency Sump: The inspectors reviewed the design basis of the containment emergency sump including requirements for the sump and associated screens to support operation of the emergency core cooling system (ECCS) under accident conditions. The inspectors reviewed NPSH analyses for post-accident ECCS recirculation operation. The inspectors reviewed procedures and recent inspection results for periodic visual inspection of the sump and screens, including as-found photographs of the emergency sump and associated screens.

- <u>Containment Vacuum Relief Check Valve (CV5080)</u>: The inspectors reviewed the design basis of the containment vacuum relief isolation check valve located outside of the containment. Test procedures and recent test results (e.g., local leak rate testing) were reviewed to verify the acceptance criteria for tested parameters were supported by calculations or other engineering documents. The tests and analyses verified the valve would function as required during accident and transient conditions.

- <u>Containment Air Cooler (E37-1)</u>: Inspectors reviewed calculations and procedures to verify the design bases and design assumptions were appropriately translated into applicable documents. The inspectors reviewed calculations related to the thermal performance of the containment air cooler (CAC) under design basis accident and transient conditions, including conditions with maximum system heat load, maximum service water system supply temperature, maximum allowed tube plugging, and minimum service water flow to the CAC. Test procedures and recent test results were reviewed to verify the acceptance criteria for tested parameters were supported by calculations or other engineering documents, and the tests and analyses validated component operation under accident and transient conditions. The inspectors also reviewed performance test results and analysis of the test result data to verify the analyzed performance would be bounded by the as-found conditions. The inspectors reviewed eddy current testing and cleaning/inspection reports to verify material condition of the CAC tubes. The inspectors also reviewed the current required by the fan motors under normal and degraded voltage conditions, and the over current trip setpoint to verify the fans would operate during a design basis event.

- <u>Auxiliary Feedwater Pump Turbine Steam Admission Valve (MS5889B)</u>: The inspectors reviewed the air-operated valve calculations, including required thrust, weak link, and maximum differential pressure, to ensure the valve would be capable of functioning under design and licensing bases conditions. Diagnostic and inservice test results were reviewed to verify acceptance criteria were met and performance degradation would be identified. Design drawings and vendor documents were reviewed to verify the installed configurations would support the design basis function under accident conditions, and had been maintained to be consistent with design assumptions and environmental qualification requirements. Design calculations and vendor documents were reviewed to verify the solenoid valve had sufficient voltage to operate during limiting design basis conditions, and that circuit protective devices were adequately sized.

- <u>Main Steam to Auxiliary Feedwater Pump Turbine Line Block Valve (MS106A)</u>: The inspectors reviewed the motor-operated valve (MOV) calculations, including required thrust, weak link, and maximum differential pressure, to ensure the valve would be capable of functioning under design and licensing bases conditions. Diagnostic and inservice results were reviewed to verify acceptance criteria were met and performance degradation would be identified. The inspectors reviewed voltage drop calculations to ensure adequate power would be available. The inspectors reviewed valve control logic and thermal overload settings.

- Component Cooling Water Pump (P43-1): Inspectors reviewed calculations and procedures to verify the design bases and design assumptions were appropriately translated into these documents. Design and operational requirements were reviewed with respect to pump flow rate, developed head, NPSH, minimum flow requirements, and the pump capability to provide the flow rate required to remove the assigned heat loads. The inspectors reviewed the pump's protection from the formation of air vortices. Test procedures and recent test results were reviewed to verify the acceptance criteria for tested parameters were supported by calculations or other engineering documents and the tests and analyses validated component operation under accident and transient conditions. The inspectors also reviewed performance test results and analysis of the test result data to verify the analyzed performance would be bounded by the as-found conditions. The inspectors reviewed elementary wiring diagrams for the supply breaker to determine whether system control logic was consistent with system design requirements stated in the UFSAR. The inspectors reviewed cable ampacity, voltage drop, and breaker protective relaying calculations to verify the pump would operate during anticipated conditions.

- Component Cooling Water to Non-Safety-Related Loads Isolation Valve (CC1495): Inspectors reviewed calculations to verify the design bases and design assumptions were appropriately incorporated into the calculations. The inspectors reviewed tests (e.g., diagnostic testing) to ensure the licensee has correctly translated the design basis into test procedures used to verify valve performance. The inspectors also reviewed performance test results and analysis of the test result data to verify the analyzed performance would be bounded by the as-found conditions. Design drawings and vendor documents were reviewed to verify the installed configurations would support the design basis function under accident conditions, and was maintained to be consistent with design assumptions. Design calculations and vendor documents were reviewed to verify the solenoid valve had sufficient voltage to operate during limiting design basis conditions, and that circuit protective devices were adequately sized.

b. Findings

(1) Vulnerability of Emergency Diesel Generator Crosstie to a Non-Essential Bus

Introduction: The inspectors identified a finding of very-low safety significance (Green), and an associated Non-Cited Violation (NCV) of Title 10, *Code of Federal Regulations* (CFR) Part 50, Appendix B, Criterion III, "Design Control," for the failure to assure applicable regulatory requirements and design basis were correctly translated into specifications, drawings, procedures, and instructions, and verifying the adequacy of design. Specifically, the licensee failed to verify the adequacy of procedures controlling alignment of non-essential busses to the EDGs during a design basis event.

Description: The inspectors reviewed protective relay setting on the safety-related 4.16 kV bus C1, and observed that the EDG output breaker protection did not coordinate with bus tie breakers to non-essential (non-safety-related) busses C2 and D2. The inspectors also noted Abnormal Procedure DB-OP-02521, "Loss of AC Bus Power Sources," provided steps for aligning the non-essential busses to the EDG powered essential busses during a design basis event. No cautions or limitations were provided specifying limited conditions for the alternate alignment.

The inspectors reviewed calculation C-EE-024.01-002, “Protective Relay Setpoint for the EDG 1-1 (AC101),” which stated the voltage controlled overcurrent relay would operate on a fault of the non-essential bus. The licensee confirmed that if the essential bus was aligned to the non-essential bus, a fault on the nonessential (non-safety-related) bus would cause the EDG output breaker to trip open. This would be followed by opening of the bus tie breakers, reclosing of the EDG output breaker, and sequencing the required essential loads back on to the EDG. The licensee could provide no analyses demonstrating the resulting sequence of events did not have an adverse impact on the safety function of the safety-related equipment. Inspectors were concerned that a failure on the non-essential bus may prevent a safety system from meeting its minimum performance requirements due to system transients or unacceptable configurations.

In response, the licensee initiated condition report (CR) 2015-02476 to document the condition, and evaluate potential effects of the unanalyzed condition. Immediate corrective actions included issuing a Standing Order to preclude the unanalyzed alignment. Additional follow-up actions will provide final resolution for this issue.

Analysis: The inspectors determined the failure to demonstrate the adequacy of cross-tying safety-related bus C1 to non-essential bus D2 during a design basis event was contrary to 10 CFR Part 50, Appendix B, Criterion III, and was a performance deficiency. The finding was determined to be more than minor because the finding was associated with the Mitigating Systems cornerstone attribute of Procedure Quality, and 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 provided procedural guidance in DB-OP-02521 that would put the plant in an unanalyzed alignment with the potential to result in the failure of safety-related equipment.

The inspectors determined the finding could be evaluated using the Significance Determination Process (SDP) in accordance with Inspection Manual Chapter (IMC) 0609, “Significance Determination Process,” Attachment 0609.04, “Phase 1-Initial Screening and Characterization of Findings,” for the Mitigating Systems cornerstone. The inspectors evaluated the finding using Appendix A, “The Significance Determination Process for Findings At-Power.” The finding screened as very-low safety significance (Green) because the inspectors were able to answer “Yes” to screening Question A1 in Exhibit 2, because the finding represents a design deficiency confirmed not to result in loss of operability.

The inspectors did not identify a cross-cutting aspect associated with this finding as it did not reflect current performance.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” requires, in part, “Measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. The 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.”

Contrary to the above, as of February 26, 2015, the licensee failed to assure the applicable regulatory requirements and the design basis were correctly translated into specifications, drawings, procedures, and instructions, and verifying the adequacy of design. Specifically, the licensee failed to verify the adequacy of procedures controlling alignment of nonessential busses to the essential busses during a design basis event.

Because this violation was of very-low safety significance, and it was entered into the licensee's CAP as CR 2015-02476, initiated a Standing Order to preclude the unanalyzed alignment, this violation is being treated as an NCV, consistent with Section 2.3.2, of the NRC Enforcement Policy. (NCV 05000346/2015008-01, Vulnerability of EDG Cross-Tie to a Nonessential Bus)

(2) Failure to Comply with Institute of Electrical and Electronics Engineers 308-1971 for the Required Independence of Safety-Related Essential Inverter Distribution Systems

Introduction: The inspectors identified a finding of very-low safety significance (Green), and an associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to demonstrate compliance with IEEE 308-1971, "IEEE Standard Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations," for the required independence of essential safety-related inverter distribution system channels. Specifically, a common mode failure due to inadequate fault protection on several outside distribution panels could cause the loss of redundant safety-related inverters.

Description: The UFSAR Section 1.2.5.1, "Principal Design Criteria," Part c., stated, "All electrical systems and associated equipment important to safety are classified as IEEE-308 Class 1E and are designed to ensure that any design basis event does not cause:

- A loss of electric power to a number of engineered safety features sufficient to jeopardize the safety of the station; or
- A loss of electrical power to equipment that could cause significant damage to the fuel or reactor coolant system."

The UFSAR Section 1.2.5.1, Part b., stated, "Sufficient, reliable, redundant, adequate, and independent power sources are provided for handling all normal and emergency conditions."

The IEEE 308-1971, Section 3.14, defined "independence" as "No common failure mode for any design basis event." Section 3.8 defined "design basis events" as "Postulated events used in the design to establish the performance requirements of the structures and systems," and Section 3.12, defined "common failure mode" as "A mechanism by which a single design basis event can cause redundant equipment to be inoperable." Section 5.4, "Vital Instrumentation and Control Power System," Section 5.4.2," Design Requirements," stated for the power supply requirements, that, "However, power must be supplied to preserve their reliability, independence, and redundancy." IEEE 308, Table 1, Design Basis Events, Natural Phenomena, includes Earthquake, Wind, Hurricane, Tornado, etc., and Postulated Phenomena, including postulated loss of preferred power supply combined with any of the above, and also various single equipment failures, component malfunctions, and maintenance outage.

The inspectors found that the essential inverters, in each of the four safety-related essential instrumentation power channels, provided an uninterruptible 120 Vac power supply to the engineered safety features actuation system (SFAS) and reactor protection system (RPS) equipment, in addition to providing vital power to other safety-related systems and equipment in the power channel. The inspectors found the inverter itself was not capable of interrupting faults on its' output in all cases, and therefore relied on an automatic transfer to a controlled voltage transformer (CVT) source that was designed and sized to provide sufficient current to operate fuses that protect circuits from faulted conditions. However, during a design basis loss of offsite power (LOOP) event, the CVT source would not be available during the time period when the LOOP occurred, and prior to the EDG supplying standby power to the Class 1E electric power system.

Specifically, if the CVT was not available, the inspectors determined for the essential distribution panel branch circuits protected by a 30-ampere fuse, the inverter may not be capable of providing sufficient current to operate the fuse on a faulted circuit. A faulted circuit condition could result during a design basis event, such as from the effects of a postulated single failure. Also, a faulted circuit condition could result during a design basis event, such as from the effects of a tornado missile.

If the CVT was not available to supply sufficient fault current, the inverter could go into a current limiting condition when providing current to a faulted circuit. The "current limiting condition" is an inherent protection feature of the inverter, whereby the voltage output of the inverter collapses as a result of a current overload condition that is above the inverter rated output capability. For the postulated condition, when the inverter output collapses due to a fault on its output, all vital instrumentation powered by the inverter channel would de-energized.

The inspectors found during a walkdown near the BWST, that multiple safety-related Class 1E freeze protection heat tracing panels were located either adjacent to or in close proximity to each other, and that the freeze protection panels were powered from multiple essential instrumentation channels. The inspectors were concerned a single tornado missile could impact multiple heat tracing panels potentially causing faulted circuit conditions on multiple train essential distribution panel branch circuits.

Since each of the subject Class 1E heat tracing panels was protected by an individual 30-ampere fuse in their respective essential distribution panel, when the CVT was not available immediately after a LOOP, each essential inverter in the affected channel(s) may be lost when providing current to operate the fuse during a faulted circuit condition due to the inherent current limiting aspect of the inverter.

The inspectors determined the potential to lose multiple essential inverter power supplies, due to the effects of a single design basis event tornado missile, did not meet the independence requirements of IEEE 308, and could result in the common mode failure of multiple essential inverters.

The licensee initiated CR 2015-01862 to evaluate the condition. Immediate corrective actions included issuing a Standing Order to identify the affected circuit breakers and require they be opened based on a tornado warning for Ottawa County. The inspectors considered these actions were acceptable to ensure the independence of the essential inverters.

Analysis: The inspectors determined the failure to conform to the independence requirements of IEEE 308-1971 was contrary to 10 CFR Part 50, Appendix B, Criterion III, and was a performance deficiency. The finding was determined to be more than minor because the finding was associated with the Mitigating Systems cornerstone attribute of Protection Against External Factors, and affected the cornerstone objective of ensuring the 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 vital 120 Vac distribution system channels for conditions involving postulated tornado generated missiles.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1-Initial Screening and Characterization of Findings," for the Mitigating Systems cornerstone. The inspectors evaluated the finding using Appendix A, "The Significance Determination Process for Findings At-Power." The inspectors answered 'Yes' to Question A.1 in Exhibit 4 because multiple trains of SFAS and RPS could potentially be adversely affected. Therefore, a detailed risk evaluation was required.

To evaluate this finding, the Senior Reactor Analysts (SRAs) utilized the methodology in NUREG/CR-4710, Appendix G, "External Wind Analysis for the St. Lucie Nuclear Power Plant," to determine the probability of a tornado missile striking a target. This was a generic methodology that would be applicable to all plants. The following definitions and assumptions from NUREG/CR-4710, Appendix G, were used:

- Ψ is defined to be the frequency of impact/missile/target area/tornado point strike frequency (in units of 1/missile/ft^2/yr).
- WF_{ψ} is defined to be the weighting factor associated with ψ related to a high, medium, or low exposure of missiles.
- The "Number of Missiles" for high/medium/low exposures is 60000/25000/5000 missiles.
- $WF_{\#\ Missiles}$ is defined to be the weighting factor associated with the number of missiles for high, medium, or low exposure of missiles.
- Area (ft^3) is the area of the target (i.e., the heat trace panels), and is conservatively set to 100 ft^2.

The tornado point strike frequency (1/yr) was obtained from NUREG/CR-4461, Revision 2, "Tornado Climatology of the Contiguous United States," as 3.77E-4/yr at the plant location of approximately 42° North latitude and 83° West longitude. It was very conservatively assumed that a missile hit on any of the heat trace panels would result in a core damage event.

A delta core damage frequency (ΔCDF) of 1.29E-7/yr was calculated as shown in the table below.

Exposure	Ψ (1/missile/ft^2/yr)	WF_{Ψ}	# Missiles	$WF_{\# Missiles}$	Area (ft^3)	Tornado Frequency (1/yr)	ΔCDF (1/yr)
High	2.42E-09	0.1	60000	0.2	100	3.77E-04	1.09E-07
Medium	8.64E-11	0.4	25000	0.6	100	3.77E-04	1.95E-08
Low	1.54E-11	0.5	5000	0.2	100	3.77E-04	2.90E-10
						Total =	**1.29E-07**

Since the total estimated change in core damage frequency was greater than 1.0E-7/yr, IMC 0609 Appendix H, "Containment Integrity Significance Determination Process," was used to determine the potential risk contribution due to LERF. The plant is a 2-loop Babcock and Wilcox Pressurized Water Reactor with a large dry containment. Sequences important to LERF include steam generator tube rupture events and inter-system LOCA events. These were not the dominant core damage sequences for this finding. Therefore, based on the detailed risk evaluation, the inspectors determined that the finding was of very-low safety significance (Green).

The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance.

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, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program."

Contrary to the above, as of February 12, 2015, the licensee failed to verify the adequacy of the safety-related essential inverters to interrupt fault conditions on its distribution system by either an analysis or testing in order to demonstrate compliance with industry standard IEEE 308-1971, for the required independence of 120 Vac safety-related inverters. Specifically, the station failed to verify that during LOOP conditions the essential inverters would continue to operate reliably due to the effects from postulated tornado generated missiles on equipment powered from the essential inverter distribution system.

Because this violation was of very-low safety significance, and it was entered into the licensee's CAP as CR 2015-01862, which initiated a Standing Order to the control room operators to de-energize the affected distribution panels based on a tornado warning for Ottawa County, this violation is being treated as an NCV, consistent with Section 2.3.2, of the NRC Enforcement Policy. (NCV 05000346/2015008-02, Failure to Comply with IEEE 308-1971 for the Required Independence of Safety-Related Essential Inverter Distribution Systems)

(3) Failure to Incorporate the Design Analysis Required Acceptance Limit into Surveillance Procedure

Introduction: The inspectors identified a finding of very-low safety significance (Green), and an associated NCV of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," for the failure to incorporate the design requirements and acceptance limits into test procedures. Specifically, the minimum voltage determined by analysis for the essential safety-related inverter YV1 bus during Modes 5 and 6 was not correctly incorporated into the surveillance procedure.

Description: The inspectors reviewed surveillance procedure, DB-SC-03042, “On-Site AC Bus Sources Lined Up and Available (Modes 5 and 6),” and found that the minimum required voltage in Table 4.1.16, Control Room, for the essential distribution panels was 114 volts. The licensee’s design analysis evaluated the essential inverter minimum rated voltage at 120 volts minus 2 percent, or 117.6 volts, in accordance with vendor specifications. The inspectors noted that minimum voltage limit in Table 4.1.4 at the output of inverter YV1 was 117.6 volts, and requested the licensee to explain the reason for the difference in the acceptance limits for minimum voltage.

The licensee initiated CR-2015-01028, and determined that the 114 volt minimum value in Table 4.1.16 was incorrect and was not supported by calculation C-EE-017.01-006, “Adequacy of 120V AC Essential Instrumentation System.” The licensee determined the correct value should have been 117.6 volts. The inspectors found the minimum value identified in surveillance procedure, DB-SC-03041, “On-Site AC Bus Sources Lined Up and Available (Modes 1, 2, 3, and 4),” for control room indication was correctly identified as 117.6 volts.

The licensee performed a 3-year review to determine if the readings recorded for essential panels Y1, Y1A, Y2, Y2A, Y3, or Y4 in Table 4.1.16 were 117.6 volts or above. The review identified one instance on Work Order 200492856, dated May 11, 2012, with a recorded reading lower than 117.6 volts. The value recorded for Y1 panel voltage was 116 volts at control room indicator EI6277. The licensee re-analyzed for the 116 volts as-found value at panel Y1, and determined the loads that did not have adequate rated voltage at the surveillance procedure minimum acceptance criteria voltage, 114 volts, would have had sufficient voltage to operate at the as-found measured voltage to perform their intended safety function such that no past operability concern existed.

Analysis: The inspectors determined the failure to incorporate the design required acceptance limit into test procedures was contrary to 10 CFR Part 50, Appendix B, Criterion XI, and was a performance deficiency. The finding was determined to be more than minor because it was associated with the Mitigating System cornerstone attribute of Procedure Quality, and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Phase 1-Initial Screening and Characterization of Findings,” for the Mitigating Systems cornerstone. The inspectors evaluated the finding using Appendix G, Attachment 1, “Shutdown Operations Significance Determination Process, Phase 1-Initial Screening and Characterization of Findings.” The inspectors answered "No" to all the Mitigating Systems Screening questions in Exhibit 3 and screened the finding as having very-low safety significance (Green) based on a re-analysis that verified the loads that did not have adequate rated voltage at the surveillance procedure minimum acceptance criteria voltage would have had sufficient voltage to operate at the as-found measured voltage.

The inspectors did not identify a cross-cutting aspect associated with this finding as it did not reflect current performance.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” requires, in part, that a test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified, and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents.

Contrary to the above, as of January 23, 2015, the licensee failed to incorporate the design analysis required acceptance limit for the essential inverter minimum voltage into test procedures. Specifically, the design minimum voltage on essential safety-related inverter YV1 bus during Modes 5 and 6 was not correctly incorporated into surveillance test procedures.

Because this violation was of very-low safety significance and was entered into the licensee's CAP as CR 2015-01028, which based on a re-analysis verified the measured voltage was acceptable, this violation is being treated as an NCV, consistent with Section 2.3.2, of the Enforcement Policy (NCV 05000346/2015008-03, Failure to Incorporate the Design Analysis Required Acceptance Limit into Surveillance Procedure)

(4) Failure to Verify Several Component Cooling Water System Manual Valves Were in the Correct Position

Introduction: The inspectors identified a finding of very-low safety significance (Green), and an associated Non-Cited Violation of TS Surveillance Requirement (SR) 3.7.7.1, for the licensee failure to verify several component cooling water (CCW) system manual valves in the flow path servicing safety-related equipment that were not locked, sealed, or otherwise secured, were in the correct position every 31 days. Specifically, the unsecured CCW pump seal water flush isolation valves (two valves per pump) for the two required operable CCW pumps were not verified open every 31 days.

Description: On December 13, 2008, Improved Technical Specifications (ITSs) were implemented by the licensee via License Amendment No. 279. Following implementation of ITS SR 3.7.7.1, required the licensee every 31 days to "Verify each CCW manual, power operated, and automatic valve in the flow path servicing safety-related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position."

During a plant walkdown, the inspectors noted that each CCW pump had two manual seal water flush isolation valves (CC5099A/B for pump 1, CC5100A/B for pump 2, and CC5101A/B for pump 3) that were not locked or secured open. The inspectors verified these CCW system manual valves were also not included in the licensee's surveillance procedures DB-SP-03063, "Component Cooling Water Train 1 Valve Verification Monthly Test," or DB-SP-03064, "Component Cooling Water Train 2 Valve Verification Monthly Test," to meet the 31 day verification requirement. A discussion with the pump vendor determined that CCW pump seal performance cannot be guaranteed without this flush supply. As such, these CCW system manual valves met the requirement of being in the flow path servicing safety-related equipment, but were not controlled in accordance with TS SR 3.7.7.1.

The licensee initiated CR 2015-01317, and verified the correct position of the six CCW pump seal flush isolation valves. The licensee planned to revise surveillance procedure DB-SP-03004, "Locked Valve Verification," and DB-OP-00008, "Operation and Control of Locked Valves," to lock open and periodically verify the locked open position of the CCW pump seal flush isolation valves.

Analysis: The inspectors determined the failure to verify the position of several CCW system manual valves in the flow path servicing the safety-related equipment that were not locked, sealed, or otherwise secured in position, was contrary to TS SR 3.7.7.1, and was a performance deficiency.

The finding was determined to be more than minor because it was similar to IMC 0612, "Power Reactor Inspection Reports," Appendix E, "Examples of Minor Issues," Example 3.c, because more than one valve was in the required position, but not locked, sealed, or otherwise secured in the correct position, and because it was associated with the Mitigating Systems cornerstone attribute of Configuration Control, and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems to respond to initiating events to prevent undesirable consequences. Specifically, a potentially mis-positioned valve in a safety-related CCW system flow path would render portions of the safety-related CCW system incapable of performing its required safety function.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1-Initial Screening and Characterization of Findings," for the Mitigating Systems cornerstone. The inspectors evaluated the finding using Appendix A, "The Significance Determination Process for Findings At-Power." The finding screened as very-low safety significance (Green) because the inspectors were able to answer "No" to all the screening questions in Exhibit 2, because the valves were verified to be in their correct position.

The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance.

Enforcement: The TS SR 3.7.7.1 states, in part, that each CCW manual valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is verified in the correct position every 31 days.

Contrary to the above, from December 13, 2008, to January 30, 2015, the licensee failed to ensure each CCW manual valve in the flow path servicing safety-related equipment, that is not locked, sealed, or otherwise secured in position, is verified in the correct position every 31 days. Specifically, the unsecured CCW pump seal water flush manual isolation valves for the two operable CCW pumps were not verified open every 31 days.

Because this violation was of very-low safety significance, and it was entered into the licensee's CAP as CR 2015-01317, which verified the valves were in their correct position, this violation is being treated as an NCV, consistent with Section 2.3.2, of the NRC Enforcement Policy. (NCV 05000346/2015008-04, Failure to Verify Several CCW System Manual Valves Were in the Correct Position)

.4 Operating Experience

a. Inspection Scope

The inspectors reviewed three operating experience issues (samples) to ensure that NRC generic concerns had been adequately evaluated and addressed by the licensee. The operating experience issues listed below were reviewed as part of this inspection:

- IN 91-56, “Potential Radioactive Leakage to Tank Vented to Atmosphere;”
- IN 2012-01, “Seismic Issues-Principally Issues Involving Tanks;” and
- IN 2012-11, “Age Related Capacitor Degradation.”

b. Findings

(1) Failure to Comply with Technical Specifications for the Borated Water Storage Tank

Introduction: The inspectors identified a finding of very-low safety significance (Green), and an associated NCV of TS 3.5.4, “BWST,” for the failure to comply with the limiting condition for operation (LCO) while the BWST was aligned to the non-seismic spent fuel pool (SFP) purification system, causing the BWST to be inoperable.

Description: While investigating the licensee’s response to IN 2012-01, “Seismic Considerations-Principally Issues Involving Tanks,” the inspectors reviewed the purification path of the BWST through the SFP demineralizer and filter. The SFP purification system consists of the SFP demineralizer and filter flow path, which is a non-safety, non-seismic system and is normally separated from the BWST by closed seismically qualified boundary valves. The BWST is seismically qualified and safety-related as described in Section 6.3 of the UFSAR. Per TS 3.5.4, the BWST is required to be operable in Modes 1 through 4, which included meeting its seismic qualification. The BWST can be manually aligned for purification through the SFP demineralizer, and filter by performing procedure DB-OP-06015, “BWST Operating Procedure,” Section 4.2. The procedure did not contain any mode restrictions for placing the BWST on SFP purification. With the BWST aligned to non-seismic piping, the inspectors questioned whether the seismic qualification of the BWST was being met.

The licensee had evaluated IN 2012-01 and justified placing the BWST on SFP recirculation above cold shutdown by: (1) stating that the design basis did not require the consideration of a concurrent LOCA and a seismic event; (2) procedure RA-EP-02820, “Earthquake,” had an immediate action to isolate the SFP demineralizer and filter by closing two manual valves following an earthquake; and (3) the presence of an interlock that closes an MOV to isolate the suction path to the SFP purification system based on low flow (this MOV, however, was not supplied with safety-related power and the control system associated with this interlock was also non-safety related). Discussions with the Office of Nuclear Reactor Regulation TS licensing branch confirmed that the licensee was in an unanalyzed condition that was outside their TS (i.e., operating outside their licensing basis) when the BWST was in a configuration in which it was aligned with the non-seismic SFP purification system.

A review of Operations logs identified two time periods since January 1, 2013, where the BWST was aligned in the SFP purification lineup while in Modes 1 through 4. The BWST was aligned for SFP purification for approximately 20 days (18 hours from December 18, 2013, through January 8, 2014), and for approximately 13 days (6 hours from April 29, 2014, through May 13, 2014). In the second instance the BWST was aligned for SFP purification beginning on April 23, 2014, while the plant was in Mode 6, and entered Mode 4 on April 29, 2014 while the BWST was aligned for SFP purification. Since TS 3.5.4 only allows the BWST to be inoperable for 1 hour, the licensee should have complied with the TS required actions while the BWST was aligned for SFP purification to place the plant in Mode 3 within 6 hours and in Mode 5 within 36 hours.

The licensee initiated CR 2015-01817 to evaluate the condition and initiated an LCO Tracking Log entry to document the BWST would be considered inoperable when placed on SFP purification in Modes 1 through 4, and to not place the BWST on SFP purification in these Modes.

Analysis: The inspectors determined the failure to comply with the required actions of TS LCO 3.5.4 while the BWST was aligned to the non-seismic SFP purification system, causing the BWST to be inoperable, was a performance deficiency.

The finding was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of Protection Against External Factors, and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems to respond to initiating events to prevent undesirable consequences. Specifically, when the BWST was aligned to the non-seismically qualified SFP purification system, the BWST no longer met its seismic qualification.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1-Initial Screening and Characterization of Findings," for the Mitigating Systems cornerstone. The inspectors evaluated the finding using Appendix A, "The Significance Determination Process for Findings At-Power." The inspectors answered 'Yes' to Question A.2 in Exhibit 2 because the finding represented the inoperability of the BWST for at least two time periods of 20 days (18 hours) and 13 days (6 hours), which was greater than the TS 3.5.4 allowed outage time of 1 hour for the BWST. Therefore, a detailed risk evaluation was performed using IMC 0609, Appendix A.

The risk evaluation was performed by Region III SRAs. Due to the nature of the performance deficiency, a seismic-induced LOCA event was considered. Using guidance from NRC's Risk Assessment Standardization Project (RASP) handbook, only the "Bin 2" seismic events were assumed to represent a ΔCDF. "Bin 2" was defined in the RASP handbook as seismic events with intensities greater than 0.3g, but less than 0.5g. Earthquakes of lesser severity were unlikely to result in large pipe failures, and earthquakes of a larger magnitude could result in major structural damage throughout the plant, which would not be representative of a differential risk (i.e., these events would represent "baseline" risk). The initiating event frequency of an earthquake in "Bin 2" was estimated to be 2.56E-5/yr using Table 4A-1 of Section 4, of the RASP handbook.

In order to bound the risk significance due to a seismic event, an evaluation was performed that conservatively assumed that a seismic event in "Bin 2" would result in a catastrophic failure of the contents of the BWST when the BWST was on SFP purification with no recovery actions credited.

In addition, the conditional probability of a small LOCA, and a medium LOCA for a seismic event in Bin 2 was obtained from Figure 4.5 of the RASP handbook (large LOCAs are not considered credible in Bin 2). These conditional probabilities were given as 4.5E-2 and 4E-3; respectively. The total probability of a LOCA for a seismic event in Bin 2 was thus 4.9E-2. The exposure time for when the BWST was on SFP purification, within a 1-year time period was approximately 34 days (i.e., 20 days [18 hours] plus 13 days [6 hours] = 34 days).

The bounding risk significance due to a seismic event was obtained by multiplying the frequency of a seismic event in Bin 2 (2.56E-5/yr) times the probability of a LOCA for a seismic event in Bin 2 (4.9E-2) multiplied by the exposure time (34 days/365 days):

$$\Delta CDF = [2.56E\text{-}5/yr] \times [4.9E\text{-}2] \times [34\ days/365\ days]$$

$$= 1.17E\text{-}7/yr.$$

Since the total estimated change in core damage frequency was greater than 1.0E-7/yr, IMC 0609 Appendix H, "Containment Integrity Significance Determination Process," was used to determine the potential risk contribution due to LERF. The plant is a 2-loop Babcock and Wilcox Pressurized Water Reactor with a large dry containment. The core damage sequences important to LERF included steam generator tube rupture (SGTR) events, and inter-system LOCA events. These events were not the dominant core damage sequences for this finding.

Therefore, based on the detailed risk evaluation, the SRAs determined that the finding was of very-low safety significance (Green).

This finding has a cross-cutting aspect in the area of Problem Identification and Resolution, because the licensee did not systematically and effectively evaluate and implement relevant external operating experience. Specifically, the licensee failed to systematically and effectively evaluate and implement relevant external operating experience from IN 2012-01. [P.5, Operating Experience]

<u>Enforcement</u>: The TS LCO 3.5.4 requires, the BWST to be OPERABLE in Modes 1 through 4. If the BWST is inoperable (for other than boron concentration or water temperature not within limits) in Modes 1 through 4, then the licensee is required to enter LCO Required Action B.1. Required Action B.1 requires that the BWST be restored to operable status within 1 hour. If not restored to operable status within 1 hour, Required Action C.1 requires the plant to be in Mode 3 within 6 hours, and Required Action C.2 requires the plant to be in Mode 5 within 36 hours.

Contrary to the above, from December 18, 2013, through January 8, 2014, and from April 29, 2014, through May 13, 2014, the licensee failed to enter LCO 3.5.4 Required Actions B.1, C.1, and C.2 when the BWST was aligned for SFP recirculation through the non-seismic SFP purification system for greater than 1 hour.

Because this violation was of very-low safety significance, and it was entered into the licensee's Corrective Action Program as CR 2015-01817, which initiated an LCO Tracking Log entry to not place the BWST on SFP purification in Modes 1 through 4, this violation is being treated as a NCV, consistent with Section 2.3.2, of the NRC Enforcement Policy. (NCV 05000346/2015008-05, Failure to Comply with TSs for the BWST)

.5 <u>Modifications</u>

a. <u>Inspection Scope</u>

The inspectors reviewed three permanent plant modifications related to selected risk significant components to verify that the design bases, licensing bases, and performance capability of the components had not been degraded through modifications. The modifications listed below were reviewed as part of this inspection effort:

- Field Change Request 86-272, Upgrade/Replacement of Essential Instrument AC System;

- Engineering Change Request (ECR 02-0809), HPI Pump Minimum Flow Requirements During Small Break LOCA; and

- ECR 03-0282, CAC Slow Speed Fan Flow Design Basis Change.

b. Findings

No findings were identified.

.6 Operating Procedure Accident Scenarios

a. Inspection Scope

The inspectors performed a detailed review of the procedures listed below associated with the selected scenario of a design basis SGTR. For the procedures listed, time critical operator actions were reviewed for reasonableness, and any interfaces with other departments were evaluated. The procedures were compared to UFSAR, design assumptions, and training materials to assure consistency. In addition, operator actions were observed during the performance of a design basis SGTR scenario on the station simulator.

The following operating procedures were reviewed in detail:

- DB-OP-02000, "RPS, SFAS, SFRCS Trip, or SG Tube Rupture," Revision 27; and
- DB-OP-02531, "Steam Generator Tube Leak," Revision 20.

The following time critical operator actions were reviewed and observed on the station simulator:

- SGTR-Initiate Reactor Coolant System Cooldown with Unaffected Steam Generator

- SGTR-Initiate Reactor Coolant System Depressurization

- SGTR-Isolate the affected Steam Generator

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

.1 Review of Items Entered Into the Corrective Action Program

a. Inspection Scope

The inspectors reviewed a sample of the selected component problems that were identified by the licensee, and entered into the CAP. The inspectors reviewed these issues to verify an appropriate threshold for identifying issues and to evaluate the effectiveness of corrective actions related to design issues. In addition, corrective action documents written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problem into the CAP. The specific corrective action documents that were sampled and reviewed by the inspectors are listed in the attachment to this report.

The inspectors also selected four issues that were identified during previous Component Design Bases Inspections to verify that the concern was adequately evaluated, and corrective actions were identified and implemented to resolve the concern, as necessary. The following issues were reviewed:

- NCV 05000346/2007007-04, Failure to Adequately Evaluate Postulated Failure of AFW Suction Piping;
- CR-G201-2007-23781, Scaffolding and 50.59 Requirements;
- Unresolved Item 05000346/2007007-05; Concern Regarding Safety-Related Battery Electrical Isolation, and NCV 05000346/2012002-02; Failure to Maintain Safety-Related DC Systems Design Control; and
- NCV 05000346/2009007-04, Inadequate Procedure for a Loss of Coolant Accident Outside Containment.

b. Findings

No findings were identified.

4OA6 Management Meetings

.1 Exit Meeting Summary

On February 27, 2015, the inspectors presented the inspection results to Mr. R. Lieb, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. The licensee indicated that none of the documents reviewed by the inspectors were considered proprietary information.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

R. Lieb, Site Vice President
M. Beier, System Engineer
D. Blakley, Supervisor Nuclear Engineering Analysis
K. Byrd, Director, Site Engineering
J. Carr, Operations Support SRO
C. Gale, Plant Engineering
E. Grindahl, Nuclear Engineer
J. Hook, Manager, Design Engineering
B. Kremer, Manager, Site Operations
G. Laird, Manager, Technical Services Engineering
S. Lorenzen, Design Electrical Engineer
B. Matty, Manager, Plant Engineering
G. Michael, Supervisor, Nuclear Mechanical/Structural Engineering
M. Murtha, Senior Consulting Engineer
M. Nelson, Nuclear Engineering Analysis
T. Summers, Director, Site Operations
V. Wadsworth, Regulatory Compliance
J. Whitright, Supervisor, Nuclear Electrical/I&C Engineering
G. Wolf, Supervisor, Regulatory Compliance

U.S. Nuclear Regulatory Commission

D. Kimble, Senior Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened/Closed

05000346/2015008-01	NCV	Vulnerability of EDG Crosstie to a Non-Essential Bus (1R21.3.b.(1))
05000346/2015008-02	NCV	Failure to Comply with IEEE 308-1971 for the Required Independence of Safety-Related Essential Inverter Distribution Systems (1R21.3.b.(2))
05000346/2015008-03	NCV	Failure to Incorporate the Design Analysis Required Acceptance Limit into Surveillance Procedure (1R21.3.b.(3))
05000346/2015008-04	NCV	Failure to Verify Several CCW System Manual Valves Were in the Correct Position (1R21.3.b.(4))
05000346/2015008-05	NCV	Failure to Comply with Technical Specifications for the Borated Water Storage Tank (1R21.4.b.(1))

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather, that selected sections of portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

CALCULATIONS

Number	Description or Title	Revision
60.019	Inadvertent Spray Actuation – Spray Temperature and Annulus Temperature Effect Addendum A01, A06	1
C-EE-002.01-010	DC CALC- Battery and Charger Sizing, Short Circuit, and Voltage Drop	31
C-EE-002.01-016	Station Battery Discharge Analysis for Beyond Design Basis Events	0
C-EE-002-01-011	Low Voltage Coordination Calculation	7
C-EE-004.01-002	Protective Relay Setpoint for Emergency Diesel Generator 1-1 (AC101)	4
C-EE-004.01-005	Protective Relay Setpoint for Component Cooling Pump Motor 1-1	1
C-EE-004.01-010	Protective Relay Setpoint for High Pressure Injection Pump Motor 1-2	2
C-EE-004.01-011	Protective Relay Setpoint for Decay Heat Pump Motor 1-1 (AC112)	2
C-EE-004.01-049	Bus C1/D1 Degraded Voltage, Loss of Voltage, & 27X-6 Relay Setpoints	15
C-EE-005.01-022	Protective Relay Setpoint for Containment Air Cooler Fan (BE110)	6
C-EE-006.01-026	Voltage Drop For GL 89-10 Valve Operators	28
C-EE-015.03-010	Short Circuit Analysis for AC Power System	2
C-EE-017.01-006	Adequacy of 120VAC Essential Instrumentation System	3
C-EE-024.01-011	Evaluation of Davis-Besse EDG Transient Response During Design Basis LOOP/LOCA, LOOP Only and Appendix R Loading	2
C-ICE-024.01-002	EDG Air Receiver Tank Pressure Indication Uncertainty	0
C-ICE-050.03-003	Auxiliary Feedwater Low Pressure Suction Setpoint	1
C-ME-016.04-041	Evaluation of the Temperature Increase of CCW System	0
C-ME-024.01-002	EDG Air Start Receivers Recharge Time	2
C-ME-050.01-004	Component Level Review Calculation for AOV MS5889A/B	5
C-ME-050.01-006	Maximum Expected D/P For Valves MS-5889A and MS-5889B	0
C-ME-050.03-129	AFW System Low Suction Pressure Switches Setpoint	1
C-ME-060.05-014	Containment Air Cooler System Fan Performance	0
C-ME-083.01-221	EN-DP-01092 Calculation of D/P Across MS106A & MS107A	2
C-ME-083.01-226	MOV Thrust-Torque Calculation for MS 106A & MS107A	12

CALCULATIONS

Number	Description or Title	Revision
C-ME-083.01-232	Effect of Temperature on MS106, MS107, MS106A and MS107A Limits	0
C-NSA-016.04-001	CCW Allowable Pump Degradation Addendum A01, A02	1
C-NSA-016.04-004	CCW Pump NPSH Requirements, Addendum A01, A02	1
C-NSA-028.01-007	Control Room, LPZ, and EAB Radiation Doses due to ECCS Leakage to the BWST and Auxiliary Building	0
C-NSA-049.02-015	DHR System Flow Requirements	1
C-NSA-049.02-026	NPSH Licensing Basis Analysis for Davis Besse LPI & CS Pumps	1
C-NSA-049.02-048	LPI, CS, and HPI Pump NPSH with Suction from the BWST	0
C-NSA-049.02-052	NPSH During Transfer from BWST to Emergency Sump	0
C-NSA-052.01-003	HPI Pump Acceptance Criteria	8
C-NSA-052.01-011	HPI NPSH on CTMT Sump Recirculation	1
C-NSA-052.01-012	Maximum Allowable Leakage Through HP31/32 or ECCS Systems	0
C-NSA-059.01-019	Water Level Inside Containment Post-LOCA	5
C-NSA-060.05-007	CAC Heat Duty at Elevated SW Inlet Temperatures Addendum A01	2
C-NSA-060.05-010	Containment Vessel Analysis Addendum A01	8

CORRECTIVE ACTION DOCUMENTS Generated Due to the Inspection

Number	Description or Title	Date
2015-00999	Deficiencies in Requested Documentation Provided to NRC	01/23/15
2015-01028	On-site AC Bus Sources Modes 5/6 Non-Conservative Acceptance Criteria	01/23/15
2015-01178	CCW Pump 1 Outboard Motor Bearing Oil Leak	01/27/15
2015-01179	White Powder on Floor Near ECCS Room Cooler Fan 3	01/27/15
2015-01203	Trouble Light above DH Pump 1	01/28/15
2015-01208	Oil on the CC1495 Actuator	01/27/15
2015-01271	Calculation C-EE-002.01-011 Uses the Melting Time Instead of Clearing Time for A25X30 Fuses	01/29/15
2015-01311	Enhancement to DB-OP-02527 Rev. 19 Loss of Decay Heat Removal – DHR Temperature Limits – Attachment 6	01/30/15
2015-01317	Position of CCW Seal Flush Valves Not Verified IAW SR 3.7.7.1	01/30/15
2015-01563	MS106A/MS107A Differential Pressure	02/05/15
2015-01762	Calculation C-EE-002.01-011 Does Not Address Impacts of Non-Safety Loads on Inverter Current Limiting	02/10/15
2015-01817	Operability of BWST while on SFP Recirculation	02/11/15
2015-01830	MS5889A and MS5889B Differential Pressure	02/11/15
2015-01839	HSDH63 & HSDH64 Enable/Disable Switch Position Verification Should Be Added to Operator Rounds	02/12/15
2015-01839	HSDH63 & HSDH64 Enable/Disable Switch Position Verification Should be Added to Operator Rounds	02/12/15
2015-01862	Inverter Supplied Loads Not Protected From Tornado Missiles	02/12/15

CORRECTIVE ACTION DOCUMENTS Generated Due to the Inspection

Number	Description or Title	Date
2015-01919	PM had No Documented Evaluation that Determined Measured CAC Air Flows Met Analytical Requirements	02/13/15
2015-01989	Calculation C-EE-002.01-010 Does Not Evaluate Inverter Load Change During a LOOP and LOCA Due to Y1A and Y2A	02/16/15
2015-02398	Extrapolation Pressure for Measured ECCS Leakage	02/25/15
2015-02419	EDG Jacket Water Heat Load at EDG 30 Minute Load Rating	02/25/15
2015-02437	Potential Preconditioning of DH14A	02/25/15
2015-02476	EDG Powering Nonessential 4160V Bus	02/26/15

CORRECTIVE ACTION DOCUMENTS Reviewed During the Inspection

Number	Description or Title	Date
2002-06701	Post-LOCA Dose from BWST with Inadvertent HP31/HP32 Failure	09/25/02
2002-07701	Control Room Operator Dose Due to ECCS Leakage Post-LOCA	10/09/02
2003-01663	Operation Of HPI Motors In Service Factor Range	03/01/03
2003-03493	CAC Slow Speed Airflow did Not Meet 58,000 cfm Acceptance Criteria	05/06/03
2015-05314	Flux-Delta Flux/Flow Trip of RPS Channel 3	10/11/05
2007-23781	Scaffolding and 50.59 Requirements	07/19/07
2008-37417	DB-PA-08-01 NG-DB-00235 Control of Time Critical Activities Implementation	03/28/08
2009-56365	HPI Pump 2 Discharge Pressure Reading Low Prior to Test	03/31/09
2009-58812	Decay Heat Pipe Vibrations While Stroking DH cooler outlet valve	05/07/09
2009-64986	CCW pump 1 Outboard motor bearing minor oil leak	09/25/09
2009-66474	2009 CDBI: Procedures for LOCA Outside Containment	10/22/09
2009-67370	Evaluate PRA Process to Strengthen Alignment with Operations Procedure	11/06/09
2010-72492	Unexpected Results For DB-PF-03100 CC1495 Air Drop Test	03/03/10
2010-75230	HPI #2 Baseline Test, Motor Data Greater Than 100% Full Load Amps	04/12/10
2010-75230	HPI #2 Baseline Test, Motor Data Greater Than 100% Full Load Amps	04/12/10
2010-80711	Output Indicator For HICDH14B Could Not Be Calibrated	08/03/10
2010-81761	Pump Basis Document Not Updated to Incorporate Results of 16RFO Baseline Testing	08/25/10
2010-86762	#2 AFW Pump Steam Inlet Vlv Stroke Time did Not Meet the Expected Range	12/08/10
2011-87769	Degraded Voltage Relays 27A-3 & 27A-4 "As Found" Values Out Of Tolerance	01/05/11
2011-96164	Undervoltage Relay Found Out Of Calibration	06/08/11
2011-98223	DC System Issues From NRC CDBI	07/26/11
2011-98357	EDG Does not Meet IEEE-387-1972	07/28/11

CORRECTIVE ACTION DOCUMENTS Reviewed During the Inspection

Number	Description or Title	Date
2011-00800	27A-3 and 27A-4 Failed Section 8.7 Degraded Voltage Setpoint Check Performing D1 Undervoltage Test	08/19/11
2011-03193	DB-PF-03100 Air Drop Test For CC1495 Indicates Negative Leakage	10/07/11
2011-04761	NE Aux Bld Hdr Air Regulator is leaking by Causing CC1495 to Fail Closed.	11/02/11
2011-04959	CACs 1, 2 and 3 Slow Speed Airflow Rate Decreased by about 5% but met 45,000 cfm Acceptance Criteria	11/05/11
2012-02697	Safety Related Inverter Static Switch Setpoints	02/20/12
2012-03009	MS5889B - Measurement Uncertainty Extent of Condition	02/27/12
2012-04105	MS5889B Low Opening Margin	03/16/12
2012-08489	Marking Operation 140 of PM 4871 As Left LLRT Step N/A for all CTMT Vacuum Breaker Check Valves Under Order 200423284	05/22/12
2012-08434	Replacement Relays for Inverter YV1-PM Exception	05/22/12
2012-08753	Marking N/A Operation 0120 of Order 200423284 As Left IST Test DB-PF-03809 for Ctmt Vacuum Breaker Check Valves	05/26/12
2012-09235	PORV Leakage Noted Following PORV Cycle Test	06/04/12
2013-00632	DH Cooler 1 Outlet Flow Control valve has higher bench set	01/16/13
2013-04432	During Calibration Of Degraded Voltage Relays, Relays Found Out of Tolerance for Setpoint but within Tech Spec Allowable values	03/25/13
2013-07070	CCW Pump 1 Inboard Temp Instrument Failed	05/04/13
2013-07067	High Bearing Temperature Alarm on CCW pump 1	05/04/13
2013-10712	Installed Frequency Meter on Panel C3615 for Emergency Diesel Generator 1 out of Desired Range	07/14/13
2013-15593	CCW Pump 1 Inboard Pump Bearing Oil Leak	10/02/13
2013-19507	CC1495 Closed During Installation of Switch Cover for LSLL3757A	12/10/13
2014-01172	CR-2014-01172 Causal Analysis for CC1495 Closed During Installation of Switch Cover	01/24/14
2014-02481	Decay Heat Cooler #1 Outlet Flow Control Valve Erratic	02/09/14
2014-02968	CV5080 did not Meet the Acceptance Criteria of Test DB-PF-03809	02/14/14
2014-03111	CCW Pump 1 Inboard Pump Bearing Oil at Low Level Mark	02/17/14
2014-03808	MS5889B Failed as Found Stroke Test	02/25/14
2014-04844	All Indicating Lights for Control Room Emergency Ventilation System Train 1	03/12/14
2014-05348	CCW Pump 1 Low Flow Alarm in at 3200 gpm is Acceptable	03/20/14
2014-07123	Train 1 Decay Heat Cooler Outlet Temperature TIDH4B1 Reading High	04/19/14
2014-07168	CC1495, CCW To Auxiliary Building Non-Essential Loads Isolation Valve Stroke Time Evaluation	04/21/14
2014-07666	High Voltage Condition Experienced on C1 and D1 Busses	04/27/14

CORRECTIVE ACTION DOCUMENTS Reviewed During the Inspection

Number	Description or Title	Date
2014-07838	CAC Fan 1 Local Differential Pressure Indicator Reads 0 with Fan Running	04/29/14
2014-09470	CAC 1 was Declared Inoperable and LCO 3.6.6 Condition C was Entered	05/25/14
2014-10133	Maintenance Rule (a)(1) Determination For CAC1	06/09/14
2014-11002	Unanalyzed Loading Condition in EDG Transient Analysis Calculation	06/26/14
2014-11043	27A-2 Degraded Voltage Rely Found Outside Setpoint Value and Inside Allowable Values	06/27/14
2014-13456	DB-SP-03152 Open Indication was Not Obtained for MS106A	08/24/14
2014-13985	HPI Pump 2 Motor Does Not Meet PO Requirements	09/09/14
2014-18368	Time Critical Operator Action Not Met During Periodic Validation – SGTR Action 4 – Cool RCS from 500 Degrees F to DHR in Service	12/15/14
2015-00709	10 CFR Part 21 Error Report for ETAP	01/19/15
2015-01129	EDG Air Receiver Inlet Check Valve Leak	01/27/15

DRAWINGS

Number	Description or Title	Revision
03-8300-N	Assy-CV2 Check Valve 8" – 45 PSI W.P. Sch 40 weld ends	K
7749-E-30-14	Schematic Diagram Safety Features Actuation System	15
7749-E-30-28	Schematic Diagram Safety Features Actuation System	15
7749-E-5-91	Connection Diagram Unit 11 Indoor Metal Clad Switchgear Bus D1	5
7749-E-8-103	A906CC5 With Latch In Low Speed	12
E-1 Sh. 1	AC Electrical System One Line Diagram	37
E-1 Sh. 2	AC Electrical System One Line Diagram	74
E-3	4.16KV Metering and Relaying One Line Diagram	44
E-6 Sh. 1	480V AC MCC (Essential) One Line Diagram	88
E-6 Sh. 2	480V AC One Line Diagram	95
E-6 Sh. 3	125/250 V.D.C. MCC No. 1 (Essential) Single Line Diagram,	43
E-6 Sh. 4	125/250 V.D.C. MCC No. 2 (Essential) Single Line Diagram,	33
E-7	250/125V DC and Instrumentation AC One Line Diagram	53
E-30B Sh. 16A	General Guide 13.8KV and 4.16KV Circuit Breakers Internal Wiring	1
E-30B Sh. 8G	General Guide - Elementary Diagrams Miscellaneous Switch Development	2
E-34B Sh. 13	Elementary Wiring Diagram 4.16KV Feed Breakers Bus C1(D1) Tripping and Lockout Relays	12
E-34B Sh. 14	Elementary Wiring Diagram 4.16KV Feed Breakers Bus C1(D1) Voltage and Aux. Relays	12
E-46B Sh. 71	Elementary Wiring Diagrams Steam & Condensate AFPT MN STM IN ISO VLV's	6
E-46B Sh. 46A	Elementary Wiring Diagram Steam Generator Aux Feed Pump Turbine Isolation Valve	21

DRAWINGS

Number	Description or Title	Revision
E-46B Sh. 46B	Elementary Wiring Diagram Steam Generator Aux Feed Pump Turbine Isolation Valve	17
E-49B Sh. 1B	Elementary Wiring Diagram Treated Water Makeup Pump	21
E-50B Sh. 15a	Elementary Wiring Diagrams CC AUX EQUIP IN VLV	5
E-50B Sh. 3C	Elementary Wiring Diagram Cooling Water System Component Cooling Pump 1 (AC113)	11
E-50B Sh. 3D	Elementary Wiring Diagram Cooling Water System Component Cooling Pump 1 (AC113)	5
E-52B Sh. 5C	Elementary Wiring Diagram Reactor Cooling System HP INJ Pump 1-1	3
E-52B Sh. 5D	Elementary Wiring Diagram Reactor Cooling System HP INJ Pump 1-2	2
E-52B Sh. 6A	Elementary Wiring Diagram Reactor Cooling System DH Pump 1-1	14
E-52B Sh. 6B	Elementary Wiring Diagram Reactor Cooling System DH Pump 1-1	16
E-52B Sh. 13	Elementary Wiring Diagrams Reactor Coolant System RC PRZR Auto Vent to Quench Tank	16
E-52B Sh. 66	Elementary Wiring Diagram – HPI-LPI Cross Conn Iso Vlvs	10
E-58B Sh. 1A	Elementary Wiring Diagram Containment Cooler Fan 1	14
E-58B Sh. 1B	Elementary Wiring Diagram Containment Cooler Fan 1	11
E-64B Sh. 1A	Elementary Wiring Diagram Emergency Diesel Generator 1-1 Breaker AC101 Control	13
E-64B Sh. 1B	Elementary Wiring Diagram Emergency Diesel Generator 1-1 Breaker AC101 Control	12
E-64B Sh. 1C	Elementary Wiring Diagram Emergency Diesel Generator 1-1 Protective Relays Tripping and Lockout Circuits	10
E-64B Sh. 1D	Elementary Wiring Diagram Emergency Diesel Generator 1-1 Breaker AC101 Control	5
E-64B Sh. 1E	Elementary Wiring Diagram Emergency Diesel Generator 1-1 Aux Relays	21
E-64B Sh. 17	Elementary Wiring Diagram Emergency Diesel Generator SFAS Sequencer Start/Stop	5
E-64B Sh. 18	Elementary Wiring Diagram Emergency Diesel Generator SFAS Sequencer Start/Stop Aux Relays	3
E-640A Sh. 1A	Essential 125V DC Distribution Panel “D1P” Channel - 1	22
E-640A Sh. 1B	Essential 125V DC Distribution Panel “D1P” Channel - 1	14
E-640A Sh. 2A	Essential 125V DC Distribution Panel “D2P” Channel - 2	21
E-640A Sh. 2B	Essential 125V DC Distribution Panel “D2P” Channel - 2	14
E-640A Sh. 4A	Essential 125V DC Distribution Panel “D2N” Channel 4	13
E-641A Sh. 1A	Essential 125VAC Instr. Distr Pnl “Y1” Channel - 1	35
E-641A Sh. 1B	Essential 125VAC Instr. Distribution Panel “Y1” Channel - 1	8
E-641A Sh. 2A	Essential 125VAC Instr. Distr Pnl “Y2” Channel - 2	39
E-641A Sh. 2B	Essential 125VAC Instr. Distribution Panel “Y2” Channel - 2	11
E-908A	Essential 120V AC Instr. Distr. Pnl. “Y1A”, Channel 1	7

DRAWINGS

Number	Description or Title	Revision
E-909A	Essential 120V AC Instr. Distr. Pnl. “Y2A”, Channel 2	7
M-003C Sh. 3	Main Steam and Reheat System	63
M-017B	Diesel Generator Air Start	47
M-030A	Reactor Coolant System	70
M-033A	High Pressure Injection	44
M-033B	Decay Heat Train 1	56
M-033C	Decay Heat Train 2	27
M-035	Spent Fuel Pool Cooling System	53
M-036A	Component Cooling Water System	30
M-036B	Component Cooling Water System	40
M-036C	Component Cooling Water System	32
M-041A	Service Water Pumps and Secondary Service Water System	30
M-041B	Primary Service Water System	72
M-041C	Service Water System for Containment Air Coolers	47
OS-001A Sh. 1	Reactor Coolant System	46
OS-001A Sh. 2	Reactor Coolant System	29
OS-017B Sh. 1	Auxiliary Feedwater Pump and Turbines	25
OS-021 Sh. 1	Component Cooling Water	37
SF-003B Sh. 13	SRFCS Schematic Diagram AFPT-1 MN STM-1 Inboard ISO Valve MS-106	2
SF-003B Sh. 22	SFRCS Schematic Diagram AFPT-2 MN STM IN ISO VLV MS-5889B	6
SK5256	Aero Corporation CAC Type “RC” coil	B

10 CFR 50.59 DOCUMENTS (SCREENINGS/SAFETY EVALUATIONS)

Number	Description or Title	Date
89-0116	Safety Evaluation for FCR 86-272	06/13/89
03-01408	10 CFR 50.59 Evaluation – Installation of New Minimum Flow Recirculation Lines for the HPI Pumps	08/22/03
07-00509	10 CFR 50.59 Evaluation – Dose Increase Caused by Leakage into the Auxiliary Building and BWST	04/12/07
13-01934	10 CFR 50.59 Screen - DB-MS-01637 - Scaffolding Erection and Removal	05/17/13

MISCELLANEOUS

Number	Description or Title	Date or Revision
	2002 CAC 1 Motor Vendor Test Data POQA 7096792	
	CCW System Health Report	Q4-2014
	Containment Isolation Valve System Health Report	Q4-2014
	Containment Air Cooler System Health Report	Q4-2014
1-02-011A	Relay Setting Manual for AC101	3
1-02-012	Relay Setting Manual for AC112	8
1-02-012A	Relay Setting Manual for AC113	0

MISCELLANEOUS

Number	Description or Title	Date or Revision
1-02-019	Relay Setting Manual for AD111	8
1-02-024A	Relay Setting Manual for AC101	7
1-08-066	Relay Setting Manual for MV106A	12
45289629	Certificate of Compliance – Power Operated Relief Valve	09/08/10
DB-SC-10111/00	Test Summary Report Channel 1 Instrument AC System Acceptance Test	06/20/90
E-005-00154	ABB Brown Boveri Instruction Manual For I-T-E Single Voltage Relays	4
E-854Q-111-1	SCI Product Manual, UPS Systems for Computer and Industrial Applications, Section VII, Maintenance and Troubleshooting	1966
EQP: DB1-004C	ASCO Solenoid Valve	13
EQP: DB1-086A	PORV Solenoid Valve Operator	0
LCOTR Log	CR 2015-01817 Has Identified that the BWST has been Determined to be Inoperable when Placed on SFP Purification in Modes 1 through 4	02/12/15
Log Entries Report	Control Room Log Entries for Miscellaneous Dates in 2013 and 2014	12/18/13 01/08/14 04/29/14 05/13/14
M-222-00004-03	Containment Vacuum Relief Check Valves (CV2 CK) Vendor Manual	03/18/05
M-314-00187-07	Instructions for Solenoid Valves	7
M-467Q-00001	Technical Manual for Davis Besse Power Operated Relief Valve Target Rock Model 08JJ-001	1
M-517-00021-06	Decay Heat Cooler Vendor Manual	03/30/05
N/A	Fourth Interval Inservice Inspection Plan	1
NEN-91-10215	Makeup/Feed and Bleed System Design Criteria	06/24/91
NEO-91-00798	Review of IN 91-56 – Potential Radioactive Leakage to Tank Vented to Atmosphere	10/16/91
Notification 600949760	Document Change Request – Enhancement for DH63/DH64 Closure Requirements	02/12/15
OE-2012-0277	IN 2012-01: Seismic Considerations – Principally Issues Involving Tanks	02/10/12
OE-2012-0293	Inappropriate Temporary Connection of Non-Seismic Systems/Components to Seismically Qualified Systems	02/15/12
PN 01	Post-It-Note – Calculation NOP-CC-3002-03, Rev. 00	02/26/15
PO 55118092	Eddy Current Report for Davis Besse CAC 1 (E37-1) 18RFO	Feb2014
System 24-01	System Health Report – EDG	07/30/14
System 48-01	System Health Report – SFAS	07/30/14
System 49-01	System Health Report – DH/LPI	07/30/14
System 52-01	System Health Report – HPI	07/30/14
System 62-01	System Health Report – RCS	07/30/14

MISCELLANEOUS

Number	Description or Title	Date or Revision
UCN 07-010	UFSAR Change Notice Form – Changes to Control Room Dose	04/27/07

MODIFICATIONS

Number	Description or Title	Date or Revision
ECP 08-0571-000	Auxiliary Feedwater Suction Piping	3
ECP 10-0313-001	Revise Setpoints for Auxiliary Feedwater Pump Suction Pressure Switches	0
ECR 02-0809	HPI Pump Min. Flow Requirements During Small Break LOCA	08/22/03
ECR 03-0282-00	CAC Slow Speed Fan Flow Design Basis Change	06/12/03
FCR 86-0272	Upgrade/Replacement of Essential Instrument AC System	4

PROCEDURES

Number	Description or Title	Revision
DBBP-OPS-1013	Control of Time Critical Actions	2
DB-ME-09107	Westinghouse DHP Breaker Refurbishment	9
DB-ME-09202	Maintenance of Essential SCI UPS	6
DB-MS-01637	Scaffold Erection and Removal	15
DB-OP-00008	Operation and Control of Locked Valves	13
DB-OP-02000	RPS, SFAS, SFRCS Trip, or SG Tube Rupture	27
DB-OP-02001	Electrical Distribution Alarm Panel 1 Annunciators	30
DB-OP-02003	ECCS Alarm Panel 3 Annunciators	16
DB-OP-02005	Primary Instrument Alarm Panel 5 Annunciators	18
DB-OP-02521	Loss of AC Bus Power Sources	23
DB-OP-02522	Small RCS Leaks	13
DB-OP-02527	Loss of Decay Heat Removal	18
DB-OP-02531	Steam Generator Tube Leak	20
DB-OP-03004	Locked Valve Verification	23
DB-OP-06003	Pressurizer Operating Procedure	30
DB-OP-06011	High Pressure Injection	29
DB-OP-06012	Decay Heat and Low Pressure Injection Operating Procedure	62
DB-OP-06014	Core Flooding System Procedure	28
DB-OP-06015	Borated Water Storage Tank Operating Procedure	3,18
DB-OP-06261	Service Water System Operating Procedure	63
DB-OP-06262	Component Cooling Water System Procedure	35
DB-OP-06315	4160 Volt Switching Procedure	15
DB-OP-06316	Diesel Generator Operating Procedure	57
DB-OP-06331	Freeze Protection & Electrical Heat Trace	25
DB-OP-06900	Plant Heatup	61
DB-OP-06903	Plant Cooldown	47

PROCEDURES

Number	**Description or Title**	**Revision**
DB-PF-03205	ECCS Train 1 Valve Test	21
DB-PF-03206	ECCS Train 2 Valve Test	20
DB-SC-03041	On-site AC Bus Sources Lined Up, Available and Isolated (Modes 1, 2, 3, 4)	14
DB-SC-03042	On-Site AC Bus Sources Lined Up and Available (Modes 5 & 6)	19
NOP-LP-4003	Evaluation of Changes, Tests and Experiments	7
RA-EP-02820	Earthquake	9

SURVEILLANCES (COMPLETED)

Number	**Description or Title**	**Date**
DB-PF-03008	Surveillance Test Procedure Containment Local Leakage Rate Tests	2/18/14
DB-PF-03011	ECCS Integrated Train 1 Leakage Test	12/21/13 03/21/14
DB-PF-03012	ECCS Integrated Train 2 Leakage Test	12/24/13 03/30/14
DB-PF-03071	CCW Train 1 Valve Test	12/19/14
DB-PF-03075	CCW Pump and Check Valve Test	03/22/14
DB-PF-03100	CCW Valve Test	12/19/14
DB-PF-03205	ECCS Train 1 Valve Test	06/28/14 09/19/14 12/12/14 02/10/13
DB-PF-03206	ECCS Train 2 Valve Test	05/20/14 08/15/14 11/05/14
DB-PF-03208	HPI Pump Comprehensive and Check Valve Forward Flow Test Train 2	04/17/14
DB-PF-03809	Surveillance Test Procedure Containment Vacuum Relief Check Valve Operability Test	2/25/14
DB-PF-04703	Decay Heat and LPI System DH Cooler 1-1 Performance Test	03/02/14
DB-SP-03019	Service Water Valve Verification Monthly Test Train 1	01/13/15
DB-SP-03026	Service Water Valve Verification Monthly Test Train 2	01/22/15
DB-SP-03063	Component Cooling Water Train 1 Valve Verification Monthly Test	01/15/15
DB-SP-03064	Component Cooling Water Train 2 Valve Verification Monthly Test	12/28/14
DB-SP-03134	Containment Emergency Sump Visual Inspection	04/28/14
DB-SP-03136	Decay Heat Train 1 Pump and Valve Test (Mode 4 – Defueled)	04/13/14
DB-SP-03219	HPI Train 2 Pump and Valve Test	12/31/14
DB-SP-03294	Surveillance Test Procedure Containment Air Cooling Unit 1 Monthly Test	05/01/14

SURVEILLANCES (COMPLETED)

Number	Description or Title	Date
DB-SP-03297	Surveillance Test Procedure Containment Air Cooling Unit 1 18 month test	01/07/13
DB-SP-03446	Decay Heat Train 1 Pump and Valve Test (Mode 1-3)	07/01/14 09/27/14 12/16/14 12/16/14
DB-SP-03447	Decay Heat Train 2 Pump and Valve Test (Mode 1-3)	05/23/14 08/06/14 11/04/14

WORK DOCUMENTS

Number	Description or Title	Date
00-000970-045	A1200Q08 Breaker Refurbishment	02/27/01
00-000970-018	A1200Q24 Breaker Refurbishment	06/07/01
200000630	A1200Q31 Breaker Refurbishment	03/28/04
200144389	DH Cooler 1-1 Performance Test per DB-PF-4703	03/02/10
200312225	PM 5480 YV1 & YRF1 RPLC Capacitors	06/02/12
200353901	Replace RC2A (PORV) With a New Design	12/10/11
200404212	A1200Q21 Breaker Refurbishment	07/14/12
200422161	AD111 Breaker Swap	12/14/12
200422715	AC112 Breaker Swap	03/04/13
200423257	PM 1325 Clean and Inspect CACs	11/09/11
200423732	PM 6578 Measured and Documented CAC air flow rates	11/07/11
200425419	CC1495 Air Drop Tested by DB-PF-03100	10/07/11
200426685	AC101 Breaker Swap	02/14/13
200432020	Replace Valve Stem and Actuator Gearing for MS106A to Increase Actuator Capability	12/22/11
200433368	AC113 Breaker Swap	03/01/13
200447335	PM 4951 BE1271*TEST TD MCCE 12B (MV106A)	08/01/13
200450461	CAC Unit 1 18 Month Test	01/07/13
200458857	PF3154-002 Valve Position Indication	05/08/13

WORK DOCUMENTS

Number	Description or Title	Date
200476037	PM 0260 MV106A*Insp* SG2>1AFPT (BE1271)	08/01/13
200485309	ECP 11-0510-002 PM 1445 HV5889B *	05/06/14
200492856	SC3042-001 04.000 On-Site AC Bus Sources FA Norm OPS, 1	05/11/12
200495143	CC1495 Valve and Actuator Replaced per ECP 11-0614 supp 03	04/23/14
200510668	PM 2375 SV5889B RPLC SOL AFPT#2	05/19/14
200512714	CCW Pump 1 Flow Test per DB-PF-03075	03//22/14
200517191	SC3261-001 Integrated Test of SFRCS Actuation	02/04/14
200532978	CC1495 Stroke Time Tested by DB-PF-03071	09/27/14

200532979	CC1495 Position Indication Tested by DB-PF-03071	09/27/14
200537005	SP3160-001, AFP 2 Quarterly	11/28/14
200538259	SP3152-001 AFW Train 1 Interlock Test	12/14/14
200540626	PF-03154-001 AFW Train 1 Valve Test	12/08/14
200540663	SP3153-001 AFW Train 1 Valve Verification	12/17/14
200541414	CC1495 Stroke Time Tested by DB-PF-03071	12/19/14
200541937	SP3162-001 AFW Train 2 Valve Verification	01/01/15
200589208	SC3042-001 04.000 On-Site AC Bus Sources FA Norm OPS, 1	04/26/14
200589501	CV5080 LLRT	02/18/14
200609717	SC3041-001 04.000 Bus Source On-Site AC Bus Sources FA Norm OPS	12/26/14

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access Management System
BWST	Borated Water Storage Tank
CAC	Containment Air Cooler
CAP	Corrective Action Program
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CR	Condition Report
CVT	Controlled Voltage Transformer
ΔCDF	Delta Core Damage Frequency
ECCS	Emergency Core Cooling System
ECR	Engineering Change Request
EDG	Emergency Diesel Generator
IEEE	Institute of Electrical & Electronic Engineers
IMC	Inspection Manual Chapter
IN	Information Notice
ITS	Improved Technical Specifications
kV	Kilovolt
LCO	Limiting Condition for Operation
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
LOOP	Loss of Off-site Power
MOV	Motor-Operated Valve
NCV	Non-Cited Violation
NPSH	Net Positive Suction Head
NRC	U.S. Nuclear Regulatory Commission
PARS	Publicly Available Records System
PORV	Power-Operated Relief Valve
PRA	Probabilistic Risk Assessment
RASP	Risk Assessment Standardization Project
PS	Reactor Protection System
SDP	Significance Determination Process
SFAS	Safety Features Actuation System
SFP	Spent Fuel Purification
SGTR	Steam Generator Tube Rupture
SR	Surveillance Requirement
SRA	Senior Reactor Analyst
UFSAR	Updated Final Safety Analysis Report
Vac	Volts Alternating Current
Vdc	Volts Direct Current

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Sincerely,
/RA/

Christine A. Lipa, Chief
Engineering Branch 2
Division of Reactor Safety

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