UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE RD. SUITE 210
LISLE, IL 60532-4352

July 21, 2015

Mr. Bryan C. Hanson
Senior VP, Exelon Generation Company, LLC
President and CNO, Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BYRON STATION, UNITS 1 AND 2 - NRC COMPONENT DESIGN BASES INSPECTION; INSPECTION REPORT 05000454/2015008; 05000455/2015008 AND NOTICE OF VIOLATION

Dear Mr. Hanson:

On June 16, 2015, the U.S. Nuclear Regulatory Commission (NRC) completed a Component Design Bases Inspection at your Byron Station, Units 1 and 2. The purpose of this inspection was 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. The enclosed report documents the results of this inspection, which were discussed on June 16, 2015, with Mr. B. Currier, and other members of your staff.

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

Based on the results of this inspection, the NRC has identified an issue that was evaluated under the risk Significance Determination Process as having very-low safety significance (Green). The NRC has also determined that a violation is associated with this issue. This violation was evaluated in accordance with the NRC Enforcement Policy. The current Enforcement Policy is included on the NRC's web site at http://www.nrc.gov/about-nrc/regulatory/enforcement/enforce-pol.html.

The violation is cited in the enclosed Notice of Violation (Notice), and the circumstances surrounding it are described in detail in the subject inspection report. The violation is being cited in the Notice because Byron Station, Units 1 and 2, failed to restore compliance and failed to have objective plans to restore compliance in a reasonable period following the NRC identification of an associated Non-Cited Violation (NCV) on June 15, 2012. The associated NCV was documented in Inspection Report 05000454/2012007; 05000455/2012007.

You are required to respond to this letter, and should follow the instructions specified in the enclosed Notice when preparing your response. If you have additional information that you believe the NRC should consider, you may provide it in your response to the Notice. The NRC review of your response to the Notice will also determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

Based on the results of this inspection, the NRC has also determined that six additional NRC-identified findings of very-low safety significance (Green) 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 NCVs in accordance with Section 2.3.2 of the NRC Enforcement Policy. These NCVs are described in the subject inspection report.

If you contest the subject or severity of the Non-Cited-Violation, 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 Byron 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 Byron 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 Nos. 50–454; 50–455
License Nos. NPF–37; NPF–66

Enclosures:
(1) Notice of Violation
(2) IR 05000454/2015008; 05000455/2015008;

cc w/encl: Distribution via LISTSERV®

NOTICE OF VIOLATION

Exelon Generation Company, LLC
Byron Station, Units 1 and 2

Docket No. 50-454; 50-455
License No. NPF-37; NPF-66

During an U.S. Nuclear Regulatory Commission (NRC) inspection conducted from April 20, 2015, through May 22, 2015, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

> Title 10, *Code of Federal Regulations* (CFR), Part 50, Appendix B, Criterion XVI, "Corrective Action," states, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected.
>
> Contrary to the above, from June 15, 2012, to May 22, 2015, the licensee failed to correct a condition adverse to quality (CAQ). Specifically, on June 15, 2012, the NRC issued a Non-Cited Violation 05000454/2012007-05; 05000455/2012007-05 for the failure to provide means to detect and isolate a leak in the emergency core cooling system within 30 minutes for Byron Station, Units 1 and 2, as described in Section 6.3.2.5 of the Updated Final Safety Analysis Report which is a CAQ. As of May 22, 2015, the licensee had not corrected the CAQ in a reasonable time period. Instead, the licensee created action tracking items to develop a plan to correct the CAQ, and the associated due date was extended at least eight times.

This violation is associated with a Green Significance Determination Process finding.

Pursuant to the provisions of 10 CFR 2.201, Exelon Generation Company, LLC, is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, Region III; and the NRC Resident Inspector at the Byron Station, Units 1 and 2, within 30 days of the date of the letter transmitting this Notice. This reply should be clearly marked as a "Reply to a Notice of Violation; VIO 05000454/2015008-09; 05000455/2015008-09," and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation or severity level; (2) the corrective steps that have been taken and the results achieved; (3) the corrective steps that will be taken; and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from ADAMS, accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html, to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

In accordance with 10 CFR 19.11, you may be required to post this Notice within two working days of receipt.

Dated this 21 day of July, 2015.

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-454; 50-455
License No:	NPF-37; NPF-66
Report No:	05000454/2015008; 05000455/2015008
Licensee:	Exelon Generation Company, LLC
Facility:	Byron Station, Units 1 and 2
Location:	Byron, IL
Dates:	April 20, 2015, through June 16, 2015
Inspectors:	N. Féliz Adorno, Senior Reactor Inspector, Lead B. Palagi, Senior Operations Engineer D. Betancourt Roldán, Reactor Inspector, Mechanical M. Jones, Reactor Inspector, Mechanical A. Greca, Electrical Contractor J. Leivo, Electrical Contractor
Approved by:	Christine A. Lipa, Chief Engineering Branch 2 Division of Reactor Safety

SUMMARY

Inspection Report 05000454/2015008; 05000455/2015008, 4/20/2015 – 6/16/2015; Byron Station, Units 1 and 2; Component Design Bases Inspection.

The inspection was a 3-week on-site baseline inspection that focused on the design of components. The inspection was conducted by four regional engineering inspectors, and two consultants. Seven Green findings were identified by the team. Six of these findings were considered Non-Cited Violations of U.S. Nuclear Regulatory Commission (NRC) regulations while one of these findings was considered a Notice of Violation of NRC regulations. The significance of inspection findings is indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red), and determined using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," dated April 29, 2015. Cross-cutting aspects are determined using IMC 0310, "Aspects Within the Cross-Cutting Areas" effective date December 4, 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.

NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green: The team identified a finding of very-low safety significance (Green), and an associated cited violation of Title 10, *Code of Federal Regulations* (CFR), Part 50, Appendix B, Criterion XVI, "Corrective Actions," for the failure to correct a Condition Adverse to Quality (CAQ). Specifically, on June 15, 2012, the U.S. Nuclear Regulatory Commission (NRC) issued a Non-Cited Violation (NCV) for the failure to provide means to detect and isolate a leak in the Emergency Core Cooling System (ECCS) within 30 minutes as described in the Updated Final Safety Analysis Report (UFSAR), which is a CAQ. As of May 22, 2015, the licensee had not corrected the CAQ. This violation is being cited because the licensee had not restored compliance, or demonstrated objective evidence of plans to restore compliance in a reasonable period following the identification of the CAQ. The licensee captured this finding into their Corrective Action Program (CAP) to promptly restore compliance.

  The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of procedure quality, and affected the cornerstone objective of ensuring the availability, reliability, and capability of mitigating systems to respond to initiating events to prevent undesirable consequences. In addition, it was associated with the Barrier Integrity cornerstone attribute of procedure quality, and affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. The finding screened as very-low safety significance (Green) because it did not result in the loss of operability or functionality, and it did not represent an actual pathway in the physical integrity of reactor containment. Specifically, the licensee reasonably demonstrated that an ECCS leak could be detected and isolated before it could adversely affect long-term cooling of the plant. The team determined that the associated finding had a cross-cutting aspect in the area of human performance because the licensee did not use a consistent and systematic approach to make decisions. Specifically, the creation and management of the associated corrective action assignments were not consistent with the instructions contained in their CAP procedure. [H.13] (Section 4OA2.1.b(1))

- Severity Level IV. The team identified a Severity Level IV NCV of 10 CFR 50.59(d)(1), "Changes, Tests, and Experiments," and an associated finding of very-low safety significance (Green) for the licensee's failure to perform a written safety evaluation that provided the bases for the determination that a change which resulted in the sharing of the refueling water storage tanks (RWSTs) of both reactor units did not require a license amendment. Specifically, the licensee did not evaluate the adverse effect of reducing reactor unit independence. The licensee captured this issue into their CAP with a proposed action to revise the associated calculation to remove the dependence on the opposite unit, and/or review the implications of crediting the opposite unit RWST under their 10 CFR 50.59 process.

  The performance deficiency was more than minor because it was associated with the Mitigating Systems cornerstone attribute of design control, and affected the cornerstone objective of ensuring the availability, reliability, and capability of mitigating systems to respond to initiating events to prevent undesirable consequences. In addition, it was associated with the Barrier Integrity cornerstone attribute of design control, and affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. In addition, the associated traditional enforcement violation was more than minor because the team could not reasonably determine that the changes would not have ultimately required NRC prior approval. The finding screened as very-low safety significance (Green) because it did not result in the loss of operability or functionality, and it did not represent an actual open pathway in the physical integrity of the reactor containment. Specifically, the licensee reviewed the affected calculation and reasonably determined that enough conservatism existed such that adequate net positive suction head (NPSH) could be maintained without sharing the RWSTs of both reactor units. The team did not identify a cross-cutting aspect associated with this finding because it was confirmed not to be reflective of current performance due to the age of the performance deficiency. (Section 1R21.5.b(1))

- Green. The team 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 licensee's failure to translate applicable design basis into Technical Specifications (TSs) Surveillance Requirement 3.5.4.2 implementing procedures. Specifically, these procedures did not verify the RWST vent line was free of ice blockage at the locations, and during all applicable MODEs of reactor operation assumed by the ECCS and containment spray (CS) pump NPSH calculation. The licensee captured this issue into their CAP to reconcile the affected procedures and calculation.

  The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of design control, and affected the cornerstone objective of ensuring the availability, reliability, and capability of mitigating systems to respond to initiating events to prevent undesirable consequences. Additionally, it was associated with the Barrier Integrity cornerstone attribute of design control, and affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. The finding screened as very-low safety significance (Green) because it did not result in the loss of operability or functionality, and it did not represent an actual open pathway in the physical integrity of reactor containment. Specifically, the licensee performed a historical review of the last 3 years of operation, and did not find any instances in which the vent path temperature fell below 35 degrees Fahrenheit.

The inspectors did not identify a cross-cutting aspect associated with this finding because it was confirmed not to be reflective of current performance due to the age of the performance deficiency. (Section 1R21.5.b(2))

- Severity Level IV. The team identified a Severity Level IV NCV of 10 CFR 50.59(d)(1), "Changes, Tests, and Experiments," and an associated finding of very-low safety significance (Green) for the licensee's failure to perform a written evaluation that provided the bases for the determination that the changes to the emergency service water cooling tower (SXCT) tornado analysis as described in the UFSAR did not require a license amendment. Specifically, the associated 10 CFR 50.59 Evaluation did not address the introduction of a new failure mode, the resulting loss of heat removal capacity during worst postulated conditions, and addition of operator actions that have not been demonstrated can be completed within the required time to restore the required SXCT heat removal capacity during worst case conditions. The licensee captured this issue in their CAP with a proposed action to revise the 10 CFR 50.59 Evaluation and submit a Licensee Amendment Request.

  The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of protection against external events, and affected the cornerstone objective of ensuring the availability, reliability, and capability of mitigating systems to respond to initiating events to prevent undesirable consequences. In addition, the associated tradition enforcement violation was determined to be more than minor because the team could not reasonably determine that the changes would not have ultimately required prior NRC approval. The finding screened as of very-low safety significance (Green) using a detailed evaluation because a loss of SXCT during a tornado event would degrade one or more trains of a system that supports a risk-significant system or function. The bounding change to the core damage frequency was less than 5.4E-8/year. The team did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance due to the age of the performance deficiency. (Section 1R21.5.b(3))

- Green. The team identified a finding of very-low safety significance and an associated NCV of TS 5.4, "Procedures," for the failure to maintain emergency operating procedures (EOPs) for transfer to cold leg recirculation. Specifically, the EOPs for transfer to cold leg recirculation did not contain instructions for transferring the ECCS and CS systems to the recirculation mode that ensured prevention of potential pump damage when the RWST is emptied. The licensee captured this finding into their CAP to create a standing order instructing operators to secure all pumps aligned to the RWST when it is emptied, and implement long term corrective actions to restore compliance.

  The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of procedure quality, and affected the cornerstone objective of ensuring the availability, reliability, and capability of mitigating systems to respond to initiating events to prevent undesirable consequences. In addition, it was associated with the Barrier Integrity cornerstone attribute of procedure quality, and affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. The finding screened as of very-low safety significance (Green) because it did not result in the loss of operability or functionality of mitigating systems, represent an actual open pathway in the physical integrity of reactor containment, and

involved an actual reduction in function of hydrogen igniters in the reactor containment. Specifically, the incorrect caution would only be used in the event that transfer to sump recirculation was not completed prior to reaching tank low-level, or if the RWST suction isolation valves fail to close. With respect to transfer to sump recirculation prior to reaching tank low-level, a review of simulator test results reasonably determined that operators reliably complete the transfer to sump recirculation prior to reaching this set point. With respect to the failure of the RWST suction isolation valves, a review of quarterly test results reasonably determined the valves would have isolated the tank when required. The team did not identify a cross-cutting aspect associated with this finding because it was not confirmed to reflect current performance due to the age of the performance deficiency. (Section 1R21.6.b(1))

- Green. The team identified a finding of very-low safety significance (Green), and an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure to make an operability determination without relying on the use of probabilistic tools. Specifically, an operability evaluation for an SXCT degraded condition used probabilities of occurrence of tornado events which was contrary to the requirements of the licensee procedure established for assessing operability of structures, systems, and components (SSCs). The licensee captured the team's concern in their CAP to revise the affected operability evaluation without using probability of occurrence of tornado events.

  The performance deficiency was more than minor because it was associated with the Mitigating Systems cornerstone attribute of protection against external events, and affected the cornerstone objective of ensuring the availability, reliability, and capability of mitigating systems to respond to initiating events to prevent undesirable consequences. The finding screened as of very-low safety significance (Green) using a detailed evaluation because a loss of SXCT during a tornado event would degrade one or more trains of a system that supports a risk-significant system or function. The bounding change to the core damage frequency was less than 5.4E-8/year. The team determined that this finding had a cross-cutting aspect in the area of human performance because the licensee did not ensure knowledge transfer to maintain a knowledgeable and technically competent workforce. Specifically, the licensee did not ensure personnel were trained on the prohibition of the use of probabilities of occurrence of an event when performing operability evaluations, which was contained in licensee procedure established for assessing operability of SSCs. [H.9] (Section 4OA2.1.b(3))

Cornerstone: Barrier Integrity

- Green. The team identified a finding of very-low safety significance, and an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure to have procedures to maintain the accuracy within necessary limits of the instrument loops used to verify compliance with the containment average air temperature TS limit of 120 degrees Fahrenheit. Specifically, in 2007, the licensee cancelled the periodic preventive maintenance (PM) intended to maintain the necessary instrument loops accuracy. The licensee entered this issue into their CAP and reasonably established that the 120 degrees Fahrenheit limit was not exceeded by reviewing applicable historical records from 2002 to time of this inspection.

The performance deficiency was determined to be more than minor because it was associated with the configuration control attribute of the Barrier Integrity Cornerstone, and adversely affected the cornerstone objective to ensure that physical design barriers protect the public from radionuclide releases caused by accidents or events. The finding screened as very-low safety significance (Green) because it did not represent an actual open pathway in the physical integrity of reactor containment or involved an actual reduction in hydrogen igniter function. Specifically, the containment integrity remained intact and the finding did not impact the hydrogen igniter function. The team determined that this finding had a cross-cutting aspect in the area of problem identification and resolution because the licensee did not identify issues completely and accurately in accordance with the CAP. Specifically, on January 15, 2015, the licensee captured the lack of periodic PM activities for the containment air temperature instrument loops in the CAP. However, the licensee failed to completely and accurately identify the issue in that it was not treated as a CAQ. As a consequence, no corrective actions were implemented. [P.1] (Section 4OA2.1.b(2))

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 (CDBI) 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 team used information contained in the licensee's PRA and the Byron Station, Units 1 and 2, Standardized Plant Analysis Risk (SPAR) Model to identify two scenarios to use as the basis for component selection. The scenarios selected were a feed and bleed of the reactor coolant system (RCS), and a loss of ultimate heat sink (UHS). Based on these scenarios, a number of risk-significant components, including those with Large Early Release Frequency (LERF) implications, were selected for the inspection.

The team also used additional component information such as a margin assessment in the selection process. This design margin assessment considered original design margin reductions caused by design modification, 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 actions, 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.

The team also identified procedures and modifications for review that were associated with the selected components. In addition, the team selected operating experience issues associated with the selected components.

This inspection constituted 16 samples (12 components, of which 3 had LERF implications, and 4 operating experience) as defined in Inspection Procedure 71111.21-05.

.3 Component Design

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report (UFSAR), Technical Specification (TS), design basis documents, drawings, calculations and other available design basis information, to determine the performance requirements of the selected components. The team used applicable industry standards, such as the American Society of Mechanical Engineers Code, and Institute of Electrical and Electronics Engineers Standards, 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 Information Notices (INs). The review verified 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 were consistent with the design bases and appropriate may have included 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 team reviewed the maintenance history, PM activities, system health reports, operating experience-related information, vendor manuals, electrical and mechanical drawings, and licensee corrective action 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 12 components (samples) were reviewed:

- Safety Injection Pump (1SI01PB): The team reviewed analyses associated with inadvertent safety injection (SI) actuation and hydraulic calculations to assess the pump capability to provide its required accident mitigation function. The reviewed hydraulic analyses included pump minimum required flow, runout flow, flow capacity/balance, minimum required net positive suction head (NPSH), and air entraining vortices. In addition, the team reviewed a sample of operating procedures associated with pump operation under normal and accident conditions to assess their consistency with applicable design basis analyses. The team also reviewed test procedures and completed surveillance tests, including quarterly and comprehensive in-service testing and flow balances, to assess the associated acceptance criteria and test results. The team also reviewed the supporting electrical calculations associated with performance of the SI pump under design basis conditions. This included review of brake horsepower requirements for the pump motor, performance under degraded voltage conditions, and motor protection to assess the capability of the motor to perform its safety function under design basis conditions. In addition, the team

reviewed voltage drop calculations to assess the availability of direct current (DC) control voltage at the associated bus needed to operate the pump circuit breaker. The team also performed a non-intrusive visual inspection of the component to assess overall material condition, configuration, and potential vulnerabilities to hazards. To assess operating trends and the licensee's ability to evaluate and correct problems, the team reviewed system health reports, selected corrective action documents, and PM procedures and records.

- Pressurizer Power-Operated Relief Valve (1RY456): The team reviewed the pressure and temperature limit report and calculations associated with the power-operated relief valve (PORV) lift settings, relief capacity, and set points for low-temperature overpressure (LTOP) scenarios to assess the PORV capability to provide its RCS overpressure protection function. The team also reviewed test procedures and completed surveillances to assess the associated acceptance criteria and test results. In addition, the team reviewed a sample of associated operating procedures to assess their consistency with applicable design basis analyses. The team also reviewed the schematic diagrams for the PORV control circuit to assess its suitability for bleed-and-feed operation as prescribed by operating procedures, and to assess the pilot solenoid and position limit switches qualification for post-accident environmental conditions. The team reviewed voltage drop calculations to assess the availability of the voltage needed at the solenoid valve to operate the PORV. The team also reviewed control wiring schematics and associated instrument loop diagrams to assess the consistency between operations and system design requirements. This review included a circuit protection evaluation intended to demonstrate that the containment electrical penetration was not adversely affected by in-containment faults. The team also reviewed documentation associated with environmental qualifications for the postulated containment accident conditions and replacement of components susceptible to aging. The team reviewed system health reports, selected corrective action documents, and PM procedures and records to assess operating trends and the licensee's ability to evaluate and correct problems.

- Power-Operated Relief Valve Accumulator (1RY32MB): The team reviewed the accumulator sizing calculation, PORV pressure set point, accumulator stress analysis, and maximum allowed accumulator leak rate to assess the accumulator capability to supply the required amount of air pressure and volume to stroke open its associated PORV on a loss of normal air supply. Additionally, the team reviewed the design calculation that established the minimum number of PORV strokes required during certain events, such as LTOP and natural circulation cooldown. The team also reviewed test procedures and completed surveillances to assess the associated acceptance criteria and test results. In addition, the team reviewed a sample of associated operating procedures to assess their consistency with applicable design basis analyses. Finally, the team reviewed system health reports, selected corrective action documents, and recent modifications and operability evaluations to assess operating trends and the licensee's ability to evaluate and correct problems.

- Refueling Water Storage Tank (1SI01T): The team reviewed a sample of associated operating procedures under normal and emergency conditions to assess their consistency with applicable design basis analyses. The team also performed a non-intrusive visual inspection of the refueling water storage

tank (RWST) to assess overall material condition, configuration, and potential vulnerabilities to hazards. To assess operating trends, component health, and the licensee's ability to evaluate and correct problems, the team reviewed system health reports, selected corrective action documents, and recent modifications. The team reviewed design analyses associated with the ability of the RWST system to maintain its design function during external events such as tornados and earthquakes. Additionally, the team reviewed design calculations related to level set points, temperature limits, and minimum required RWST volume to mitigate a loss of coolant accident (LOCA), and to support feed-and-bleed scenarios. The team also reviewed the schematic diagrams and instrument uncertainty calculations to assess the low-low RWST level signal (i.e., LO-2) capability to automatically open the containment sump isolation valves (i.e., 1SI8811A/B) following a LOCA, and its consistency with the associated set point calculation including instrument uncertainty considerations. To assess operating trends, component health, and the licensee's ability to evaluate and correct problems, the team reviewed system health reports, selected corrective action documents, recent modifications, and PM/calibration procedures and records.

- Emergency Service Water Makeup Pump (0SX02PA): The team reviewed design documents and procedures to assess consistency with vendor specifications. The team reviewed calculations associated with pump capability and performance to assess the pump capability to perform its design function of providing sufficient inventory to the associated Emergency Service Water Cooling Tower (SXCT) basin under different postulated scenarios. The team reviewed the water inventory availability from the suction source under routine service as well as extreme conditions. This review included low and high-river water levels and temperatures, pump NPSH, pump suction submergence, and minimum flow protection. The team also reviewed procedures associated with protection against flooding, seismic, and tornado events since the makeup pump is credited to some extent during these postulated events. The team also performed a non-intrusive visual inspection of the pump to assess overall material condition, configuration, and potential vulnerabilities to hazards. Work orders and maintenance procedures were reviewed to verify effectiveness of site maintenance. The team also reviewed test procedures and completed surveillances to assess the associated acceptance criteria and test results. To assess operating trends, component health, and the licensee's ability to evaluate and correct problems, the team reviewed system health reports and selected corrective action documents.

- Emergency Service Water Makeup Pump Diesel Engine (0SX02PA-K): The team reviewed design documents and procedures to assess consistency with vendor specifications. The team reviewed diesel fuel oil day tank level alarm response procedures and sizing analyses including the engine diesel fuel oil consumption rate calculation, tank capacity, vortexing calculation, level indicators, and alarm setpoint. In addition, the team reviewed the control circuit electrical diagram to assess the consistency between operations and design basis requirements. The team also reviewed the set point calculation for the SXCT basin level switch associated with the starting logic of the diesel engine to assess consistency between the specified setting and applicable design basis requirements. In addition, the team reviewed recent level instrument calibration

results. The team also reviewed circuit protection and control voltage to assess the diesel engine capability to start on demand. The inspectors reviewed completed work orders to assess the as-found and as-left condition of the diesel engine following recent maintenance activities. The team also reviewed test procedures and completed surveillances to assess the associated acceptance criteria and test results. The team also performed a non-intrusive visual inspection of the engine to assess overall material condition, configuration, and potential vulnerabilities to hazards. To assess operating trends and the licensee's ability to evaluate and correct problems, the team reviewed system health reports, selected corrective action documents, modifications, and PM procedures and records.

- Emergency Service Water Cooling Tower (0SX02AA/B and 0SX03CA/H): The team reviewed design calculations and procedures associated with fan performance, basin sizing, heat transfer, and makeup requirements during postulated events including LOCA, tornado, and seismic events. The electrical calculations associated with fan performance under design basis conditions were reviewed to assess consistency with the design bases and the motor capability to perform its specified safety function. This review considered fan motor brake horsepower requirements, performance under degraded voltage conditions, and motor protection. The team reviewed voltage drop calculations to assess the availability of the DC control voltage needed at the associated load center for the closing and tripping of the cooling tower fan circuit breakers. The team also reviewed the alternating current (AC) and DC electrical distribution systems to assess the SXCT capability to perform its specified safety function assuming a single failure of electrical components. The team also reviewed control wiring diagrams of the deep well pump and associated control valves to assess consistency between their operation and design requirements. The team also performed a non-intrusive visual inspection of the SXCT basin structure, fan motors, valve houses, and electrical equipment rooms to assess overall material condition, configuration, and potential vulnerabilities to hazards. The team also reviewed test procedures and completed surveillances to evaluate the associated acceptance criteria and test results. To assess operating trends and the licensee's ability to evaluate and correct problems, the team reviewed system health reports, selected corrective action documents, operability evaluations, modifications, and PM procedures and records.

- 4160 Volts Alternating Current Bus 142: The team reviewed voltage drop calculations to assess the availability of the DC control voltage needed at the associated bus for the operation of the associated circuit breakers. The team reviewed calculations associated with load flow, degraded voltage, and protective settings for selected electrical load paths served by the bus and associated with the inspection samples to assess the bus capability to support the loads required safety functions under design basis conditions. The team also performed a non-intrusive visual inspection of the switchgear to assess overall material condition, configuration, and potential vulnerabilities to hazards or extreme service environments. To assess operating trends and the licensee's ability to evaluate and correct problems, the team reviewed system health reports, selected corrective action documents, and selected PM procedures and records.

- 120 Volts Alternating Current Instrument Bus 111: The team reviewed the DC voltage drop calculations to assess the availability of the voltage needed for the proper operation of the associated inverter, including during a loss of AC power. The team also reviewed the bus loading and breaker ratings to assess the bus and loads protection against spurious tripping. In addition, the team reviewed a modification which installed forced air cooling units for the inverter serving the bus to assess the modification implementation and any potential impact on the inverter. To assess operating trends and the licensee's ability to evaluate and correct problems, the team reviewed system health reports, selected corrective action documents, and PM procedures and records for the bus.

- 125 Volts Direct Current Bus 111: The team reviewed bus loading and short circuit calculations as well as cable, bus, and circuit breaker ratings to assess bus and cable capabilities of carrying the maximum anticipated loading and protection against faulted conditions. The team also reviewed voltage drop and battery sizing calculations to assess the capability to support momentary and continuous loading for the duration of the duty cycle during accident conditions and the loss of all AC power (i.e., station blackout). Additionally, the team reviewed the battery charger sizing calculation to assess its capability of maintaining the battery in a charged state and recharging the battery in a timely manner following a loss of AC power event. The team also reviewed room heat-up calculations to ensure that the DC components were not adversely affected by steam line breaks in the turbine building. In addition, the team reviewed purchase specifications, vendor documents, seismic test reports, certificate of compliance, and cable separation to assess consistency of the installed component to the design requirements. For the battery, this review included an assessment of the inter-cell resistance conformance to voltage drop calculations. Breaker/fuse coordination was also reviewed to assess the capability to interrupt overloads and faulted conditions. The team also reviewed testing procedures and associated recent results, recent system health reports, molded-case circuit breaker testing, maintenance activities, and recent corrective action documents to assess component health history.

- 24 Volts Direct Current Bus 035-2: The team reviewed the sizing calculation for the diesel start system and the control batteries to assess their capability of providing adequate voltage to the associated components for the duration of the duty cycle during accident conditions and loss of all AC power. The team also reviewed components and wiring schematics related to the diesel start and control logic to assess the bus capability to perform its intended function. Additionally, the team reviewed the battery charger sizing calculation to assess its capability to maintain the batteries in a charged state, and to recharge them in a timely manner following a loss of AC power event. The team reviewed purchase specifications, vendor documents, seismic test report, and certificate of conformance to assess consistency of the installed component to the design requirements. The team also reviewed testing procedures and associated recent results, health reports, maintenance activities, and recent corrective action documents to assess component health history.

- 480 Volts Alternating Current Motor Control Center 132Z1: The team assessed conformance to the applicable design and licensing basis by performing an engineering review of the motor control center (MCC) loading, MCC and control

circuits degraded voltage and maximum voltage, electrical protection, and electrical isolation/physical circuit separation of the MCC from non-safety class loads. The loads considered during this review were the SXCT riser motor operated valves (MOVs) (i.e., 0SX163E/F), SXCT makeup MOV (i.e., 0SX157A), and basin bypass MOV (i.e., 0SX162B). The team reviewed the calculations that determined minimum terminal voltages for these MOVs to assess consistency with the associated MOV thrust calculations. The team also reviewed the thermal overload sizing calculations for these MOV circuits to assess their protection against premature thermal overload trip and the minimum voltage calculations for the 120 volts alternating current (VAC) service to the SXCT basin level control system to assess the availability of the voltage needed for the level instrumentation under design basis conditions. To evaluate whether there were adverse operating trends and to assess the licensee's ability to evaluate and correct problems, the team reviewed system health reports, selected corrective action documents, and PM procedures and records for the MCC.

b. Findings

(1) Question Regarding the Maximum Wet Bulb Temperature Value Assumed in the Emergency Service Water Cooling Tower Tornado Analysis

Introduction: The team identified an unresolved item (URI) regarding the maximum wet-bulb temperature value assumed in the SXCT tornado analysis. Specifically, the team noted the analysis used a value which was less restrictive than the highest 3-hour wet-bulb temperature recorded for the site as described in the UFSAR.

Description: In Section 3.5.4 of the UFSAR, "Analysis of Missiles Generated by a Tornado," stated that, "An analysis of the UHS cooling capability for a tornado missile event has been made." It also stated that, "A maximum outside air wet-bulb temperature of 78 degrees Fahrenheit is assumed and is conservatively held constant throughout the transient." In addition, this UFSAR section stated that, "The analysis was performed using service water cooling tower performance curves generated using the method described in UFSAR Section 9.2.5.3.1.1.2 [...]." The analysis of the UHS cooling capability for a tornado missile event was calculation BYR09-002, "UHS Capability with Loss of SX [Emergency Service Water] Fans due to a Tornado Event," which used a constant maximum outside air wet-bulb temperature value of 78 degrees Fahrenheit consistent with UFSAR Section 3.5.4.

However, the team noted the assumed maximum outside air wet-bulb temperature value of 78 degrees Fahrenheit appeared to be inconsistent with the method described in UFSAR Section 9.2.5.3.1.1.2, "Steady State Tower Performance Analysis." Specifically, it stated that, "The design wet-bulb temperature during warm weather operation is 82 degrees Fahrenheit (Refer to UFSAR Section 2.3.1.2.4)." In Section 2.3.1.2.4 of the UFSAR, "Ultimate Heat Sink Design," stated that, "This analysis [described in Section 9.2.5.3.1.1] includes scenarios with the highest 3-hour wet-bulb temperature, 82 degrees Fahrenheit, which was recorded on July 30, 1961, at 3:00 pm." This UFSAR section also stated that, "Per Regulatory Guide 1.27, the ultimate heat sink must be capable of performing its cooling function during the design basis event for this worst case 3-hour wet-bulb temperature." In addition, it stated, "However, the design operating wet-bulb temperature of the ultimate heat sink is 78 degrees Fahrenheit (ASHRAE 1 percent exceedance value)."

This issue is unresolved pending further review by the Office of Nuclear Reactor Regulation (NRR) of the licensing basis related to the wet-bulb temperature value applicable for the SXCT tornado analysis, and the team determination of further NRC actions to resolve the issue. (URI 05000454/2015008-01; 05000455/2015008-01, Question Regarding the Maximum Wet-Bulb Temperature Value Assumed in the SXCT Tornado Analysis)

(2) Maximum Wet-Bulb Temperature Value Assumed in Emergency Service Water Cooling Tower Analysis Was Not Monitored

Introduction: The team identified an URI regarding the lack of monitoring the maximum wet-bulb temperature value assumed in SXCT analysis. Specifically, the team noted the maximum wet-bulb temperature value was a critical parameter for the SXCT analyses, but the licensee had not established a testing program to verify actual values were bounded.

Description: In Section 3.5.4 of the UFSAR, "Analysis of Missiles Generated by a Tornado," stated that, "An analysis of the UHS cooling capability for a tornado missile event has been made." It also stated that, "A maximum outside air wet-bulb temperature of 78 degrees Fahrenheit is assumed, and is conservatively held constant throughout the transient."

In Section 9.2.5.3.1.1 of the UFSAR, "Design Basis Reconstitution," stated that, "The design basis event for the Byron ultimate heat sink is a LOCA coincident with a loss-of-off-site power (LOOP) in one unit, and the concurrent orderly shutdown from maximum power to cold shutdown of the other unit using normal shutdown operating procedures." It also stated that, "The design wet-bulb temperature during warm weather operation is 82 degrees Fahrenheit (Refer to the UFSAR Section 2.3.1.2.4)." In Section 2.3.1.2.4 of the UFSAR, "Ultimate Heat Sink Design," stated that, "This analysis [described in Section 9.2.5.3.1.1] includes scenarios with the highest 3-hour wet-bulb temperature, 82 degrees Fahrenheit, which was recorded on July 30, 1961, at 3:00 pm."

The analysis of the UHS cooling capability for a tornado missile event was calculation BYR09-002, "UHS Capability with Loss of SX Fans due to a Tornado Event," which used a constant maximum outside air wet-bulb temperature value of 78 degrees Fahrenheit consistent with UFSAR Section 3.5.4. The analysis of the UHS cooling capability for a LOCA coincident with a LOOP was calculation UHS-01, "Ultimate Heat Sink Design Basis LOCA Single Failure Scenarios," which used a constant maximum outside air wet-bulb temperature value of 82 degrees Fahrenheit consistent with the UFSAR Section 9.2.5.3.1.1.

However, the licensee had not established a testing program to verify actual environmental conditions were bounded by these analyses and design basis limits. In response to the team questions, the licensee stated that this approach was acceptable because historical data showed wet-bulb temperature had a cyclic nature, maximum wet-bulb temperature lasted for relatively short durations, and the analyses assumed constant wet-bulb temperature values.

This issue is unresolved pending further NRR review of the acceptability of the licensee approach to ensure the SXCT analyses bounded actual environmental conditions, and the team determination of further NRC actions to resolve the issue. (URI 05000454/2015008-02; 05000455/2015008-02, Maximum Wet-Bulb Temperature Value Assumed in SXCT Analysis Was Not Monitored)

.4 Operating Experience

a. Inspection Scope

The team reviewed four 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 2013-05, "Battery Expected Life and Its Potential Impact on Surveillance Requirements;"
- IN 2010-26, "Submerged Electrical Cables;"
- IN 2013-12, "Improperly Sloped Instrument Sensing Lines;" and
- IN 2012-01, "Refueling Water Storage Tank Degradation."

b. Findings

No findings were identified.

.5 Modifications

a. Inspection Scope

The team reviewed five 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:

- Engineering Change (EC) 385951, "Multiple Spurious Operation – Scenario 14, 1SI8811A/B;"
- EC396016, "Increase U1 Pressurizer PORV Accumulator Tank Operating Pressure to Increase Number of PORV Open/Close Cycles from Accumulator;"
- EC388735, "Detailed Review of the FC Purification for Use of Non-Safety Related Portion Connected to Safety Related Piping;"
- DRP 11-052, "Clarify References to RWST Internal Pressure in the ECCS and the CS Pumps NPHS Analysis;" and
- EC385829, "Tornado Missile Design Basis for the Essential Service Water Cooling Towers."

b. Findings

(1) Failure to Evaluate the Adverse Effects of Sharing the Refueling Water Storage Tanks of Both Reactor Units

Introduction: The team identified a Severity Level IV NCV of 10 CFR 50.59(d)(1), "Changes, Tests, and Experiments," and an associated finding of very-low safety significance (Green) for the licensee's failure to perform a written safety evaluation that provided the bases for the determination that a change which resulted in the sharing of the RWSTs of both reactor units did not require a license amendment. Specifically, screening 6E-05-0172, "UFSAR Change Package (DRP) 11-052," did not address the reduction in reactor unit independence associated with sharing the RWSTs air space of both reactor units.

Description: Each reactor unit has one RWST, which supplies borated water to both trains of the Emergency Core Cooling System (ECCS) and Containment Spray (CS) systems during the injection phase of a LOCA recovery. The UFSAR Section 6.3, "Emergency Core Cooling System," and UFSAR Section 6.5.2, "Containment Spray Systems," described the NPSH analyses for the ECCS and CS pumps when their suctions are aligned to their associated RWST. Before November 16, 2005, these UFSAR sections described the RWST as being under atmospheric pressure during the injection mode. The licensee changed these UFSAR descriptions following the discovery that the RWST would not be under atmospheric pressure because the RWST vent did not have the capacity to prevent vacuum during the high outflow expected during the injection phase, and the vent vacuum relief device was not safety related. This discovery was captured in the CAP as AR00239280.

The licensee reviewed this UFSAR change in Title 10, *Code of Federal Regulations* (CFR), Part 50.59 screening 6E-05-0172, "Clarify References to RWST Internal Pressure in the ECCS and CS Pumps NPSH Analysis." The screening concluded that the change did not require a 10 CFR 50.59 safety evaluation and, consequently, NRC prior approval because the change did not result in an adverse effect to the ECCS and CS systems. Specifically, the licensee determined the expected vacuum would not affect the structural integrity of the tank. In addition, the licensee determined in calculation BYR 04-016, "[Residual Heat Removal] RHR, SI, [Chemical and Volume Control] CV, and CS Pump NPSH during ECCS Injection Mode," that the available NPSH for the pumps while taking suction from the RWST remained adequate when considering the expected vacuum.

However, the team noted that revised calculation BYR 04-016 credited the entire RWST vent line, which was common to the RWSTs of both reactor units. Consequently, the change credited the free air space of both tanks to mitigate the vacuum expected during tank drawdown. The team also noted that UFSAR Section 3.1.2.1.5, "Evaluation Against Criterion 5 – Sharing of Structures, Systems, and Components," described those SSCs important to safety shared by the two reactor units, and the RWSTs were not included as shared SSCs. Thus, the team noted the licensee implemented a change to the facility as described in the UFSAR that resulted in a reduction of reactor unit independence. Changes to the facility as described in the UFSAR that reduce reactor unit independence adversely impact 10 CFR 50.59 change evaluation criteria because they result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety. Since the licensee failed to appropriately

evaluate this adverse effect in a 10 CFR 50.59 safety evaluation, the team could not reasonably determine that the change would not have ultimately required NRC prior approval.

The licensee captured this issue in their CAP as AR 02496142. The corrective actions considered at the time of this inspection were to revise calculation BYR04-016 to not credit the opposite unit RWSTs air space and/or revise 10 CFR 50.59 screening 6E-05-0172 to consider the implications of crediting the opposite unit RWST air space.

The team also noted the licensee did not correctly implement this change into associated surveillance procedures intended to verify RWST operability. This separate concern is discussed in detail in Section 1R21.5.b(2) of this report.

Analysis: The team determined that the failure to provide a written evaluation that provided the bases for the determination that a change which resulted in the sharing of the RWSTs of both reactor units did not require a license amendment, was contrary to the requirements of 10 CFR 50.59(d)(1), and was a performance deficiency. The performance deficiency was more than minor because it was associated with the Mitigating Systems cornerstone attribute of design control, and affected the cornerstone objective of ensuring the availability, reliability, and capability of mitigating systems to respond to initiating events to prevent undesirable consequences. In addition, it was associated with the Barrier Integrity cornerstone attribute of design control, and affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the change did not ensure the RWST capability to support ECCS and CS mitigating and barrier functions because it eliminated the capability to achieve the RWST supporting function while maintaining separation of the reactor units.

In addition, the associated violation was determined to be more than minor because the team could not reasonably determine the changes would not have ultimately required NRC prior approval.

Violations of 10 CFR 50.59 are dispositioned using the traditional enforcement process instead of the Significance Determination Process (SDP) because they are considered to be violations that potentially impede or impact the regulatory process. This violation is associated with a finding that has been evaluated by the SD, and communicated with an SDP color reflective of the safety impact of the deficient licensee performance. The SDP, however, does not specifically consider the regulatory process impact. Thus, although related to a common regulatory concern, it is necessary to address the violation and finding using different processes to correctly reflect both the regulatory importance of the violation and the safety significance of the associated finding.

In this case, the team determined that the finding could be evaluated using the SDP in accordance with Inspection Manual Chapter (IMC) 0609, “Significance Determination Process” by using Attachment 0609.04, “Initial Characterization of Findings.” Since the finding impacted the Mitigating Systems and Barrier Integrity cornerstones, the inspectors screened the finding through IMC 0609 Appendix A, “The Significance Determination Process for Findings At-Power,” using Exhibit 2, “Mitigating Systems Screening Questions,” and Exhibit 3, “Barrier Integrity Screening Questions.” The finding screened as very-low safety significance (Green) because it did not result in the loss of operability or functionality, and it did not represent an actual open pathway in the physical integrity of the reactor containment. Specifically, the licensee reviewed

calculation BYR 04-016, and reasonably determined that enough conservatism existed such that adequate NPSH could be maintained without sharing the RWSTs of both reactor units.

In accordance with Section 6.1.d of the NRC Enforcement Policy, this violation is categorized as Severity Level IV because the resulting change was evaluated by the SDP as having very-low safety significance (i.e., Green finding).

The inspectors did not identify a cross-cutting aspect associated with this finding because it was confirmed not to be reflective of current performance. Specifically, the finding occurred approximately 10 years ago.

Enforcement: Title 10 CFR 50.59, "Changes, Tests, and Experiments," Section (d)(1) requires, in part, the licensee to maintain records of changes in the facility, of changes in procedures, and of tests and experiments made pursuant 10 CFR 50.59(c). These records must include a written evaluation which provides the bases for the determination that the change, test, or experiment does not require a license amendment pursuant to Paragraph (c)(2) of this section. Paragraph (c)(2)(ii) states, in part, that a licensee shall obtain a license amendment pursuant to 10 CFR 50.90 prior to implementing a proposed change, test, or experiment if the change, test, or experiment would result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR. In the UFSAR Sections 6.3 and 6.5 describe the NPSH evaluations for ECCS and CS pumps when their suctions are aligned to their associated RWST. Additionally, UFSAR Section 3.1.2.1.5 states that "Those systems, structures, and components important to safety shared by the two units are the ultimate heat sinks and the associated Byron makeup water systems; various heating, ventilating, and air conditioning systems within the shared auxiliary and fuel handling building; and a component cooling heat exchanger which can be valved to serve one unit or the other." The RWSTs are not included as shared SSCs.

Contrary to the above, on November 16, 2005, the licensee failed to maintain a record of a change in the facility made pursuant to 10 CFR 50.59(c) that included a written evaluation which provided the bases for the determination that the change did not require a license amendment pursuant to 10 CFR 50.90(c)(2). Specifically, the licensee changed the ECCS and CS pumps NPSH calculation for their injection mode of operation (i.e., calculation BYR 04-016) to credit the entire vent line common to the RWSTs of both reactor units and, consequently, the free air space of both tanks to mitigate the vacuum expected during tank drawdown. However, the licensee failed to perform a written evaluation that provided the bases for the determination that the change effect of reducing reactor unit independence by sharing their RWSTs did not result in more than a minimal increase in the likelihood of occurrence of a malfunction of the RWSTs and their supported safety systems.

The licensee is still evaluating its planned corrective actions. However, the team determined that the continued non-compliance does not present an immediate safety concern because the licensee reasonably determined that the affected analysis contained enough conservatism such that adequate NPSH could be maintained without sharing the RWSTs of both reactor units.

Because this was a Severity Level IV violation and was entered into the licensee Corrective Action Program (CAP) as AR 02496142, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. (NCV 05000454/2015008-03; 05000455/2015008-03; Failure to Evaluate the Adverse Effects of Sharing the RWSTs of Both Reactor Units)

The associated finding is evaluated separately from the traditional enforcement violation and, therefore, the finding is being assigned a separate tracking number. (FIN 05000454/2015008-04; 05000455/2015008-04; Failure to Evaluate the Adverse Effects of Sharing the RWSTs of Both Reactor Units)

(2) Failure to Adequately Implement a Design Change Associated with the RWSTs

Introduction: The team 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 licensee's failure to translate applicable design basis into TS Surveillance Requirement (SR) 3.5.4.2 implementing procedures. Specifically, these procedures did not verify RWST vent line was free of ice blockage at the locations and during all applicable MODEs of reactor operation assumed by the ECCS and CS pump NPSH calculation.

Description: Each reactor unit has one RWST, which supplies borated water to both trains of the ECCS and CS systems during the injection phase of a LOCA recovery. The TS 3.5.4, "Refueling Water Storage Tank," required the RWSTs to be operable when their associated reactor unit is in MODEs 1, 2, 3, or 4. A vent line is installed at the top of each RWST. The vent lines are routed into the auxiliary building where they connect to a common header which joins to a filtration system. Because the header is common to both vents, the free air spaces of the RWSTs are communicated via their vent lines. The vent line portions located between the tanks and the auxiliary building are exposed to outside ambient conditions. For this reason, TS SR 3.5.4.2 stated, "Verify RWST vent path temperature is ≥35 degrees Fahrenheit." The associated TS Basis explained that "Heat traced portions of the RWST vent path should be verified to be within the temperature limit needed to prevent ice blockage and subsequent vacuum formation in the tank during rapid level decreases caused by accident conditions." The licensee established procedures 1/2 BOSR 01-1,2,3, "Modes 1, 2, and 3 Shiftily and Daily Operating Surveillance," and 1/2 BOSR 01-4, "Mode 4 Shiftily and Daily Operating Surveillance," as the implementing procedures for SR 3.5.4.2.

Originally, the RWSTs design assumed they were atmospheric tanks by crediting their associated vent line capability to prevent vacuum during tank drawdown. However, on November 16, 2005, the licensee implemented a design change to credit the vent lines capability to communicate the free air space of both tanks following the discovery that the RWST vents did not have the capacity to prevent vacuum during the high outflow expected during the injection phase, and the vent vacuum relief devices were not safety related. This discovery was captured in the CAP as AR00239280.

As a result, calculation BYR 04-016, "RHR, SI, CV and CS Pump NPSH during ECCS Injection Mode," credited the vent lines of both RWSTs to mitigate the vacuum expected during the drawdown of one tank during accident conditions. However, the team noted this change was not correctly implemented into procedures 1/2 BOSR 01-1,2,3 and 1/2 BOSR 01-4. Specifically, these procedures were reactor unit specific in that their instructions only required verifying the RWST vent line portions that were associated

with the applicable reactor unit RWST; that is, the portions between the associated RWST and the auxiliary building. As a consequence, the team was concerned because, if one vent line is found to be blocked with ice, the procedures would only recognize one RWST as being inoperable. In addition, the procedures were only implemented when the associated reactor unit was in MODEs 1, 2, 3, or 4 consistent with the applicability requirements of TS 3.5.4. Thus, the team was also concerned that a potentially inoperable condition would not be detected because the procedures would not verify both vent lines were free of ice blockage when one reaction unit is in MODE 5 or 6 while the other reactor unit is in MODE 1, 2, 3, or 4.

The licensee captured the team concerns in their CAP as AR 02496766. The immediate corrective action was to verify that outside air temperatures were not forecasted to fall below 35 degrees Fahrenheit for the foreseeable future. Additionally, the licensee determined the RWSTs remained operable during the last 3 years by performing a historical review which did not find instances in which the vent lines temperature fell below 35 degrees Fahrenheit. The proposed corrective actions to restore compliance at the time of this inspection included revising the applicable calculations to remove dependence on the opposite unit, and/or revising the affected procedures to be consistent with the applicable calculation.

The team also noted the licensee did not perform a written safety evaluation that provided the bases for the determination that this change, which resulted in a reduction of reactor unit independence, did not require a license amendment. This separate concern is discussed in detail in Section 1R21.5.b(1) of this report.

Analysis: The team determined the failure to translate applicable design basis into TS SR 3.5.4.2 implementing procedures was contrary to 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” and was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of design control, and affected the cornerstone objective of ensuring the availability, reliability, and capability of mitigating systems to respond to initiating events to prevent undesirable consequences. Additionally, it was associated with the Barrier Integrity cornerstone attribute of design control, and affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, TS SR 3.5.4.2 implementing procedures were inadequate to verify RWST operability because they did not verify all critical assumptions made by the design calculations. The RWST supports ECCS, which is a mitigating system, and CS, which is part of the physical design barrier.

The team determined the finding could be evaluated using the SDP in accordance with IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Initial Characterization of Findings.” Since the finding impacted the Mitigating Systems and Barrier Integrity cornerstones, the inspectors screened the finding through IMC 0609, Appendix A, “The Significance Determination Process for Findings At-Power,” using Exhibit 2, “Mitigating Systems Screening Questions,” and Exhibit 3, “Barrier Integrity Screening Questions.” The finding screened as very-low safety significance (Green) because it did not result in the loss of operability or functionality, and it did not represent an actual open pathway in the physical integrity of reactor containment. Specifically, the licensee performed a historical review of the last 3 years of operation and did not find any instances in which the vent path temperature fell below 35 degrees Fahrenheit.

The inspectors did not identify a cross-cutting aspect associated with this finding because it was confirmed not to be reflective of current performance due to the age of the performance deficiency. Specifically, the finding occurred approximately 10-years ago.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that design changes, including field changes, be subjected to design control measures commensurate with those applied to the original design.

Contrary to the above, on November 16, 2005, the licensee performed a design change and failed to subject it to design control measures commensurate to those applied to the original design. Specifically, the licensee changed the ECCS and CS pump NPSH calculation for their injection mode of operation (i.e., calculation BYR 04-016) to credit the capability of the vent lines of both RWSTs to support the operability of any one RWST. However, the design control measures failed to correctly translate the new design basis into procedures 1/2 BOSR 01-1,2,3 and 1/2 BOSR 01-4 in that they were not revised to verify the capability of the vent lines of both RWSTs to support the operability of any one RWST.

The licensee is still evaluating its planned corrective actions. However, the team determined that the continued non-compliance does not present an immediate safety concern because outside air temperatures were not forecasted to fall below 35 degrees Fahrenheit for the foreseeable future. Additionally, a corrective action tracking item was created to develop compensatory actions if compliance is not restored prior to the next season when temperatures can potentially decrease below 35 degrees Fahrenheit.

Because this violation was of very-low safety significance and was entered into the licensee's CAP as AR 02496766, this violation is being treated as a NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. (NCV 05000454/2015008-05; 05000455/2015008-05; Failure to Adequately Implement a Design Change Associated with the RWSTs)

(3) Failure to Evaluate the Adverse Effects of Changing the Emergency Service Water Cooling Tower Tornado Analysis as Described in the Updated Final Safety Analysis Report

Introduction: The team identified a Severity Level IV NCV of 10 CFR 50.59(d)(1), "Changes, Tests, and Experiments," and an associated finding of very-low safety significance (Green) for the licensee's failure to perform a written evaluation that provided the bases for the determination that the changes to the SXCT tornado analysis as described in the UFSAR did not require a license amendment. Specifically, 50.59 Evaluation 6G-11-0041, "Tornado Missile Design Basis for the Essential Service Water Cooling Towers," did not address the introduction of a new failure mode, the resulting loss of heat removal capacity during worst postulated conditions, and addition of operator actions that have not been demonstrated can be completed within the required time to restore the required SXCT heat removal capacity during worst case conditions.

Description: During the 2005 NRC Safety Systems Design, Performance and Capability (SSDPC) inspection, the inspectors noted that the UFSAR-described tornado analysis for the SXCT had not been updated to reflect changes that increased the heat load. The SSDPC documented this concern as URI 05000454/2005002-07;

05000455/2005002-07. In 2007, this URI was subsequently closed to NCV 05000454/ 2007004-03; 05000455/2007004-03. As a result, on February 14, 2012, the licensee completed EC 385829, "UHS Capability with Loss of SX Fans Due to Tornado Missiles," to change the UHS tornado missile design basis as described in Revision 7 of the UFSAR. The EC 385829 evaluated these design basis changes in 10 CFR 50.59 safety evaluation 6G-11-004, "Tornado Missile Design Basis for the Essential Service Water Cooling Towers," dated February 9, 2012. This 10 CFR 50.59 safety evaluation concluded that the design basis changes could be implemented without obtaining a license amendment.

However, the team noted that the licensee did not address the adverse effects of the changes in the 10 CFR 50.59 safety evaluation. Specifically, the change reduced the amount of missiles from "multiple" to "single," and changed the SXCT design from natural draft cooling to mechanical draft cooling (i.e., from passive to active system). These changes adversely impacted 10 CFR 50.59 change evaluation criteria because they would result in more than a minimal increase in the likelihood of occurrence of a malfunction of the SXCT during a tornado event. Specifically:

- The change introduced a new failure mode (i.e., fan failures) that was not bounded by the previous analysis. Specifically, Revision 7 of the UFSAR Section 3.5.4, "Analysis of Multiple Missiles Generated by a Tornado," stated that the SXCT fans, fan motors, and fan drives were not protected from tornado missiles. It also stated, "An analysis of cooling tower capacity without fans [emphasis added] has been made." In contrast, this statement was revised to, "An analysis of the UHS cooling capability for a tornado missile event has been made." The new analysis required multiple operating fans to ensure enough cooling capacity to mitigate the effects of a single tornado missile. The fans, fan motors, and fan drives were not modified to add tornado missile protection. In addition, Revision 7 of the UFSAR Section 9.2.5.3.2, "Essential Service Water Cooling Towers," stated "An analysis of the effect of multiple [emphasis added] tornado missiles on the essential service water cooling towers has been performed." This statement was revised to delete the word "multiple." Following this revision, the analysis only considered the effects of one tornado-generated missile.

- Revision 1 of NEI 96-07, "Guidelines for 10 CFR 50.59 Evaluations," which has been endorsed by the NRC in Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," stated, in part, that a change would result in less than a minimal increase in the likelihood of occurrence of an SSC malfunction provided it "…satisfies applicable design basis requirements." In contrast, this change did not satisfy the design basis requirements for protection against natural phenomena as described in the USAR Section 3.1.2.1.2, "Evaluation Against Criterion 2 - Design Bases for Protection Against Natural Phenomena." Specifically, Revision 7 and the revision in effect at the time of this inspection of UFSAR Section 3.1.2.1.2 stated, "The systems, components, and structures important to safety have been designed to accommodate, without loss of capability [emphasis added], effects of the design-basis natural phenomena along with appropriate combinations of normal and accident conditions." However, this change would result in the loss of SXCT capability to perform its safety function during the worst case conditions in that the required number of fans would not be available necessitating operator

actions to delay shutdown cooling initiation until an adequate number of SXCT fans are available to support the shutdown cooling heat load and, consequently, transition to MODE 5 where design basis accidents (DBAs) are not postulated.

- The change involved a new operator action that supports the SXCT function which is not reflected in plant procedures and training programs. Specifically, UFSAR Section 3.5.4 was revised to credit new operator actions "…to delay RHR initiation until an adequate number of SXCT fans are available for shutdown cooling [emphasis added]…" and to stagger RHR initiation for the two units. The revised UFSAR-described analysis assumed "For the worst case design conditions the first unit is assumed to be placed on RHR cooling 24 hours after the event and the second unit at 30 hours after the event." NEI 96-07 states, in part, that a new operator action that supports a design function credited in a safety analysis results in less than a minimal increase in the likelihood of occurrence of an SSC malfunction provided the action is reflected in plant procedures and training programs, and these actions have been demonstrated can be completed in the time required considering the aggregate effects. However, the licensee had not created procedures and training material to restore an adequate number of SXCT fans. In addition, the licensee had not demonstrated that these actions can be completed in the time required considering the aggregate effects, such as the expected conditions when the actions are required.

In addition, the change would create a possibility for an SXCT malfunction with a different result than any previously evaluated in the UFSAR because:

- Nuclear Energy Institute (NEI) 96-07 states, "A malfunction that involves an initiator or failure whose effects are not bounded by those explicitly described in the UFSAR is a malfunction with a different result." In contrast, this change would result in the loss of SXCT capability to perform its safety function during the worst case conditions in that the required number of fans would not be available to support RHR initiation necessitating a delay of RHR initiation until an adequate number of fans are available. The previous UFSAR-described analysis assumed the SXCT design remained capable of performing its safety function during the worst case conditions because it did not require any fans to support RHR initiation and operation; and

- NEI 96-07 stated, "An example of a change that would create the possibility for a malfunction with a different result is a substantial modification… that creates a new or common cause failure that is not bounded by previous analyses or evaluations." In contrast, this change introduced a new failure that was not bounded by previous analysis as previously explained.

The licensee captured the team concern in their CAP as AR 2506214 to request a license amendment. The potential operability implications of this issue are discussed in Section 4OA2.1.b(3) of this report.

Analysis: The team determined that the failure to perform a written evaluation that provided the bases for the determination that the changes to the SXCT tornado analysis as described in the UFSAR did not require a license amendment was contrary to the requirements of 10 CFR 50.59(d)(1) and was a performance deficiency. The

performance deficiency was more than minor because it was associated with the Mitigating Systems cornerstone attribute of protection against external events, and affected the cornerstone objective of ensuring the availability, reliability, and capability of mitigating systems to respond to initiating events to prevent undesirable consequences. Specifically, the change did not ensure the SXCT reliability and availability during and following a tornado event because it introduced a new failure mode, and added reliance on operator actions that have not been demonstrated can be completed in the required time. The change also did not ensure the SXCT capability to perform its safety function during the worst case conditions during and following a tornado event in that the required number of fans would not be available necessitating timely operator action to restore the required heat removal capability.

In addition, the associated violation was determined to be more than minor because the team could not reasonably determine the changes would not have ultimately required NRC prior approval.

Violations of 10 CFR 50.59 are dispositioned using the traditional enforcement process instead of the SDP because they are considered to be violations that potentially impede or impact the regulatory process. This violation is associated with a finding that has been evaluated by the SDP, and communicated with an SDP color reflective of the safety impact of the deficient licensee performance. The SDP, however, does not specifically consider the regulatory process impact. Thus, although related to a common regulatory concern, it is necessary to address the violation and finding using different processes to correctly reflect both the regulatory importance of the violation and the safety significance of the associated finding.

In this case, the team determined the finding could be evaluated using the SDP in accordance with IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Initial Characterization of Findings.” Because the finding impacted the Mitigating System cornerstone, the team screened the finding through IMC 0609, Appendix A, “The Significance Determination Process for Findings At-Power,” using Exhibit 2, "Mitigating Systems Screening Questions." In accordance with Exhibit 2, the team screened the finding using Exhibit 4, “External Events Screening Questions,” because the finding involved the degradation of equipment or function specifically designed to mitigate a severe weather initiating event. The team conservatively screened the finding as necessitating a detailed risk evaluation because the loss of UHS during a tornado event would degrade one or more trains of a system that supports a risk-significant system or function.

The Senior Reactor Analysts (SRAs) performed a bounding risk evaluation for the delta core damage frequency (ΔCDF) of tornado missile strike(s) causing a core damage event at Byron due to damage to the SXCT fans:

- The SRAs assumed that a tornado with wind speed exceeding 100 mph would be required to generate damaging missiles;

- The frequency of this tornado for Byron is approximately 1.13E-4/yr from the Risk Assessment Standardization Project (RASP) website;

- The tornado missiles were assumed to cause damage and fail an entire set of SXCT fans in addition to a set of fans that were initially out of service (i.e., 4 fans conservative assumption); and

- The SRAs further assumed that the tornado also caused a severe weather loss of offsite power event.

The Byron SPAR Model Version 8.27 and Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) Version 8.1.2 software were used by the SRAs to evaluate the risk significance of this finding. Using the Byron SPAR model, the Conditional Core Damage Probability (CCDP) (i.e., if the tornado event occurred and damaged one train of SXCT fans) is approximately 4.8E-4. Thus, a bounding ΔCDF calculated due to the SXCT vulnerability to missiles is approximately 5.4E-8/yr (i.e., 1.13E-4/yr x 4.8E-4 = 5.4E-8/yr).

Based on the detailed risk evaluation, the SRAs determined that the finding was of very-low safety significance (Green). As a result, this violation is categorized as Severity Level IV in accordance with Section 6.1.d of the NRC Enforcement Policy.

The team did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance. Specifically, the change was evaluated through the licensee 50.59 process in February 9, 2012.

Enforcement: Title 10 CFR 50.59, "Changes, Tests, and Experiments," Section (d)(1) requires, in part, the licensee to maintain records of changes in the facility, of changes in procedures, and of tests and experiments made pursuant 10 CFR 50.59(c). These records must include a written evaluation which provides the bases for the determination that the change, test, or experiment does not require a license amendment pursuant to paragraph (c)(2) of this section. Paragraph (c)(2)(ii) states, in part, that a licensee shall obtain a license amendment pursuant to 10 CFR 50.90 prior to implementing a proposed change, test, or experiment if the change, test, or experiment would result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR. In addition, 10 CFR(c)(2)(vi) states, in part, that a licensee shall obtain a license amendment pursuant to 10 CFR 50.90 prior to implementing a proposed change, test, or experiment if the change, test, or experiment would create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the Final Safety Analysis Report (FSAR)—as updated.

The UFSAR Section 9.2.5.3.2 in effect prior to the change implementation stated, "An analysis of the effect of multiple [emphasis added] tornado missiles on the essential service water cooling towers has been performed." In addition, UFSAR Sections 3.5.4.1 and 9.2.5.3.2 in effect prior to the change implementation stated, "An analysis of cooling tower capacity without fans [emphasis added] has been made." Moreover, UFSAR Section 3.1.2.1.2 in effect prior to the change implementation and at the time of this inspection stated, "The systems, components, and structures important to safety have been designed to accommodate, without loss of capability [emphasis added], effects of the design-basis natural phenomena along with appropriate combinations of normal and accident conditions."

Contrary to the above, on February 9, 2012, the licensee failed to maintain a record of a change in the facility made pursuant to 10 CFR 50.59(c) that included a written evaluation which provided the bases for the determination that the change did not require a license amendment pursuant to 10 CFR 50.59(c)(2). Specifically, the licensee made changes to the UFSAR-described SXCT tornado analysis and evaluated this change in 50.59 Evaluation 6G-11-0041. However, this evaluation did not consider the adverse effects of the introduction of a new failure mode, the resulting loss of heat removal capacity during worst postulated conditions, and addition of operator actions that have not been demonstrated can be completed in the required time to restore the required SXCT heat removal capacity during worst case conditions. As a result, the evaluation did not provide a basis for the determination that the change did not result in a more than a minimal increase in the likelihood of occurrence of a malfunction of the SXCT during and following a tornado event, and would not create a possibility for a malfunction of the SXCT with a different result than any previously evaluated.

The licensee is still evaluating its planned corrective actions to restore compliance. As an immediate corrective action, the licensee performed an operability evaluation. At the time of the CDBI exit meeting on June 16, 2015, the team was still reviewing the revised operability evaluation with the assistance of NRR.

Because this was a Severity Level IV violation, and was entered into the licensee's CAP as AR 02506214, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. (NCV 05000454/2015008-06; 05000455/2015008-06, Failure to Evaluate the Adverse Effects of Changing the SXCT Tornado Analysis as Described in the UFSAR)

The associated finding is evaluated separately from the traditional enforcement violation and, therefore, the finding is being assigned a separate tracking number. (FIN 05000454/2015008-07; 05000455/2015008-07, Failure to Evaluate the Adverse Effects of Changing the SXCT Tornado Analysis as Described in the UFSAR)

.6 Operating Procedure Accident Scenarios

a. Inspection Scope

The team performed a detailed reviewed of the procedures listed below. The procedures were chosen because they were associated with feed-and-bleed of the RCS, a loss of UHS, and other aspects of this inspection. For the procedures listed time critical operator actions were reviewed for reasonableness, in plant action were walked down with a licensed operator, and any interfaces with other departments were evaluated. The procedures were compared to the UFSAR, design assumptions, and training materials to assess consistency.

The following operating procedures were reviewed in detail:

- 1BFR-H1, "Response to Loss of Secondary Heat Sink Unit1," Revision 203;
- 0BOA PRI-7, "Loss of Ultimate Heat Sink Unit 0," Revision 1;
- 1BOA PRI-7, "Essential Service Water Malfunction Unit 1," Revision 106;
- 1BOA PRI-5, "Control Room Inaccessibility," Revision 108;

- 1BOA ELEC-5, “Local Emergency Control of Safe Shutdown Equipment,” Revision 106;
- 1BEP ES-1.3, “Transfer to Cold Leg Recirculation Unit 1,” Revision 204; and
- 1BCA-1.2, “LOCA Outside Containment Unit 1,” Revision 200.

b. Findings

(1) Failure to Provide Proper Direction for Low Level Isolation of the Refueling Water Storage Tank in Emergency Operating Procedures

Introduction: The team identified a finding of very-low safety significance (Green), and an associated NCV of TS 5.4, “Procedures,” for the failure to EOPs for transfer to cold leg recirculation. Specifically, Revision 204 of EOPs 1/2BEP ES-1.3, “Transfer to Cold Leg Recirculation,” did not contain instructions for transferring the ECCS and CS systems to the recirculation mode that ensured prevention of potential pump damage when the RWST is emptied following a LOCA.

Description: Procedures 1/2BEP ES-1.3 were established as the implementing EOPs for transferring ECCS and CS system suction from the RWST to containment sump recirculation. These EOPs were intended to be consistent with the technical guidelines of Westinghouse Owners Group Guidelines (WOG) Emergency Response Guideline (ERG) ES-1.3, “Transfer to Cold Leg Recirculation,” dated April 30, 2005. The technical guideline of WOG ERG ES-1.3 included the following caution statement: “Any pumps taking suction from the RWST should be stopped if RWST level decreases to (U.03).” The ERG defined (U.03) as “RWST empty alarm set point in plant specific units.” It also stated, “Based on pump suction piping configuration, the plant specific value of (U.03) may need to consider the possibility of vortexing and air entrainment.” The ERG basis for this caution stated, “Any pumps taking suction from the RWST must be stopped when the level in the tank reaches the empty alarm set point in order to prevent loss of suction flow and potential pump damage.” The licensee established 9 percent RWST level as the empty alarm set point to prevent air-entraining vortices and ensured adequate pump NPSH.

In 1996, the licensee changed EOPs 1/2BEP ES-1.3 to include a deviation to this ERG caution. Specifically, the revised EOP caution stated “Any pumps taking suction from the RWST should be stropped if level drops to 9 percent, unless a flow path also exists from the CNMT [containment] sump.” The EOP deviation document stated “This will allow continuing with switchover without securing pumps if an acceptable flow path exists.” It also stated “CNMT pressure should isolate the RWST flow path once aligned to the sump.” However, the licensee did not perform any evaluation to support this rationale.

The team was concerned because the revised caution did not assure to prevent air entrainment into the piping system to avoid ECCS and CS pump air binding and/or cavitation leading to potential damage. The licensee captured the team concern in their CAP as AR 02495580. The immediate corrective action was to create a standing order instructing operators to secure all pumps aligned to the RWST when it reaches 9 percent level. The proposed corrective actions to restore compliance at the time of this inspection included performing a detailed engineering analysis of the hydrodynamic fluid mechanics with a dual suction source option or removing the dual suction source option.

Analysis: The team determined that the failure to maintain an EOP for transfer to cold leg recirculation was contrary to TS 5.4, "Procedures," and was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of procedure quality, and affected the cornerstone objective of ensuring the availability, reliability, and capability of mitigating systems to respond to initiating events to prevent undesirable consequences. In addition, it was associated with the Barrier Integrity cornerstone attribute of procedure quality, and affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, failure to maintain an EOP for transfer to cold leg recirculation does not ensure that air entrainment into the piping system is prevented. As a consequence, the availability, reliability, and capability of the ECCS pumps to meet their mitigating function are not ensured. Similarly, the performance deficiency does not provide reasonable assurance the CS pumps would remain capable of supporting the reactor containment barrier function.

The team determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Initial Characterization of Findings." Because the finding impacted the Mitigating Systems and Barrier Integrity cornerstones, the team screened the finding through IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power," using Exhibit 2, "Mitigating Systems Screening Questions," and Exhibit 3, "Barrier Integrity Screening Questions." The finding screened as of very-low safety significance (Green) because it did not result in the loss of operability or functionality of mitigating systems, represent an actual open pathway in the physical integrity of reactor containment, and involved an actual reduction in function of hydrogen igniters in the reactor containment. Specifically, the incorrect caution would only be used in the event that transfer to sump recirculation was not completed by 9 percent tank level or if the RWST suction isolation valves fail to close. With respect to transfer to sump recirculation by 9 percent tank level, this is a time critical operator action that is tested and verified periodically on the plant simulator. A review of these simulator test results reasonably determined that operators reliably complete the transfer to sump recirculation prior to reaching this set point. With respect to the failure of the RWST suction isolation valves, these valves are test quarterly to demonstrate operability. A review of these test results for the last 3 years reasonably determined the valves would have isolated the tank when required.

The team did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance. Specifically, the inadequate caution had been added to 1/2BEP ES-1.3 in 1996.

Enforcement: In TS Section 5.4.1b states, in part, that written procedures shall be established, implemented, and maintained covering the EOPs required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter (GL) 82-33, Section 7.1. NUREG-0737, Supplement 1, Section 7.1.c, states, "Upgrade EOPs to be consistent with Technical Guidelines and an appropriate procedure Writer's Guide." The applicable technical guideline contained in WOG ERG ES-1.3, "Transfer to Cold Leg Recirculation," dated April 30, 2005, stated, "Any pumps taking suction from the RWST should be stopped if RWST level decreases to (U.03)." The ERG defined (U.03) as "RWST empty alarm set point in plant specific units." It also stated, "Based on pump suction piping configuration, the plant specific value of (U.03) may need to consider the possibility of vortexing and air entrainment."

The licensee established Revision 204 of 1/2BEP ES-1.3, "Transfer to Cold Leg Recirculation," as the implementing procedures for WOG ERG ES-1.3 to specify the actions required for transfer to containment sump recirculation. In addition, the licensee established 9 percent RWST level as the empty alarm set point, in part, to prevent air entrainment.

Contrary to the above, between 1996 to at least May 4, 2015, the licensee failed to maintain a written procedure covering the EOPs required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in GL 82-33, Section 7.1. Specifically, the licensee did not upgrade EOPs 1/2BEP ES-1.3 to be consistent with the technical guideline contained in WOG ERG ES-1.3 in that the EOPs did not instructed operators to stop any pumps taking suction from the RWST if level decreases below the 9 percent RWST empty alarm set point when a flow path from the containment sump existed.

The licensee is still evaluating its planned corrective actions. However, the team determined that the continued non-compliance does not present an immediate safety concern because the licensee created a standing order instructing operators to secure all pumps aligned to the RWST when it reaches 9 percent level.

Because this violation was of very-low safety significance, and was entered into the licensee's CAP as AR 02495580, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. (NCV 05000454/2015008-08; 05000455/2015008-08, Failure to Provide Proper Direction for Low Level Isolation of the RWST in EOPs)

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

.1 Review of Items Entered Into the Corrective Action Program

a. Inspection Scope

The team reviewed a sample of the selected component problems identified by the licensee, and entered into the CAP. The team 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 sampled and reviewed by the team are listed in the attachment to this report.

The team also selected three issues identified during previous CDBIs 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 05000454/2012007-01; 05000455/2012007-01, "Non-Conforming 480/120 VAC Motor Control Contactors;"
- NCV 05000454/2012007-03; 05000455/2012007-03, "Non-Conservative Calibration Tolerance Limits for Electrical Relay Settings;" and

- NCV 05000454/2012007-05; 05000455/2012007-05, "Failure to Provide Means to Detect Leak in Emergency Core Cooling Flow Path."

b. Findings

(1) Failure to Promptly Correct an NRC-Identified Non-Cited Violation Associated with the Capability to Detect and Isolate Emergency Core Cooling System Leakage

Introduction: A finding of very-low safety significance (Green), and an associated cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions," was identified by the team for the failure to correct a condition adverse to quality (CAQ). Specifically, on June 15, 2012, the NRC issued an NCV for the failure to provide means to detect and isolate a leak in the ECCS within 30 minutes as described in the UFSAR, which is a CAQ. As of May 22, 2015, the licensee had not corrected the CAQ.

Description: On June 15, 2012 the NRC identified that the licensee had failed to provide a means to detect and isolate a leak in the ECCS flow path within 30 minutes, as described in UFSAR 6.3.2.5, "System Reliability." Specifically, UFSAR 6.3.2.5 stated, in part, that the design of the auxiliary building and related equipment was based upon handling of leaks up to a maximum of 50 gallons per minute (gpm). In addition, it stated "Means were provided to detect and isolate such leaks in the emergency core cooling flow path within 30 minutes." The 2012 CDBI team identified that the licensee had failed to provide a means to detect and isolate an ECCS leak within 30 minutes. This issue was documented as NCV 05000454/2012007-05; 05000455/2012007-05, "Failure to Provide Means to Detect Leak in ECCS Flow Path," in Inspection Report (IR) 05000454/ 2012007; 05000455/2012007.

The licensee captured this NCV in their CAP as AR 01378257 and AR 01398434. The assigned corrective action tracking item (CA) was AR01378257-04, which stated:

> "Investigate the bases/sources of the values assigned to the single failure (50 gpm and 30 minutes), including whether there is a commitment associated. Create additional corrective actions (CA type) as necessary. If UFSAR change is determined feasible, include an action to determination of the impact of the leak duration lasting longer than 30 minutes on flood level inside containment and the Auxiliary Building."

The CA due date was extended eight times and, eventually, the CA was downgraded to an action tracking item (ACIT) because the licensee recognized that it did not correct the issue. Procedure PI-AA-125, "Corrective Action Program Procedure," defined ACIT as "Action items that are completed to improve performance, or correct minor problems that do not represent CAQ." On February 18, 2015, the licensee discovered that a new CA type assignment was not generated to address the NCV following the AR 01378257-04 downgrade from a CA to an ACIT type. This was inconsistent with step 4.5.2 of procedure PI-AA-125 in that it required, in part, the creation of a CA for any planned action necessary to correct a CAQ. This discovery was captured in the CAP as AR 02454767. The associated CA assignment stated:

> "Design Engineering will determine if UFSAR section 6.3.2.5 requires revision using the information provided in IR 01378257 and IR 1398434. If it is concluded a revision is required, an additional CA to track the change will be created."

During this inspection period, the team noted that the actions assigned by this CA were similar to those of AR 01378257-04, which the licensee had previously determined did not correct the NCV. The team was concerned because, as of May 22, 2015, the licensee failed to restore compliance and failed to have objective plans to restore compliance in a reasonable period following the NRC identification of the NCV on June 15, 2012.

The licensee captured the team's concern in their CAP as AR 02501454 to promptly restore compliance. As an immediate corrective action, the licensee reasonably determined ECCS remained operable by reviewing procedures and calculations. Specifically, the licensee reasonably determined procedures used when responding to postulated events would direct operators to detect and isolate an ECCS leak before it could adversely affect the system mitigating function or result in a radionuclide release in excess of applicable limits.

Analysis: The team determined that the failure to correct an NRC-identified NCV associated with the capability to detect and isolate ECCS leakage, which is a CAQ, was contrary to 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," and was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of design control, and affected the cornerstone objective of ensuring the availability, reliability, and capability of mitigating systems to respond to initiating events to prevent undesirable consequences. In addition, it was associated with the Barrier Integrity cornerstone attribute of design control, and affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the failure to detect and isolate a leak in the ECCS flow path within 30 minutes could compromise long term cooling, adversely affecting its capability to mitigate a DBA. In addition, a detection and isolation time greater than the time assumed by the design basis for an ECCS leak following an accident would result in greater radionuclide release to the auxiliary building, and the environment and, thus, does not assure that physical design barriers protect the public from radionuclide releases caused by accidents or events.

The team determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Initial Characterization of Findings." Because the finding impacted the Mitigating Systems and Barrier Integrity cornerstones, the team screened the finding through IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power," using Exhibit 2, "Mitigating Systems Screening Questions," and Exhibit 3, "Barrier Integrity Screening Questions." The finding screened as very-low safety significance (Green) because it did not result in the loss of operability or functionality, and it did not represent an actual pathway in the physical integrity of reactor containment. Specifically, the licensee reasonably demonstrated that an ECCS leak could be detected and isolated before it could adversely affect long-term cooling of the plant.

The team determined that the associated finding had a cross-cutting aspect in the area of human performance because the licensee did not use a consistent and systematic approach to make decisions. Specifically, the licensee downgraded the original CA to an ACIT without creating a new CA, which was inconsistent with the instructions contained in procedure PI-AA-125. Additionally, when the licensee subsequently discovered a CA type assignment was not created to address the NCV, the licensee

created a CA assignment to track actions that were similar to those tracked by the ACIT, which was inconsistent with the licensee previous determination that those actions did not correct the NCV. [H.13]

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

Contrary to the above, from June 15, 2012, to at least May 22, 2015, the licensee failed to correct a CAQ. Specifically, on June 15, 2012, the NRC issued NCV 05000454 /2012007-05; 05000455/2012007-05 for the failure to provide means to detect and isolate a leak in the ECCS within 30 minutes for Byron Station, Units 1 and 2, as described in UFSAR Section 6.3.2.5, which is a CAQ. As of May 22, 2015, the licensee had not corrected the CAQ in a reasonable period. Instead, the licensee created ACTI to develop a plan to correct the CAQ, and the associated due date was extended at least eight times.

The licensee is still evaluating corrective actions. However, the team determined that the continued non-compliance does not present an immediate safety concern because the licensee reasonably demonstrated that a leak could be detected and isolated before it could adversely affect long-term cooling of the plant or result in a radionuclide release in excess of applicable limits.

This violation is being cited as described in the Notice, which is enclosed with this IR. This is consistent with the NRC Enforcement Policy, Section 2.3.2.a.2, which states, in part, that the licensee must restore compliance within a reasonable period of time (i.e., in a timeframe commensurate with the significance of the violation) after a violation is identified. The NRC identified NCV 05000454/2012007-05; 05000455/2012007-05 on June 15, 2012, and documented it in IR 05000454/2012007. The team determined that the licensee failed to restore compliance within a reasonable time following issuance of this NCV and failed to have objective plans to restore compliance. (VIO 05000454 /2015008-09; 05000455/2015008-09, Failure to Promptly Correct an NRC-Identified NCV Associated with the Capability to Detect and Isolate ECCS Leakage)

(2) Failure to Maintain the Accuracy of the Instrument Loops Used to Verify Compliance with the Containment Average Air Temperature Technical Specification Limit

Introduction: The team identified a finding of very-low safety significance (Green), and an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure to have procedures to maintain the accuracy within the necessary limits of instrument loops used to verify compliance with the containment average air temperature TS limit of 120 degrees Farhenheit. Specifically, in 2007, the licensee cancelled the periodic PMs intended to maintain the instrument accuracy necessary for verifying compliance with the limiting condition for operation (LCO) of TS 3.6.5, "Containment Air Temperature."

Description: The team reviewed selected corrective action documents initiated by the licensee as a result of their recent Focused Self-Assessment titled, "Readiness Review for 2015 NRC Component Design Basis Inspection." The reviewed corrective action document sample included AR 02437973. This corrective action document was initiated on January 15, 2015, in part, for the discovery that the four instrument loops used for

determining containment average air temperature (i.e., loops 1/2VP-030, 1/2VP-031, 1/2VP-032, and 1/2VP-033) were removed from the PM Program in 2007 via Service Request 47654. The corrective action document also noted that the PMs were last performed in 2001 for 1VP-030; 2002 for 1/2VP-031, 1/2VP-032, 2VP-030, and 2VP-033; and 2009 for 1VP-033.

This corrective action document created an ACIT to determine if the PMs should be reestablished. Procedure PI-AA-125, "Corrective Action Program Procedure," defined ACIT as "Action items that are completed to improve performance, or correct minor problems that do not represent CAQ." On March 3, 2015, the ACIT concluded that there was no need to reestablish the PMs due to the instrument loop reliability, previous calibration history, loop design, redundancy, and daily monitoring which the licensee believed would notice instrument drift. However, the team noted that TS SR 3.6.5.1 required verifying containment air temperature is less than 120 degrees Fahrenheit by averaging the instrument readings and, thus, instrument reading variability was expected. In addition, the team noted the licensee had not established a variability limit (i.e., acceptance criteria) among the instrument loops and relied on operator judgment to identify adverse drifts.

The team was concerned because these instrument loops were not maintained to ensure their accuracy was within the necessary limits to verify compliance with the containment average air temperature TS limit of 120 degrees Fahrenheit. Containment average air temperature is an initial condition used in DBA analyses, and is an important consideration in establishing the containment environmental qualification operating envelope for both pressure and temperature. This TS limit ensures that initial conditions assumed in these analyses are met during unit operations.

The licensee captured the team's concern in their CAP as AR 02502846. As an immediate corrective action, the licensee reasonably established that the 120 degrees Fahrenheit limit was not exceeded by reviewing applicable historical records from 2002 to time of this inspection. The proposed corrective action to restore compliance at the time of this inspection was to reconstitute PM procedures for these instrument loops to assure they are maintained.

Analysis: The team determined that the failure to have procedures to maintain the accuracy within necessary limits of the instrument loops used during SR 3.6.5.1 was contrary to 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," and was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the configuration control attribute of the Barrier Integrity Cornerstone, and adversely affected the cornerstone objective to ensure that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the failure to have procedures to maintain the accuracy of the containment air temperature instrumentation loops within necessary limits does not ensure the instrument loop accuracy is maintained such that SR 3.6.5.1 activities are effective at verifying compliance with the containment average air temperature TS limit. As a result, the potential exists for an inoperable condition to go undetected.

The team determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Initial Characterization of Findings." Because the finding impacted the Barrier Integrity

cornerstone, the team screened the finding through IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power," using Exhibit 3, "Barrier Integrity Screening Questions." The finding screened as of very-low safety significance (Green) because it did not represent an actual open pathway in the physical integrity of reactor containment or involved an actual reduction in hydrogen igniter function. Specifically, the containment integrity remained intact and the finding did not impact the hydrogen igniter function.

The team determined that this finding had a cross-cutting aspect in the area of problem identification and resolution because the licensee did not identify issues completely and accurately in accordance with the CAP. Specifically, on January 15, 2015, the licensee captured the lack of periodic PM activities for the containment air temperature instrument loops in the CAP. However, the licensee failed to completely and accurately identify the issue in that it was not treated as a CAQ. As a consequence, no corrective actions were implemented. [P.1]

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

Contrary to the above, since 2007 to at least May 22, 2015, the licensee failed to have a procedure for maintaining the accuracy within the necessary limits of the instrument loops used while implementing SR 3.6.5.1. Specifically, in 2007, the licensee cancelled the PMs intended to maintain the instrument loops accuracy necessary for verifying compliance with LCO 3.6.5 limit.

The licensee is still evaluating its planned corrective actions. However, the team determined that the continued non-compliance does not present an immediate safety concern because containment average air temperature readings were significantly lower than the associated TS limit, and are reasonably expected to maintain that margin in the foreseeable future based on past performance.

Because this violation was of very-low safety significance, and was entered into the licensee's CAP as AR 02502846, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000454/2015008-10; 05000455/2015008-10, Failure to Maintain the Instrument Loops Used to Verify Compliance with the Containment Average Air Temperature TS Limit)

(3) Operability Evaluation Relied on Probabilities of Occurrence of the Associated Event

Introduction: The team identified a finding of very-low safety significance (Green), and an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure to make an operability determination without relying on the use of probabilistic tools. Specifically, an operability evaluation related to an SXCT degraded condition used probabilities of occurrence of tornado events which was contrary to the requirements of Revision 16 of procedure OP-AA-108-115, "Operability Determinations."

Description: Revision 7 of UFSAR Section 3.5.4, "Analysis of Multiple Missiles Generated by a Tornado," stated that the SXCT fans, fan motors, and fan drives were not protected from tornado missiles. It also stated that "An analysis of cooling tower

capacity without fans has been made." In addition, it stated that "Using the most conservative design conditions, it is predicted if the plant is shut down under non-LOCA conditions with loss of offsite power, the temperature of the service water supplied to the plant will not exceed 110 degrees Farhernheit." However, during the 2005 NRC SSDPC inspection, the inspectors noted that this analysis had not been updated to reflect changes that increased the heat load. The SSDPC documented this concern as URI 05000454/2005002-07; 05000455/2005002-07. In 2007, this URI was subsequently closed to NCV 05000454/2007004-03; 05000455/2007004-03. As a result, on February 14, 2012, the licensee completed EC 385829, "UHS Capability with Loss of SX Fans Due to Tornado Missiles," to change the UHS tornado missile design basis to require a minimum of two SXCT fans and motors for cooling following a tornado event. The change did not include adding tornado protection to the fans, fan motors, and fan drives.

On August 9, 2013, the licensee initiated corrective action document IR 01545153 for the NRC discovery that the associated written safety evaluation intended to provide the bases for the determination that this change did not require a license amendment failed to consider the change adverse effects. On August 14, 2013, the licensee initiated corrective action document AR 1546621 to address the associated technical implications. This corrective action document resulted in Revision 0 of Operability Evaluation 13-007, "Ultimate Heat Sink Capability with Loss of Essential Service Water Cooling Tower Fans," intended to reasonably demonstrate UHS operability until corrective actions to restore compliance were implemented.

During this inspection period, the CDBI team noted that Operability Evaluation 13-007 relied on the probability of occurrence of a tornado. Specifically, it stated "The UHS is capable of providing the required cooling because, given a tornado strike under the design conditions in the UFSAR, the probability of occurrence is less than the acceptance criteria of 10E-7 /year in SRP 2.2.3." It also stated that "The software used to determine the missile hit probability is called [Tornado Missile Risk Evaluation Methodology] TORMIS." In addition, it stated that "The software uses site specific factors such as predicted tornado characteristics, tornado occurrence rates, building layout, potential missile sources and types, missile distribution and the number of potential missiles." The supporting analysis used the UFSAR Section 2.3.1.2.2, "Tornadoes and Severe Winds." tornado probability of occurrence value of 21E-4 per year.

Procedure OP-AA-108-115, "Operability Determinations," Section 4.5.13, "Use of PRA," stated:

> "PRA is a valuable tool for evaluating accident scenarios because it can consider the probabilities of occurrence of accidents or external events. Nevertheless, the definition of operability is that the SSC must be capable of performing its specified function or functions, which inherently assumes that the event occurs and that the safety function or functions can be performed. Therefore, the use of PRA or probabilities of occurrence of accidents or external events is not consistent with the assumption that the event occurs, and is not acceptable for making operability decisions."

Thus, the team determined that the use of TORMIS, the probability for occurrence of tornados, and the probabilities of missile strikes was not acceptable and contrary to licensee procedure OP-AA-108-115. The team, in consultation with NRR, also

determined that this procedure requirement was consistent with Attachment C.06 of NRC IMC 0326, “Operability Determinations & Functionality Assessments for Conditions Adverse to Quality or Safety,” which was established to assist NRC inspectors review of licensee determinations of operability and resolution of degraded or nonconforming conditions.

In addition, the team noted that Byron had not obtained NRC approval for the site specific use of TORMIS as stated in Regulatory Issue Summary (RIS) 2008-14, “Use of TORMIS Computer Code for Assessment of Tornado Missile Protection.” Specifically, the RIS stated that “The initial use of the TORMIS methodology as described in this RIS requires a license amendment in accordance with 10 CFR 50.59(c)(2)(viii) and subsequent revision to the plant licensing basis because it is a ‘Departure from the method of evaluation described in the FSAR, as updated, used in establishing the design bases or in the safety analysis’ as defined in 10 CFR 50.59(a)(2).”

The team was concerned because Operability Evaluation 13-007 did not reasonably demonstrate the degraded UHS would be capable of performing its function following a tornado event. The licensee captured the team concern in their CAP as AR 2504624 to revise Operability Evaluation 13-007 without using PRA tools.

Analysis: The team determined that the failure to make an operability determination without relying on the use of probabilistic tools was contrary to licensee procedure OP-AA-108-115 and was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of protection against external events, and affected the cornerstone objective of ensuring the availability, reliability, and capability of mitigating systems to respond to initiating events to prevent undesirable consequences. Specifically, failure to perform an adequate operability evaluation does not ensure the SXCT would remain capable of performing its safety function, and had the potential to allow an inoperable condition to go undetected.

The team determined the finding could be evaluated using the SDP in accordance with IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Initial Characterization of Findings.” Because the finding impacted the Mitigating System cornerstone, the team screened the finding through IMC 0609, Appendix A, “The Significance Determination Process for Findings At-Power,” using Exhibit 2, "Mitigating Systems Screening Questions." In accordance with Exhibit 2, the team screened the finding using Exhibit 4, “External Events Screening Questions,” because the finding involved the degradation of equipment or function specifically designed to mitigate a severe weather initiating event. The team conservatively screened the finding as necessitating a detailed risk evaluation because the loss of UHS during a tornado event would degrade one or more trains of a system that supports a risk-significant system or function.

The SRAs performed a bounding risk evaluation for the ΔCDF of tornado missile strike(s) causing a core damage event at Byron due to damage to the SXCT fans:

- The SRAs assumed that a tornado with wind speed exceeding 100 mph would be required to generate damaging missiles.

- The frequency of this tornado for Byron is approximately 1.13E-4/yr from the RASP website;

- The tornado missiles were assumed to cause damage and fail an entire set of SXCT fans in addition to a set of fans that were initially out of service (i.e., 4 fans - conservative assumption); and

- The SRAs further assumed that the tornado also caused a severe weather loss of offsite power event.

The Byron SPAR Model Version 8.27 and SAPHIRE Version 8.1.2 software were used by the SRAs to evaluate the risk significance of this finding. Using the Byron SPAR model, the CCDP (i.e., if the tornado event occurred and damaged one train of SXCT fans) is approximately 4.8E-4. Thus, a bounding ΔCDF calculated due to the SXCT vulnerability to missiles is approximately 5.4E-8/yr (i.e., 1.13E-4/yr x 4.8E-4 = 5.4E-8/yr).

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

The team determined that this finding had a cross-cutting aspect in the area of human performance because the licensee did not ensure knowledge transfer to maintain a knowledgeable and technically competent workforce. Specifically, the licensee did not ensure personnel were trained on the prohibition of the use of probabilities of occurrence of an event when performing operability evaluations, which was contained in procedure OP-AA-108-115. [H.9]

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

The licensee established Revision 16 of procedure OP-AA-108-115, "Operability Determinations," as the implementing procedure for assessing operability of SSCs, an activity affecting quality. Section 4.5.13, "Use of Probabilistic Risk Assessment," stated "[…] the use of PRA or probabilities of occurrence of accidents or external events is not consistent with the assumption that the event occurs, and is not acceptable for making operability decisions."

Contrary to the above, on August 20, 2013, the licensee failed to follow Section 4.5.13 of procedure OP-AA-108-115. Specifically, the licensee used a PRA tool (i.e., TORMIS) and probabilities of occurrence of an external event (i.e., tornado) when making an operability decision related to the SXCT degradation when mitigating tornado events. Establishing a reasonable expectation of operability is an activity affecting quality.

As an immediate corrective action, the licensee revised the affected operability evaluation without using PRA tools. At the time of the CDBI exit meeting on June 16, 2015, the team was still reviewing the revised operability evaluation with the assistance of NRR.

Because this violation was of very-low safety significance and was entered into the licensee's CAP as AR 2504624, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. (NCV 05000454/2015008-11; 05000455/2015008-11, Operability Evaluation Relied on Probabilities of Occurrence of the Associated Event)

4OA6 Management Meetings

.1 Interim Exit Meeting Summary

On May 22, 2015, the team presented the inspection results to Mr. R. Kearney, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors had outstanding questions that required additional review and a follow-up exit meeting.

.2 Exit Meeting Summary

On June 16, 2015, the team presented the inspection results to Mr. B. Currier, and other members of the licensee staff. The licensee acknowledged the issues presented. The team asked the licensee whether any materials examined during the inspection should be considered proprietary. Several documents reviewed by the team were considered proprietary information and were either returned to the licensee or handled in accordance with NRC policy on proprietary information.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

R. Kearney, Site Vice President
T. Chalmers, Plant Manager
C. Keller, Engineering Director
B. Currier, Senior Manager of Design Engineering
D. Spitzer, Regulatory Assurance Manager
J. Cunzeman, Mechanical/Structural Design Manager
A. Corrigan, NRC Coordinator

U.S. Nuclear Regulatory Commission

C. Lipa, Chief, Engineering Branch 2
J. Ellegood, Chief, Reactor Projects Branch 3 (Acting)
N. Féliz Adorno, Senior Reactor Inspector
C. Zoia, Senior Resident Inspector (Acting)
J. Draper, Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000454/2015008-01; 05000455/2015008-01	URI	Question Regarding the Maximum Wet Bulb Temperature Value Assumed in the SXCT Tornado Analysis (Section 1R21.3.b(1))
05000454/2015008-02; 05000455/2015008-02	URI	Maximum Wet Bulb Temperature Value Assumed in SXCT Analysis Was Not Monitored (Section 1R21.3.b(2))
05000454/2015008-03; 05000455/2015008-03	NCV	Failure to Evaluate the Adverse Effects of Sharing the RWSTs of Both Reactor Units (Section 1R21.5.b(1))
05000454/2015008-04; 05000455/2015008-04	FIN	Failure to Evaluate the Adverse Effects of Sharing the RWSTs of Both Reactor Units (Section 1R21.5.b(1))
05000454/2015008-05; 05000455/2015008-05	NCV	Failure to Adequately Implement a Design Change Associated with the RWSTs (Section 1R21.5.b(2))
05000454/2015008-06; 05000455/2015008-06	NCV	Failure to Evaluate the Adverse Effects of Changing the SXCT Tornado Analysis as Described in the UFSAR (Section 1R21.5.b(3))
05000454/2015008-07; 05000455/2015008-07	FIN	Failure to Evaluate the Adverse Effects of Changing the SXCT Tornado Analysis as Described in the UFSAR (Section 1R21.5.b(3))
05000454/2015008-08; 05000455/2015008-08	NCV	Failure to Provide Proper Direction for Low Level Isolation of the RWST in EOPs (Section 1R21.6.b(1))
05000454/2015008-09; 05000455/2015008-09	VIO	Failure to Promptly Correct an NRC-Identified NCV Associated with the Capability to Detect and Isolate ECCS Leakage (Section 4OA2.1.b(1))

05000454/2015008-10; 05000455/2015008-10	NCV	Failure to Maintain the Instrument Loops Used to Verify Compliance with the Containment Average Air Temperature TS Limit (Section 4OA2.1.b(2))
05000454/2015008-11; 05000455/2015008-11	NCV	Operability Evaluation Relied on Probabilities of Occurrence of the Associated Event (Section 4OA2.1.b(3))

Closed

05000454/2015008-03; 05000455/2015008-03	NCV	Failure to Evaluate the Adverse Effects of Sharing the RWSTs of Both Reactor Units (Section 1R21.5.b(1))
05000454/2015008-04; 05000455/2015008-04	FIN	Failure to Evaluate the Adverse Effects of Sharing the RWSTs of Both Reactor Units (Section 1R21.5.b(1))
05000454/2015008-05; 05000455/2015008-05	NCV	Failure to Adequately Implement a Design Change Associated with the RWSTs (Section 1R21.5.b(2))
05000454/2015008-06; 05000455/2015008-06	NCV	Failure to Evaluate the Adverse Effects of Changing the SXCT Tornado Analysis as Described in the UFSAR (Section 1R21.5.b(3))
05000454/2015008-07; 05000455/2015008-07	FIN	Failure to Evaluate the Adverse Effects of Changing the SXCT Tornado Analysis as Described in the UFSAR (Section 1R21.5.b(3))
05000454/2015008-08; 05000455/2015008-08	NCV	Failure to Provide Proper Direction for Low Level Isolation of the RWST in EOPs (Section 1R21.6.b(1))
05000454/2015008-10; 05000455/2015008-10	NCV	Failure to Maintain the Instrument Loops Used to Verify Compliance with the Containment Average Air Temperature TS Limit (Section 4OA2.1.b(2))
05000454/2015008-11; 05000455/2015008-11	NCV	Operability Evaluation Relied on Probabilities of Occurrence of the Associated Event (Section 4OA2.1.b(3))

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
4391/19D-11	Sizing of Replacement Battery Charger for Diesel Driven Pumps	0
BYR08-035	Essential Service Water Cooling Tower Basin Level Indication Uncertainty Analysis	0
BYR12-070	Auxiliary Building Environment following a High Energy Line Break in the Turbine Building	2
BYR12-072	Thermal Endurance Evaluation of the Safety Related Electrical Equipment in the Essential Service Water (SX) Cooling Tower Switchgear Rooms	0
BYR97-193	Battery Duty Cycle and Sizing for the Byron Diesel Driven Auxiliary Feedwater Pumps and the Byron Diesel Driven Essential Service Water Makeup Pumps	1-1E
BYR97-205	125VDC Battery Charger Sizing Calculation	2
BYR97-204	125 VDC Battery Sizing Calculation	3-3K
BYR97-224	125Vdc Voltage Drop Calculation	4-4A
BYR97-226	125 V DC System Short Circuit Calculation	4
BYR97-239	SX Cooling Tower Basin Level Auto Start Level Set Point Analysis	1
BYR97-336	SX Cooling Tower Basin – Time to Reach the Low Level Alarm Set Point	1
BYR2000-136	Voltage Drop Calculation for 4160V Switchgear Breaker Control Circuits	1
BYR2000-191	Voltage Drop Calculation for 480V Switchgear Breaker Control Circuits	0 -0C
4391/19-AN-3	Protective Relay Settings for 4.16 kV ESF Switchgear	16
19-AQ-24	Voltage Drop on 480-120V AC Control Transformer Circuits	8
19-AQ-63	Division Specific Degraded Voltage Analysis	7A
19-AQ-69	Evaluation of the Adequacy of the 120 Vac Distribution Circuit at the Degraded Voltage Setpoint	16
19-AQ-75	Essential Service Water Cooling Tower 480V Buses Maximum Voltage	1
19-AU-4	480 V Unit Substation Breaker and Relay Settings	19
19-G-1	Cable Ampacity	2
19-T-5	Diesel Generator Loading During LOOP/LOCA	7
BYR01-068	Environmental Parameters of EQ Zones	2
BYR01-084	Generic Thermal Overload Heater Sizing Calculation for Motor Operated Valves	000

CALCULATIONS

Number	Description or Title	Revision
BYR01-095	Motor Operated Valves (MOV) Actuator Motor Terminal Voltage and Thermal Overload Sizing Calculation – Essential Service Water (SX) System	1
BYR06-111	Model APT-30K-11 SXCT Fan Blade Pitch Setting	1
BYR12-042	Essential Service Water Discharge Header Temperature Indication Uncertainty	0
BYR95-005	120 VAC Instrument Bus/SSPS Cabinet Fuse Sizing and Coordination	0
BYR96-128	Refueling Water Storage Tank (RWST) Level Alarm Bistables and Level Indication Accuracy	2
DIT BB-EPED-0189	Design Information Transmittal: Minimum Starting/Running Voltages for Essential Motors	5/14/93
DIT BB-EXT-0406	Design Information Transmittal: Essential Service Water Cooling Tower Fan Motors [starting duty]	12/9/92
DIT-BRW-2002-033	Design Information Transmittal: Basis for EDG loading	10/15/02
SI-90-01	Minimum Containment Flood Level	11
BYR04-016	RHR, SI, CV, and CS Pump NPSH During ECCS Injection Mode	2
BYR14-053	Pressurizer PORV Air Accumulator Tank Requirements	0
BYR06-029	Byron/Braidwood SI/RHR/CS/CV system hydraulic analysis in support of GSI-191	5
BYR06-058	NPSHA for RHR & CS Pumps During Post-LOCA Recirculation	0
BYR07-055	Determination of the Correlation for the Critical Submergence Height (Vortexing) for the RWST	0
SM-SI0930	RWST Level	D
SITH-1	Refueling Water Storage Tank (RWST) Level Set points	8
CN-RRA-00-47	Byron/Braidwood Natural Circulation Cooldown TREAT Analysis for RSG and Uprating Programm	3
CN-RRA-00-47	Byron/Braidwood Natural Circulation Cooldown TREAT Analysis for RSG and Uprating Program	4
CQD-200074	PORV Accumulator Tank	Z2
8.1.16	Refueling Water Storage Tanks Analysis and Design	5
BYR97-287	Determination of RWST Free Air Volume above Maximum RWST Water Level	2
SM-SI0930	RWST Level	D
SM-SI0931	RWST Level	D
SM-SI0932	RWST Level	D
SM-SI0933	RWST Level	D
ATD-0062	Heat Load to the Ultimate Heat Sink During a Loss of Coolant Accident	5
BYR03-131	Evaluation of UHS Make Up for CST-based Cooldown Profile	1
BYR05-018	Tornado Missile Risk Assessment of Vulnerable Targets of Essential Service Water Cooling Towers	0
BYR06-111	Model APT-30K-11 SXCT Fan Blade Pitch Setting	1

CALCULATIONS

Number	Description or Title	Revision
BYR09-002	UHS Capability with Loss of SX Fans due to a Tornado Event	1
BYR09-002	UHS Capability with Loss of SX Fans due to a Tornado Event	1
BYR97·239	SX Cooling Tower Basin Level Auto Start Setpoint Error Analysis	1
BYR97-034	Essential Service Water Cooling Tower Basin Minimum Volume Versus Level and Minimum Usable Volume Calculation	0a
BYR97-034	Essential Service Water Cooling Tower Basin Minimum Volume Versus Level and Minimum Usable Volume Calculation	0A
BYR97-127	Byron Ultimate Heat Sink Cooling Tower Performance Calculations	1
BYR97-134	Heat Load on the UHS - 2 Unit Shutdown	3
BYR97-366	SX Cooling Tower Basin - Time to Reach the Low Level Alarm Set Point	1
BYRO8-035	Essential Service Water Cooling Tower Basin Level Indication Uncertainty Analysis	0
NED-M-MSD-009	Byron Ultimate Heat Sink Cooling Tower Basin Temperature Calculation: Part IV	8B
NED-M-MSD-014	Byron Ultimate Heat Sink Cooling Tower Basin Makeup Calculation	9
UHS-01	Ultimate Heat Sink Design Basis LOCA Single Failure Scenarios	4
SL-101	ELMS-AC Report: Running Voltage Summary, Division 12	1/21/15
SL-102	ELMS-AC Report: Short Circuit Summary for High Voltage Buses	1/21/15
SL-109	ELMS-AC Report: Connection Loading, Division 12	1/21/15
SL-112	ELMS-AC Report: Single Bus Summary, Bus 142	4/20/15

CORRECTIVE ACTION DOCUMENTS Generated Due to the Inspection

Number	Description or Title	Date
AR02488878	2015 CDBI – Design Analysis Inconsistency Identified	4/21/15
AR02489108	NRC CDBI: Loose Parts Found During Walkdown of RWST	4/22/15
AR02489149	CDBI – Bucket Collecting Diesel Fuel Drips from 0DO088A	4/22/15
AR02489198	CDBI – SX Make-Up Pump Temperature Recorder Panel Memory Full	4/22/15
AR02489297	CDBI – Outdated Information in SystemIQ	4/22/15
AR02489456	NRC ID: Jumpers Not Readily Available for 1/2BOA PRI-5	4/22/15
AR02489360	Negative Vibration Reading on Idle 0E SXCT Fan	4/22/15
AR02490324	CDBI – ID 1RY456 WO As-Found Not as Expected, No IR Written	4/24/15
AR02493191	CDBI - Issues Identified in Calculation BYR 97-224	4/30/15
AR02493990	CDBI – Issue Identified in Calculation 19-AQ-69	5/1/15
AR02495580	CDBI Question Related to BEP ES-1.3 Cold Leg Recirculation	5/4/15

CORRECTIVE ACTION DOCUMENTS Generated Due to the Inspection

Number	Description or Title	Date
AR02495584	CDBI – FC Purification Flow Not Considered in RWST NPSH Calc	5/4/15
AR02495866	CDBI - NRC Identified Issues in BYR97-193	5/5/15
AR02496142	CDBI – 50.59 and DRP did not explicitly evaluate GDC 5	5/5/15
AR02495973	NRC CDBI - Error Discovered in EACE Investigation	5/6/15
AR02496766	CDBI – RWST Calc May Lead to Inconsistent Application of TS	5/6/15
AR02497347	NRC CDBI: Procedure Enhancement for ECCS Flow Balancing	5/6/15
AR02497940	CDBI Deficiency Identified - THD Testing for Instrument Inverter	5/8/15
AR02497925	Lightning Rod on SX Cooling Tower Bent; Clarify Inspection WO Instructions	5/8/15
AR02501392	CDBI 2015 – VTIP for Containment DP Has Limited Lead Length	5/15/15
AR02501454	CDBI – CA Created for NCV Does Not Resolve Issue	5/15/15
AR02502846	No Routine PM on Containment Temperature Loops	5/19/15
AR02504624	CDBI Concern Regarding Op Eval 13-007	5/22/15
AR02504475	CDBI – TS Clarification Needed for Transition to LTOPs	5/22/15
AR02506214	2012 50.59 for SXCT Tornado Analysis	5/19/15

CORRECTIVE ACTION DOCUMENTS Reviewed During the Inspection

Number	Description or Title	Date
AR00301744	Design of RWST Vacuum Relief System	2/15/05
AR00239280	RWST Vent / Vacuum Breaker Design Basis Issues	7/27/04
AR00880223	0A SX M/U PP Failures	2/13/09
AR00881611	0A SX MU Pump Did Not Stop When Local CS Taken to Off	2/17/09
AR01053940	1DC08E Battery, 1DC08E 123 Bus and DC 123 Batt Low	4/8/10
AR01115570	DC Bus 123 Low Voltage	9/21/10
AR01204963	Megger Test of Submerged Cable (1SX172)	4/20/11
AR01217212	Check/Adjust Charger 123 Float Voltage	5/17/11
AR01263407	0A SX MU PP Failed to Start at the Desired Setpoint SPC	9/15/11
AR01318043	0A SX M/U PP Battery Bank Test	1/25/12
AR01362643	Replace Breaker for MCC 035-2-C5 (0CW03PC-C)	5/4/12
AR01368220	CDBI ESF MCC Contactors not Tested at Assumed Pickup Volt	5/18/12
AR01376793	CDBI Follow-up on MCC Contactors (IR 1368220)	6/11/12
AR01377764	NRC CDBI - Protective Relay Setting Tolerances	6/12/12
AR01378259	Need Engineering to Evaluate Test Frequency	6/15/12
AR01380744	Action Tracking Needed for Size 3 and 4 Contactors	6/22/12
AR01387518	The Station 111 ESF Battery Needs to Be Replaced in B1R19	7/11/12
AR01387520	The Station 112 ESF Battery Needs to be Replaced in B1R19	7/11/12
AR01390648	Protective Relay Tolerances Require Fleet Review	7/19/12

CORRECTIVE ACTION DOCUMENTS Reviewed During the Inspection

Number	Description or Title	Date
AR01398419	NRC ID'D CDBI Green NCV Non-Conforming 480/120 VAC Motor Contactors	6/15/12
AR01398426	NRC CDBI Green NCV Non-Conservative Cal Tolerance for Elec Relays	6/15/12
AR01413695	Engineering Evaluate Frequency of Battery Capacity Test	9/16/12
AR01502583	0A SX Makeup Pump Failed to Auto Start per 0BOSR 7.9.6-1	4/16/13
AR01518720	Breaker Will Not Reset During Oden Testing	5/29/13
AR01570572	0A SX M/U PP Had To Be Tripped During Monthly Run	10/10/13
AR01588590	Loss of Instrument Bus 111	11/21/13
AR01589264	Need New Contingency Work Order ofr Instrument Inverter 111	11/23/13
AR01590368	NRC ID - PCM Template/Vendor Manual Recommendation	11/26/13
AR01611287	0A SX Makeup Pump Auto Start Level Setpoint	1/23/14
AR01654589	Erratic Reading on Ammeter (111-IP001) for Inverter 111	4/30/14
AR01658463	Specific Gravity of Battery Cell Still Low After Equalize	5/10/14
AR01680303	0A SX MU PP Trouble Alarm Continues to Alarm	7/10/14
AR01693147	Gradual Float Current Trend on 111 Battery Charger	4/15/14
AR02407275	0SX02PA Kept Running	11/5/14
AR02417160	Pump "As Found" Condition/Dry Start Improvement Opportunity	11/25/14
AR02440865	Thermography Needed on FRT for Instrument Inverter 111	11/29/14
AR02448283	0A SX MU Failed Surveillance	2/5/15
AR01299897	Replace Breaker for MCC 132Z1-A4 (0SX157A)	12/8/11
AR01056715	NER-NC-10-008-Y – Buried Cable	4/14/10
AR01322720	B2F26 Bus 142 Undervoltage Relay	2/3/12
AR01409309	Safety-Related Cable Vault 1M1G(1G1) Inspection - Repairs	9/5/12
AR01417720	MCC 132Z1-A5 Tripped Out of Tolerance	9/24/12
AR01425642	Safety-Related Cable Vault 1J2 Inspection - Repairs	10/12/12
AR01592242	Operating Experience Applicable to Byron (SXCT Fan Reverse Rotation)	12/2/13
AR01625774	Degraded Voltage Relay Target did not Change State	2/25/14
AR01648079	Step Change Identified in Unit 1 Containment Air Temperature in PI	4/16/14
AR01687277	Safety Related Cable Vault PM and Engineering Inspections	7/30/14
AR02437410	Cable Vault PM and Engineering Inspections	1/14/15
AR02437973	CDBI FASA – Review of Robinson and Wolf Creek Findings	1/15/15
AR00239280	RWST Vent/Vacuum Breaker Design Basis Issue	7/27/04
AR01360789	U-1 RWST level	4/30/12
AR01361308	U-1 RWST on FC Purification	5/2/12
AR01361838	U-1 RWST level loss During Purification	5/3/12
AR0128230	NRC Information Notice 2012-01: Seismic Considerations – Principally Issues Involving Tanks	5/9/12
AR01398434	NRC CDBI Green NCV-Leak Detection for ECCS Flowpath Lacking	6/15/12
AR01378257	CDBI, Question about ECCS leakage	6/15/12

CORRECTIVE ACTION DOCUMENTS Reviewed During the Inspection

Number	Description or Title	Date
AR01465872	Review of Braidwood IR 1459353 Pzr PORV Accumlator Press	1/23/13
AR01635829	1B PZR PORV Accum Failed Decay Test	3/19/14
AR02454767	NOS ID: No CA to Correct an NRC NCV	2/18/15
IR298958	SSD&PC: Inaccurate Setpoints Referenced in BYR97-034	6/30/05
AR 01546621	Inadequate 50.59 for EC 385829 (SXCT Tornado Missiles)	8/14/13
AR295141	Ssd&pc Question on Tornado Anaylsis Supporting UFSAR Stmnt	1/28/05
AR1677584	Clarification Needed on UHS Passive Failure Design	7/1/14
AR1567903	NRC Question and Feedback on UHS Temperature Analysis	10/3/13
AR1677513	UFSAR Section 2.4.11.6 Needs Revision	7/1/14
AR1677646	Recommendation from UHS Assessment	7/1/14
AR1546621	Inadequate 50.59 for EC 385829	2/9/12
AR2406579	Failed “Spider” Bearing on 0A SX Makeup Pump	11/4/14
AR1269014	Obsolete SX Makeup Pump D/O Storage Tank Level Indicator	9/28/11
AR2437508	Review of Flow Anomaly On 0B SX Makeup	1/14/15
AR2448283	0A SX MU Failed Surveillance	2/5/15

DRAWINGS

Number	Description or Title	Revision
S-529	Essential Service Cooling Tower Drainage Duct Plan, Section Details	H
6E-0-4030SX09	Schematic Diagram - Essential Service Water Make-up Pump 0A 0SX02PA	P
6E-0-4030SX23	Schematic Diagram - Essential Service Water Make-up Pump 0A Control Cabinet (Diesel Driven) 0SX02JA	S
6E-0-4030SX24	Schematic Diagram - Essential Service Water Make-up Pump 0A Control Cabinet (Diesel Driven) 0SX02JA Annunciator	F
6E-0-4030CW11	Schematic Diagram – Essential service Water Cooling Tower 0A & 0B Well Water Make-up Valves 0CW100A & B	D
6E-0-4030WW01	Schematic Diagram – Deep Well Pump 0A – 0WW01PA	M
6E-0-4030WW02	Schematic Diagram – Deep Well Pump 0B – 0WW01PB	H
6E-0-4030WW05	Schematic Diagram – Essential service Water Cooling Tower 0A & 0B Circulating Water Make-up Valves 0WW019A & B	E
6E-1-4001A	Station One Line Diagram	P
6E-1-4001E	Station Key Diagram	O
6E-1-4002E	Single Line Diagram – 120V AC ESF Instrument Inverter Bus 111 and 113, 125V DC ESF Distribution Center 111	K
6E-1-4007A	Byron - Unit 1 - Key Diagram 480V ESF Substation Bus 131X (1AP10E)	M
6E-1-4010A	Key Diagram – 125V DC ESF Distribution Center Bus 111 (1DC05E) Part 1	M

DRAWINGS

Number	Description or Title	Revision
6E-1-4010B	Key Diagram – 125V DC ESF Distribution Center Bus 111 (1DC05E) Part 2	G
6E-1-4010C	Key Diagram – 125V DC Non Safety Related Distribution Panel 113 (1DC05EB)	K
6E-1-4030DC05	Schematic Diagram – 125 VDC ESF Distribution Center, Bus 111, Part 1, 1DC05E	U
6E-1-4030IP01	Schematic Diagram 7.5KVA Fixed Frequency Inverter for Instrument Bus 111 (1IP05E)	0
6E-1-4030RC31	Schematic Diagram - Reactor Coolant System High Pressure & Low Temperature Control & Alarms	G
6E-1-4030RH02	Schematic Diagram - Residual Heat Removal Pump 1B - 1RH01PB	N
6E-1-4030RY14	Schematic Diagram - Pressurizer Pressure & Level Control Safety Related & Non-Safety Related (Div 12)	F
6E-1-4030RY17	Schematic Diagram - Pressurizer Power Relief Valves - 1RY455A & 1RY456; Pressurizer Relief Tank Primary Water Supply Isolation Valve - 1RY8030; Pressurizer Relief Tank Drain Isolation Valve 1RY8031	V
6E-1-4031RC26	Loop Schematic Diagram - Reactor Coolant System Cold Overpressurization System Control 1A & 1D Control Cabinet 5 & 6	S
6E-1-4031RY15	Loop Schematic Diagram - Pressurizer Pressure & Level Control Cabinet 6 (1PA06J) Part 1	O
6E-1-4031RY19	Loop Schematic Diagram - Pressurizer Pressure Safety Valve Discharge Temp & Pressure Control (ITE-0464) Control Cabinet 7 (1PA07J)	F
M-42 Sh. 6	Diagram of Essential Service Water	BC
M-60 Sh. 5	Diagram of Reactor Coolant	AO
M-2042 Sh. 5	P&ID/C&I Diagram ESS Service Water System - SX	F
6E-0-1003	Duct Runs, Outdoor Plan, Southeast Area	AC
6E-0-1004	Duct Runs, Outdoor Plan, Southwest Area	Y
6E-0-1009	Duct Runs, Sections	F
6E-0-3502	Electrical Installation, ESW Cooling Tower 0A Plan – Switchgear Room, Elev. 874’-6”	AZ
6E-0-3502CT1	Conduit Tabulation, ESW Cooling Tower 0A Plan – Switchgear Room, Elev. 874’-6”	T
6E-0-3502D01	Electrical Installation, ESW Cooling Tower 0A Switchgear Room Partial Plans and Sections	N
6E-0-3507	Electrical Installation, ESW Cooling Tower 0B Plan – Switchgear Room, Elev. 874’-6”	BN
6E-0-3507CT1	Conduit Tabulation, ESW Cooling Tower 0B Plan – Switchgear Room, Elev. 874’-6”	Y
6E-0-3507D01	Electrical Installation, ESW Cooling Tower 0B Switchgear Room Partial Plans and Sections	W
6E-0-4030SX01	Schematic Diagram, Essential Service Water Cooling Tower 0A, Fan 0A	V
6E-0-3680	Duct Run Routing Outdoor – West of Station	AC

DRAWINGS

Number	Description or Title	Revision
6E-0-4030SX02	Schematic Diagram, Essential Service Water Cooling Tower 0A, Fan 0B	U
6E-0-4030SX03	Schematic Diagram, Essential Service Water Cooling Tower 0A, Fan 0C	U
6E-0-4030SX04	Schematic Diagram, Essential Service Water Cooling Tower 0A, Fan 0D	W
6E-0-4030SX05	Schematic Diagram, Essential Service Water Cooling Tower 0B, Fan 0E	V
6E-0-4030SX06	Schematic Diagram, Essential Service Water Cooling Tower 0B, Fan 0F	W
6E-0-4030SX07	Schematic Diagram, Essential Service Water Cooling Tower 0B, Fan 0G	W
6E-0-4030SX08	Schematic Diagram, Essential Service Water Cooling Tower 0B, Fan 0H	W
6E-1-4001A	Station One Line Diagram	P
6E-1-4006B	Key Diagram, 4160V ESF Switchgear Bus 142	J
6E-1-4008AN	Key Diagram, 480V ESW Cooling Tower ESF MCC 132Z1	R
6E-1-4012A	Key Diagram, 120 Vac Instrument Bus 111	W
6E-1-4018B	Relaying & Metering Diagram, 4160 ESF Switchgear Bus 142	U
6E-1-4030AP115	Schematic Diagram, Tripping Circuit, 480V ESW Cooling Tower MCC 131Z1A, 132Z1A	A
6E-1-4030RY17	Schematic Diagram, Pressurizer Power Relief Valve 1RV456	V
6E-1-4030SI02	Schematic Diagram, Safety Injection Pump 1B	N
6E-1-4030SI14	Schematic Diagram, Containment Sumps 1A and 1B Isolation Valves SI8811A & B	Q
6E-1-4031VP11	Loop Schematic Diagram [containment inside/outside differential pressure]	K
M-61, Sh. 1B	Diagram of Safety Injection	AX
M-136, Sh. 1	Diagram of Safety Injection	BB
M-63, Sh. 1A	Diagram of Fuel Pool Cooling and Clean up	BI
S-1404	Refueling Water Storage Tank Sections & Details	I
M-60, Sh. 8	Diagram of Reactor Coolant	AA
98Z512-001-2, Sh. 1	Pressurizer PORV Air Relief Valve	0
M-60, Sh.5	Diagram of Reactor Coolant	AO

10 CFR 50.59 DOCUMENTS (Screenings/Safety Evaluations)

Number	Description or Title	Date
6G-97-0110	DCP 9600355 ESW Cooling Tower Basin Level Switch	7/3/97
EC385829	Tornado Missile Design Basis for the Essential Service Water Cooling Tower	0
6G-11-004	Tornado Missile Design Basis for the Essential Service Water Cooling Towers	2/9/12
EC385951	Multiple Spurious Operation – Scenario 14, 1SI8811A/B	12/9/11
6E-05-0172	UFSAR Change Package (DRP) 11-052	11/16/05

10 CFR 50.59 DOCUMENTS (Screenings/Safety Evaluations)

Number	Description or Title	Date
6E-15-035	Increase Pressurizer PORV tank Operating Pressure to Increase Margin for PORV Operation (Unit 1)	0
6H-00-0155	Technical Requirements Manual (TRM) Revision to Delete TLCO 3.4.a, “Pressurizer Safety Valves-Shutdown”	9/19/00

MISCELLANEOUS

Number	Description or Title	Date or Revision
	IST Program Plan – Service Water System	8/26/14
Standing Order 15-020	Emergency Operating Procedure Cold Leg Recirc.	5/15/15
DW-09-004	ERG Feedback	2/27/09
	Stewart & Stevenson Certificate of Conformance for Battery Chargers Serial No. 2165, 2167, 2170, 2174, 4 Batterrie 20 Cells/Set and 8 Battery Racks, Purchase Order No. 203731	11/4/81
06EN003246	FLT Series Flex Switch – Flow, Level, Temperature Switch Monitor	2
01492090-03	Level 3 OPEX Evaluation – NRC IN 2013-05: Battery Expected Life and Its Potential Impact on Surveillance Requirements	5/16/13
CQD-009436	Seismic Qualification Test Report for Nife Ni-Cad Batteries H-410 (1,2 AF01EA-A, EA-B, EB-A, EB-B/0SX02EA, EB-A, EC-A, ED-A	8/17/83
CQD-012527	Review of Seismic Qualification Test Report for Battery Chargers (1&2 DC03E, 04E)	10/2/13
CQD-049161	Justification for the Application of Permatex Form A Gasket with EPT Diaphragms	1
CQD-200164	Dynamic Qualification of Battery Chargers 0SX02EA-1 through 0SX02ED-1; 1,2AF01EA-1 and 1,2AF01EB-1	5/29/86
NEC-06-6066	Procurement of Safety Related 125 Volt Batteries	B
604990-70-F1	Reliance Electric Dimension Sheet [SX Cooling tower fan motor data sheet]	4/4/78
EQ-GEN023	EQ Binder for NAMCO EA180 limit switches	13
EC-397415	EQ Evaluation – Pressurizer PORV Diaphragm Design Pressure	0
EQER-06-98-002	EQ Evaluation for PORVs 1(2) FSV-RY-455A & 1(2)FSV-RY-456	2/29/99
	Low Temperature Protection (LTOP) System Evaluation for Byron and Braidwood Units 1 and 2 Measurement Uncertainty Recapture (MUR) Power Uprate Program	9/7/10
Simulator Work Request 13961	PZR PORV Testing reveals lower than design flow	4/25/12
	Byron Unit 1 Pressure and Temperature Limits Report	3/14
EC 381986	Summary of the Design and Licensing Basis for Inadvertent ECCS Actuation at Power	0

MODIFICATIONS

Number	Description or Title	Date or Revision
EC394865	Ultimate Heat Sink Capability with Loss of Essential Service Water Cooling Tower Fans	2
EC385829	UHS Capability with Loss of SX Fans Due to Tornado Missiles	2/14/12
M6-1(2)-87-142	Install Fan Cooling to Instrument Power Inverter Cubicles	10/17/90
EC385951	Multiple Spurious Operation – Scenario 14, 1SI8811A/B	12/9/11
EC388735	Detailed Review of FC Purification System for Use of Non Safety Related Portion Connected to Safety Related Piping	0
EC396016	Increase U1 Pressurizer PORV Accumulator Tank Operating Pressure to Increase number of PORV Open/Close Cycles from Accumulator	0

OPERABILITY EVALUATIONS

Number	Description or Title	Date
13-001	Capacity of the Pressurizer PORV Air Accumulator During Natural Circulation Cooldown	5
13-007	Ultimate Heat Sink Capability with Loss of Essential Service Water Cooling Tower Fans	1

PROCEDURES

Number	Description or Title	Revision
1BOA PRI-5	Control Room Inaccessibility	108
1BOA ELEC-5	Local Emergency Control of Safe Shutdown Equipment	106
0BOA PRI-7	Loss of Ultimate Heat Sink Unit 0	1
1BOA PRI-7	Essential Service Water Malfunction Unit 1	106
1BEP ES-1.3	Transfer to Cold Leg Recirculation Unit 1	204
1BCA-1.2	LOCA Outside Containment Unit 1	200
OP-AA-102-106	Operator Response Time Program	3
OP-BY-102-106	Operator Response Time Program at Byron Station	7
1BOA S/D-2	Shutdown LOCA Unit 1	105
1BOSR XRS-Q1	Unit One Remote Shutdown Panel Quarterly Surveillance	13
1BFR-H1	Response to Loss of Secondary Heat Sink Unit1	203
0BHSR 8.4.2-1	Unit Zero Comprehensive Inservice Testing (IST) Requirements for Essential Service Water Makeup Pump 0A	8
0BHSR SX-1	Unit 0 0A SX Makeup Pump Battery Bank A Capacity Test	0
0BHSR SX-5	0A SX Makeup Pump Battery Bank D Capacity Test	0
0BISR 7.a.4-200	Calibration of Essential Service Water Cooling Tower Basin 0A Level Switch (SX)	7
0BOSR Z.7.a.2-1	Unit Common Deepwell Pump Operability Monthly Surveillance	1
0BOSR 7.9.6-1	Essential Service Water Makeup Pump 0A Monthly Operability Surveillance	32
0BVSR SX-1	Unit 0 0A SX Makeup Pump Battery Bank A Capacity Test	3
0BVSR SX-4	Unit 0 0A SX Makeup Pump Battery Bank D Capacity Test	3
0BVSR WW-1	Biennial Deep Well Pump Structure Inspection	2
1BHSR 8.4.2-1	Unit 1 Bus 111 125V Battery Charger Operability	1

PROCEDURES

Number	Description or Title	Revision
1BHSR 8.4.3-1	Unit 1 125 Volt Battery Bank 111 Service Test	3
1BHSR 8.6.6-1	Unit 1 Battery 111 125 Volt Battery Bank 5 Year Modified Performance Test	0 & 2
1BHSR AF-1AA	Unit 1 1B Diesel Aux Feed Pump Battery Bank A Battery A (1AF01EA-A) Capacity Test	1
1BOA ELEC-1	Loss of DC Bus Unit 1	103
1BOSR 8.4-1	125V DC Bus 111 Load Shed When Cross-Tied to DC Bus 211	12
2BHSR 8.4.2-1	Unit 2 Bus 211 125V Battery Charger Operability	1
BISR 3.1.10-206	Pressurizer Pressure Protection Channel II (RY) Test Report Package)	8
BISR 3.1.10-207	Pressurizer Pressure Protection Channel III (RY) Test Report Package)	8
BISR 4.12.8-200	Wide Range Reactor Coolant Pressure Loop 1A Hot Leg (RC)	7
BOP-AP-93	MCC 035-2 Outage	1
BOP SX-3	Essential Service Water Make-up Pump Startup	30
BOP SX17	Shutdown of SX Makeup Pump Battery Chargers	3
BOP SX18	Placing the SX Makeup Pump Battery Chargers in Operation/Equalize	8
CC-AA-308	Control and Tracking of Electrical Load Changes	4
ER-AA-310-1004	Maintenance Rule - Performance Monitoring	13
MA-BY-026-1001	Seismic Housekeeping	2
MA-BY-721-060	125 Volt Battery Bank 18 Month Surveillance	11
MA-BY-721-061	125 Volt Battery Bank Quarterly Surveillance	12 & 15
MA-BY-723-053	Station Battery Charger 18 Month Surveillance	18
MA-BY-723-053-001	0B SX Makeup Pump A Battery Charger 0SX02EA Battery Charger Test	0
MA-BY-723-053-002	0B SX Makeup Pump D Battery Charger 0SX02ED Battery Charger Test	1
MA-BY-723-053-003	0B SX Makeup Pump B Battery Charger 0SX02EB Battery Charger Test	0
MA-BY-723-053-004	0B SX Makeup Pump C Battery Charger 0SX02EC Battery Charger Test	1
MA-BY-723-054	Nickel Cadmium Battery Bank Surveillance	14
0BHSR SX-3	Annual Surveillance for Essential Service Water Cooling Tower Fan Motors	2
0BOSR 7.9.4-1	ESW Cooling Tower Fan Monthly Surveillance	6
1BOSR IP-R1	Instructions to Cycle Instrument Bus 111 Distribution Panel Molded Case Circuit Breakers	0
1BOSR 3.2.9-1	Train A Manual Safety Injection Initiation and Manual Phase A Initiation Surveillance	22
1BOSR 8.9.1-2	Unit 1 ESF Onsite Power Distribution Weekly Surveillance Division 12	10
BOP MP-19	Adjusting Reactive Load	12

PROCEDURES

Number	Description or Title	Revision
ER-AA-300-150	Cable Condition Monitoring Program	1
MA-AA-723-330	Electrical Testing of AC Motors Using Baker Instrument Advanced Winding Analyzer	3
MA-AA-725-102	Preventative Maintenance on Westinghouse Type DHP 4kv, 6.9kv, and 13.8kv Circuit Breakers	8
1BGP-100-5	Plant Shutdown and Cooldown	68
BOP FC-7	Startup of the Purification System to Purify or Recirculate the Refueling Water Storage Tank	13
1BEP ES-0.2	Natural Circulation Cooldown Unit 1	202
BAR 1-12-C4	RCS Press High at Low Temp	2
1BOSR 5.C.3.1	Safety Injection System Cold Leg Flow Balance	3
2BOSR 0.1-4	Unit 2 Mode 4 Shiftly and Daily Operating Surveillance	25
1BOSR 0.1-1,2,3	Unit 1 Mode 1,2,3 TRM and Tech Spec and Non Tech Spec Data Sheet D5	56
BIP 2500-088	Calibration of Refueling Water Storage Tank Outlet Temperature Loop (SI)	5
1BOSR 5.5.8.SI.5-2C	Unit 1 Comprehensive Inservice Testing (IST) Requirements for Safety Injection Pump 1SI01PB	5
1BOSR 5.5.8.SI.5-2a	Unit 1 Group A Inservice Testing (IST) Requirements for Safty Injection Pumps 1SI01PB	1
0BOSR NLO-TRM	Non-Licensed Operator TRM, ISFSI, and NPDES Data Daily Logs	18
1BGP 100-5	Plant Shutdown and Cooldown	68
BOP SX-T2	SX Basin Level Tree	5
BOP SX-11	SXCT Fan Startup	9
BOP SX-12	Makeup to an Essential Service Water Mechanical Draft Cooling Tower	10
0BOA ENV-1	Adverse Weather Conditions	114
1BOA PRI-5	Control Room Inaccessibility	108
1BOA ELEC-5	Local Emergency Control of Safe Shutdown Equipment Unit 1	106
1BEP-1	Reactor Trip or Safety Injection	207
1BEP ES-0.1	Reactor Trip Response	203
1BEP ES-0.2	Natural Circulation Cooldown	202
BOP RH-6	Operation of the RH System In Shutdown Cooling	46
OP-AA-108	Oversight and and Control of Operator Burdens	2
BOP CC-1	Component Cooling Water System Startup	12

SURVEILLANCES (Completed)

Number	Description or Title	Date or Revision
0BHSR SX-1	0A SX Makeup Pump Battery Bank A Capacity Test	6/14/12
0BHSR SX-5	0A SX Makeup Pump Battery Bank D Capacity Test	9/14/12
0BISR 7.a.4-200	Calibration of Essential Service Water Cooling Tower Basin 0A Level Switch (SX)	8/7/14
0BOSR 5.5.8.SX.5-1c	0SX02PA Comprehensive IST Req for SX Makeup Pump	2/5/15

SURVEILLANCES (Completed)

Number	Description or Title	Date or Revision
0BOSR 7.9.6-1	0A SX Makeup Pump Operability Surveillance	3/12/13
0BOSR 7.9.6-1	0A SX Makeup Pump Operability Surveillance	2/4/15
0BOSR 7.9.6-1	0A SX Makeup Pump Battery Bank A Capacity Test	3/11/15
0BVSR SX-1	0A SX Makeup Pump Battery Bank A Capacity Test	10/17/06
0BVSR SX-4	0A SX Makeup Pump Battery Bank D Capacity Test	6/19/06
1BHSR 8.4.2-1	Unit 1 Bus 111 125V Battery Charger Operability Test	11/8/11
1BHSR 8.4.2-1	Unit 1 Bus 111 125V Battery Charger Operability Test	9/17/13
1BHSR 8.4.3-1	111 “A” Train 125V Battery Bank Service Test	3/20/14
1BHSR 8.6.6-1	111 “A” Train 125V Battery Bank 5Yr Capacity Test	4/1/08
1BHSR 8.6.6-1	111 “A” Train 125V Battery Bank 5Yr Capacity Test	9/11/12
BISR 3.1.10-206	Pressurize Pressure Protection Channel 2 Loop 1RY-0456	4/6/15
BISR 3.1.10-207	Pressurizer Pressure Protection Channel 3 Loop 1RY-0457	4/13/15
BISR 4.12.8-200	Cal of Wide Range RC Pressure Loop 1A Hot Leg 1P-406	4/28/14
M A-BY-721-060	125 Volt Battery Bank Quarterly Surveillance	9/11/12
M A-BY-721-060	125 Volt Battery Bank Quarterly Surveillance	3/20/14
M A-BY-721-061	125 Volt Battery Bank 18 Months Surveillance	9/16/12
M A-BY-721-061	125 Volt Battery Bank 18 Months Surveillance	3/22/14
M A-BY-721-061	125 Volt Battery Bank 18 Months Surveillance	9/15/14
M A-BY-721-061	125 Volt Battery Bank 18 Months Surveillance	12/16/14
MA-BY-723-053	EM 18 Month Battery Charger Surveillance – 0B SX M/U Pump 0B Batt Chgr # 0SX02EB-1	1/15/13
MA-BY-723-053	EM 18 Month Battery Charger Surveillance – 0A SX M/U Pump 0A Batt Chgr # 0SX02EA-1	2/6/14
MA-BY-723-053	EM 18 Month Battery Charger Surveillance – 0A SX M/U Pump 0D Batt Chgr # 0SX02ED-1	8/5/14
MA-BY-723-053	EM 18 Month Battery Charger Surveillance – 0B SX M/U Pump 0C Batt Chgr # 0SX02EC-1	3/27/15
MA-BY-723-053-001	0B SX Makeup Pump A Battery Charger 0SX02EA Battery Charger Test	2/4/14
MA-BY-723-053-002	0B SX Makeup Pump D Battery Charger 0SX02ED Battery Charger Test	8/6/14
MA-BY-723-053-003	0B SX Makeup Pump B Battery Charger 0SX02EB Battery Charger Test	1/15/13
MA-BY-723-053-004	0B SX Makeup Pump B Battery Charger 0SX02EC Battery Charger Test	3/27/15
MA-BY-723-054	Quarterly 24 VDC NiCad Battery Surveillance M/U Diesel SX- 0SX02ED-A	8/5/14
MA-BY-723-054	Quarterly 24 VDC NiCad Battery Surveillance M/U Diesel SX- 0SX02EA-A	9/5/14

SURVEILLANCES (Completed)

Number	Description or Title	Date or Revision
MA-BY-723-054	Quarterly 24 VDC NiCad Battery Surveillance M/U Diesel SX- 0SX02EA-A	10/30/14
MA-BY-723-054	NiCad Battery Surveillance M/U Diesel SX- 0SX02E	11/6/14
WO01579586	Unit 1 Pressurizer PORV Accumulator Press Decay Test	3/19/14
WO01774289	SI pump ECCS Flow Balance Test (After System Alteration)	10/5/14
WO01243123	OP 2BOSR 5.C.3-2 Unit 2 SI to HL Flow Balance	4/2/10
WO01243120	Unit 1 Safety Injection System Hot Leg Flow Balance	9/4/09
WO01243119	SI pump ECCS Flow Balance Test (After System Alterations)	9/4/09
WO01582134	1SI01PB Comprehensive IST RQMTS For Safety Injection Pump	1/28/14
WO01425077	1SI01PB Comprehensive IST RQMTS For Safety Injection Pump	8/9/12
WO01451296	STT/PIT For 1RY455A and 1RY456	9/28/12
WO01585186	STT/PIT For 1RY455A and 1RY456	2/7/14
PMID 140860	0BOSR 7.9.6-1 0A SX Makeup Pump Operability Review	4/18/13

TRAINING DOCUMENTS

Number	Description or Title	Date or Revision
BY 14-2-2	Requalification Simulator Scenario Guide	1
10-1-5	Requalification Simulator Scenario Guide	0
P1-SPBY-1401	BEP-1, BEP-2	2
OPBYLLORT5	BFR H, Heat Sink Series	8/28/13

WORK DOCUMENTS

Number	Description or Title	Date or Revision
00961518	Replace Entire Solenoid to Meet EQ Requirements – EM ASCO Solenoid Valve Replacement (EQ) – 1FSV-RY456-2	4/1/08
01057719	Test All MCC Breakers in This MCC in a Bus Outage – Assembly 480V RSH MCC 035-2	5/2813
01094421	Replace Float and Equalize Voltage Adjustment Potentiometer	11/29/11
01490541	111 “A” Train 125 V Battery Charger Operability Test	9/18/13
01536066	Essential Service Water Cooling Tower Level 0SX-064 IM Calibration	3/3/14
01558514	B1R19 Replace 111 ESF Batteries	3/29/14
01578627	Test Replace Actuator Hose 1RY456	3/14/14
01599481	Calibration of Wide Range RC Pressure Loop 1A Hot Leg Pressure Loop 1RC-0406	4/28/14
01600072	Clean/Inspect/Check Connections on DC Bus/Panel 111 and Perform Therm. on Distr. Panel Breakers	3/30/14
01621944	Support Diver Insp./Cleaning RSH South 0B Intake/SED PM ID 30	6/25/13
01652815	211 “A” Train 125 V Battery Charger Operability Test	5/14/14

WORK DOCUMENTS

Number	Description or Title	Date or Revision
01017127	Perform Dynamic Baker Testing – 1SI01PB Motor	8/26/08
01085998	Perform Static Baker Test and MA-AA-723-310 Inspection of SX Cooling Tower Fan Motor 0SX03CC	4/27/09
01117942	PM for 4kV Bus 142, breaker ACB 1425Z	9/21/09
01119375	Lightning Protection System 5 Year Inspection [Includes Document 1 attachment to WO]	11/18/09
01120491	PM for 4kV Bus 142, breaker ACB 1424	9/29/09
01129028	Inspection of SX Cooling Tower Fan Motor 0SX03D	10/28/09
01136617	PM for 4kV Bus 142, breaker ACB 1422	3/15/09
01141049	Perform Static Baker Test and MA-AA-723-310 Inspection of SX Cooling Tower Fan Motor 0SX03CB	3/19/10
01216011	Perform Dynamic Baker Testing – 1SI01PB Motor	8/26/10
01258194	Calibration of OLS-XS097	1/6/11
01265167	PM for 4kV Bus 142, breaker ACB 1421	10/26/11
01287321	Inspection of SX Cooling Tower Fan Motor 0SX03CE	9/1/11
01299949	Containment Inside/Outside DP Loop 1VP-231	6/30/11
01343409	Inspection of SX Cooling Tower Fan Motor 0SX03CH	11/21/11
01367641	PM for 4kV Bus 142, breaker ACB 1SI01PB	2/21/12
01372340	PM for 4kV Bus 142, breaker ACB 1422	11/11/12
01382271	Perform Static Baker Test and MA-AA-723-310 Inspection of SX Cooling Tower Fan Motor 0SX03CC	6/12/12
01384474-01	Inspection of SX Cooling Tower Fan Motor 0SX03CF	11/26/12
01380551-01	Inspection of SX Cooling Tower Fan Motor 0SX03CA	6/8/12
01393782	Inspection of SX Cooling Tower Fan Motor 0SX03CG	10/30/11
01401180	Calibration of OLS-XS097	8/24/12
01419437	PM for 4kV Bus 142, breaker ACB 1425Z	9/23/12
01419758	Test All MCC 132Z1 Breakers – Oden Testing	9/23/12
01420365	PM for 4kV Bus 142, breaker ACB 1424	9/23/12
01421751	Unit 1 Train A Manual SI and Manual Phase A Initiation Surveillance	9/11/12
01433378-01	Inspection of SX Cooling Tower Fan Motor 0SX03CD	3/12/13
01453350	Containment Inside/Outside DP Loop 1VP-231	3/19/15
01471461	Calibration of OLS-XS096	9/6/11
01473594-01	Perform Static Baker Test and MA-AA-723-310 Inspection of SX Cooling Tower Fan Motor 0SX03CB	5/17/13
01480666-01	Testing of Power Cables 2AP178	4/20/13
01486337	Calibration of OLS-XS096	2/8/13
01538412	PM for 4kV Bus 142, breaker ACB 1423	11/30/13
01564018-01	Testing of Power Cables 1AP178 (North SX towers)	3/18/14
01569220	Calibration of OLS-XS097	6/2/14
01585654-02	Testing of Power Cables 2AP183 (Bus 242, Cubicle 20)	10/6/14
01615167	Calibration of OLS-XS096	8/8/14
01621573-01	Perform Surveillance of SX Cooling Tower Fan Motor 0SX03CE	9/16/14
01639602	PM for 4kV Bus 142, breaker ACB 1421	11/19/14

WORK DOCUMENTS

Number	Description or Title	Date or Revision
01644724-01	Perform Surveillance of SX Cooling Tower Fan Motor 0SX03CH	11/20/14
01652671	PM for 4kV Bus 142, breaker ACB 1SI01PB	3/29/15
01667453	Calibration of 1SX-015 Loop	2/17/15
01680518	Calibration of 1SX-016 Loop	3/31/15
01543156	Calibration of 2SX-015 Loop	2/12/14
01716477	Calibration of 2SX-016 Loop	3/23/15
01734645-01	SX Cooling Tower Fan Motor Surveillance - 0SX03CG	11/4/14
01734645-02	SX Cooling Tower Fan Motor Surveillance & Triannual Inspection - 0SX03CG	11/5/14
01760801	PM for 4kV Bus 142, breaker ACB 1423	1/30/15
01805922	ESW Cooling Tower Fan Monthly Surveillance	3/10/15
01419750	Replace Actuator Diaphragm	9/20/12
01515448	Refueling Water Storage Tank Outlet Temp LOOP 1SI-058	2/24/14
01186461	Refueling Water Storage Tank Outlet Temp LOOP 1SI-058	4/21/10
01544629	Calibration of Refueling Water Storage Tank (RWST) level	9/20/13
01374939	Calibration of Refueling Water Storage Tank (RWST) level	2/28/12
00915331	Minor Leakage from 0A WW Pump Well Head	8/20/08
00768385	0B WW PP 10 Year Rebuild	11/09/06
01754077	Received 0A SX Make Up Pp Trouble alarm	7/17/14
00921203	SXCT Fan Assembly Replacement EC 356417	8/23/12
00921198	SXCT Fan Assembly Replacement EC 356417	1/10/07
01634644	Replace Start Contactor Relay K1B at 0SX02PA-B	4/17/13
01682260	Support Diver Insp/Cleaning SXCT South 0B Basin	10/31/14
01691008	Support Diver Insp/Cleaning SXCT South 0A Basin	11/14/14

LIST OF ACRONYMS USED

ΔCDF	Delta Core Damage Frequency
AC	Alternating Current
ACIT	Action Tracking Item
ADAMS	Agencywide Document Access Management System
CA	Corrective Action Tracking Item
CAP	Corrective Action Program
CAQ	Condition Adverse to Quality
CCDP	Conditional Core Damage Probability
CDBI	Component Design Bases Inspection
CFR	*Code of Federal Regulations*
CNMT	Containment
CS	Containment Spray
CV	Chemical and Volume Control
DBA	Design Basis Accident
DC	Direct Current
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
EC	Engineering Change
ECCS	Emergency Core Cooling System
EOP	Emergency Operating Procedure
ERG	Emergency Response Guideline
FSAR	Final Safety Analysis Report
gpm	Gallons per Minute
IMC	Inspection Manual Chapter
IN	Information Notice
IR	Inspection Report
LCO	Limiting Condition for Operation
LERF	Large Early Release Frequency
LLC	Limited Liability Corporation
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LTOP	Low Temperature Overpressure Protection
MCC	Motor Control Center
MOV	Motor-Operated Valve
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NOV	Notice of Violation
NPSH	Net Positive Suction Head
NRC	U.S. Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
PARS	Publicly Available Records System
PM	Preventive Maintenance
PORV	Power-Operated Relief Valve
PRA	Probabilistic Risk Assessment
RASP	Risk Assessment Standardization Project
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RIS	Regulatory Issue Summary
RWST	Refueling Water Storage Tank

SAPHIRE	Systems Analysis Programs for Hands-on Integrated Reliability Evaluations
SDP	Significance Determination Process
SI	Safety Injection
SPAR	Standardized Plant Analysis Risk
SR	Surveillance Requirement
SRA	Senior Reactor Analyst
SSC	System, Structure, and Component
SSDPC	Safety Systems Design, Performance and Capability Inspection
SX	Emergency Service Water
SXCT	Emergency Service Water Cooling Tower
TORMIS	Tornado Missile Risk Evaluation Methodology
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
UHS	Ultimate Heat Sink
URI	Unresolved Item
VAC	Volts Alternating Current
VDC	Volts Direct Current
WOG	Westinghouse Owners Group

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 Nos. 50–454; 50–455
License Nos. NPF–37; NPF–66

Enclosures:
(1) Notice of Violation
(2) IR 05000454/2015008; 05000455/2015008;

cc w/encl: Distribution via LISTSERV®

DISTRIBUTION w/encl:
Kimyata MorganButler
RidsNrrDorlLpl3-2 Resource
RidsNrrPMByron Resource
RidsNrrDirsIrib Resource
Cynthia Pederson
Darrell Roberts
Richard Skokowski
Allan Barker
Carole Ariano
Linda Linn
DRPIII
DRSIII
Jim Clay
Carmen Olteanu
ROPreports.Resource@nrc.gov

ADAMS Accession Number ML15203A042
☒ Publicly Available ☐ Non-Publicly Available ☐ Sensitive ☒ Non-Sensitive
To receive a copy of this document, indicate in the concurrence box "C" = Copy without attach/encl "E" = Copy with attach/encl "N" = No copy

OFFICE	RIII		RIII		RIII		RIII	
NAME	MJones for NFeliz-Adorno:cl		CLipa:					
DATE	07/21/15		07/21/15					

OFFICIAL RECORD COPY