UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
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January 11, 2013

Mr. Larry Weber
Senior Vice President and
Chief Nuclear Officer
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

SUBJECT: D. C. COOK NUCLEAR POWER PLANT, UNITS 1 AND 2, COMPONENT DESIGN BASES INSPECTION 05000315/2012007; 05000316/2012007(DRS)

Dear Mr. Weber:

On December 31, 2012, the U.S. Nuclear Regulatory Commission, (NRC) completed a Component Design Bases Inspection, (CDBI) at your D. C. Cook Nuclear Power Plant, Units 1 and 2. The enclosed report documents the results of this inspection, which were discussed on December 31, 2012, with Mr. M. Carlson, and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, one NRC-identified finding of very low safety significance was identified. The finding involved a violation of NRC requirements. However, because of its very low safety significance, and because the issue was entered into your Corrective Action Program, the NRC is treating this issue as a Non-Cited Violation (NCV) in accordance with Section 2.3.2 of the NRC Enforcement Policy.

If you contest the subject or severity of this NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the D.C. Cook Nuclear Power Plant.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Ann Marie Stone, Chief
Engineering Branch 2
Division of Reactor Safety

Docket Nos. 50-315; 50-316
License Nos. DPR-58; DPR-74

Enclosure: Inspection Report 05000315/2012007; 05000316/2012007(DRS)
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	05000315; 05000316
License Nos:	DPR-58; DPR-74
Report No:	05000315/2012007; 05000316/2012007(DRS)
Licensee:	Indiana Michigan Power Company
Facility:	D. C. Cook Nuclear Power Plant, Units 1 and 2
Location:	Bridgman, MI
Dates:	July 23, 2012, through December 31, 2012
Inspectors:	C. Tilton, Senior Reactor Engineer, Lead R. Baker, Operations Engineer C. Brown, Reactor Engineer, Electrical J. Corujo-Sandín, Reactor Engineer, Mechanical W. Sherbin, Mechanical Contractor G. Skinner, Electrical Contractor
Approved by:	Ann Marie Stone, Chief Engineering Branch 2 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000315/2012007; 05000316/2012007(DRS); 07/23/2012 – 12/31/2012; D.C. Cook Nuclear Power Plant, Units 1 and 2; Component Design Bases Inspection (CDBI).

The inspection was a 3-week onsite baseline inspection that focused on the design of components. The inspection was conducted by regional engineering inspectors and two consultants. One Green finding was identified by the inspectors. The finding was considered a Non-Cited Violation (NCV) of NRC regulations. The significance of inspection findings are indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using IMC 0609, "Significance Determination Process," dated June 2, 2011. Cross-cutting aspects are determined using IMC 0310, "Components within the Cross Cutting Areas," dated October 28, 2011. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy dated June 7, 2012. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to ensure sufficient water volume in the condensate storage tank when both units' auxiliary feedwater (AFW) pumps are aligned to a single condensate storage tank (CST.) Specifically, the licensee failed to perform a calculation to demonstrate sufficient volume and level to prevent net positive suction head and vortex issues when a single CST is providing water to all six AFW pumps as allowed by procedures. The licensee's corrective action included performing a formal calculation and increasing the available water volume in the CST when both units' AFW pumps are cross-tied.

  The performance deficiency was determined to be more than minor because it was associated with the Mitigating System Cornerstone attribute of equipment performance and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding screened as of very low safety significance (Green) because it was a design deficiency confirmed not to result in loss of operability. Specifically, the licensee performed an operability determination which concluded the actual useable tank level during the previous 12 months had been sufficient. The inspectors determined the cause of this finding did not represent current licensee performance and, thus, no cross-cutting aspect was assigned. (Section 1R21.3b.(1))

B. Licensee-Identified Violations

No violations were identified.

REPORT DETAIL

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

1R21 Component Design Bases Inspection (71111.21)

.1 Introduction

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

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

.2 Inspection Sample Selection Process

The inspectors used information contained in the licensee's PRA and the D.C. Cook Nuclear Power Plant Standardized Plant Analysis Risk-Model to identify a scenario to use as the basis for component selection. The design basis accident scenario selected was steam generator tube rupture coincident with a loss of offsite power. Based on this scenario, a number of risk significant components were selected for the inspection.

The inspectors also used additional component information such as a margin assessment in the selection process. This design margin assessment considered original design 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, 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 inspectors also identified procedures and modifications for review that were associated with the selected components. In addition, the inspectors selected operating experience issues associated with the selected components.

This inspection constituted 19 samples as defined in Inspection Procedure 71111.21-05.

.3 Component Design

a. Inspection Scope

The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR), Technical Specifications (TS), design basis documents, drawings, calculations and other available design basis information, to determine the performance requirements of the selected components. The inspectors used applicable industry standards, such as the American Society of Mechanical Engineers (ASME) Code, Institute of Electrical and Electronics Engineers (IEEE) Standards and the National Electric Code, to evaluate acceptability of the systems' design. The inspectors also evaluated licensee actions, if any, taken in response to NRC-issued operating experience, such as Bulletins, Generic Letters (GLs), Regulatory Issue Summaries (RISs), and Information Notices (INs). The review was to verify 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 the component condition and tested capability was consistent with the design bases and was appropriate may include installed configuration, system operation, detailed design, system testing, equipment and environmental qualification, equipment protection, component inputs and outputs, operating experience, and component degradation.

For each of the components selected, the inspectors reviewed the maintenance history, preventive maintenance activities, system health reports, operating experience-related information, vendor manuals, electrical and mechanical drawings, and licensee Corrective Action Program documents. Field walkdowns were conducted for all accessible components to assess material condition and to verify 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 14 components were reviewed:

- Unit 1 East Essential Service Water (ESW) Pump (1-PP-7E): The inspectors reviewed design analyses associated with the ESW pump capacity, net positive suction head (NPSH), and minimum flow to verify the equipment's capacity to perform its required functions. The pump's test procedures and recent results were reviewed to verify the actual capability of the installed equipment. The potential susceptibility of the pump to external flooding events was reviewed to verify the capability of the pump to perform its required function. Also reviewed was the potential susceptibility of the pump in the event of a Turbine Building high energy line break (HELB.) In addition, the inspectors reviewed the capability of the ESW pump to serve as the safety-related source of backup water supply to the AFW system. The inspectors reviewed control circuitry for the motor to determine whether manual and automatic functions were consistent with the design bases. The inspectors reviewed protective relaying schemes and calculations to determine whether the motor was adequately protected and whether it was susceptible to spurious tripping.

  Unit 1 East Essential Service Water (ESW) Pump Discharge Strainer (1-OME-34E): Inspectors reviewed design and licensing bases documents for the component. Included in the items reviewed was the safety-related backflush function of the strainer and the differential pressure setpoints associated with the backflush function. In addition, the inspectors reviewed maintenance and operational history of the component. A sample of procedures associated with the components' operations was also reviewed.

- Unit 1 East Component Cooling Water (CCW) Heat Exchanger Outlet Shutoff Valve (1-WMO-733): The inspectors reviewed the design and licensing bases associated with the component. Calculations related to weak link components and thrust analyses were reviewed. The inspectors reviewed recent test and maintenance records for the component. Automatic functions/actuations were reviewed in order to ensure the component could meet its safety-related functions. The inspectors also reviewed control circuitry for the motor to determine whether manual and automatic functions were consistent with the design bases. The inspectors reviewed the thermal overload (TOL) protection scheme for the motor operated valve (MOV), including drawings, calculations, and test procedures to determine whether it was consistent with NRC Regulatory Guide 1.106, Position C.2. The inspectors reviewed voltage and torque calculations for the MOV to determine whether they were conservative and properly incorporated appropriate correction factors specified in limitorque technical data.

- Unit 1 East Motor Driven Auxiliary Feedwater Pump (1-PP-3E): The inspectors reviewed design analyses associated with the motor driven auxiliary feedwater (MDAFW) pump's capacity. The licensing bases documents for the AFW system were reviewed, in particular as they relate to the UFSAR credited source of water (CST) and the safety-related backup source of water (ESW). The seismic qualification of the system and some of the support systems (CST) were also reviewed by the inspectors. A sample of procedures used during normal and emergency conditions was reviewed. The inspectors emphasized their review on the interaction between the AFW system and its safety-related backup source of water, including review for potential air binding of the pump as a result of the swap of water sources. The inspectors reviewed industry experience issues associated with potential fouling of the pump's minimum flow recirculation line. The maintenance and testing history of the pump were also reviewed. The inspectors reviewed control circuitry for the motor to determine whether manual and automatic functions were consistent with the design bases. The inspectors review protective relaying schemes and calculations to determine whether the motor was adequately protected and whether it was susceptible to spurious tripping.

- 4kV Bus (T-11D): The inspectors reviewed bus loading calculations to determine whether the 4.16kV system had sufficient capacity to support its required loads under worst case accident loading and grid voltage conditions. The inspectors reviewed the design of the degraded voltage protection scheme to determine whether it afforded adequate voltage to safety-related devices at all voltage distribution levels. This included review of degraded voltage relay setpoint calculations, and a review of the degraded voltage logic scheme. The inspectors reviewed the overcurrent protection scheme for the 4.16kV buses including drawings and calculations to determine whether loads were adequately protected and immune from spurious tripping. The inspectors reviewed calculations and procedures used to determine operability of the offsite power supplies to the 4160V buses. This included review of switchyard voltage drop criteria and the interface agreements between the station and the transmission system operator. The inspectors reviewed calculations and drawings for the Reserve Auxiliary Transformer load tap changer (LTC) control to determine whether it would have adequate voltage to operate under design basis conditions. The inspectors reviewed 125Vdc system voltage drop calculations to determine whether 4.16kV bus circuit breakers had adequate control voltage.

- Condensate Storage Tanks (1-TK-32, 2-TK-32): The inspectors reviewed design calculations to ensure the CST contained sufficient volume to meet the TS requirement and to ensure vortexing would not occur prior to operators taking manual actions to lineup the AFW pumps to their safety-related source of water. Each unit's AFW pumps' suction pipe is normally aligned to a CST, and there is no automatic swap-over to a safety-related source of water for pump suction in the event of a loss of CST. The inspectors reviewed the seismic, tornado wind, and tornado missile licensing basis requirements for the tanks. The inspectors reviewed structural calculations to verify seismic and tornado wind design capabilities of the tanks to maintain the required water volume during these natural hazards. In addition, the inspectors reviewed operating procedures that would be entered during a loss of CST to ensure design bases requirements for AFW pumps' suction are maintained.

- Emergency Diesel Generator (EDG) (1-CD): The inspectors reviewed the load voltage drop calculation, maximum and minimum voltage profile, and DC field flashing circuit design to ensure that the EDG met the design requirements. The inspectors also verified that the EDG would properly start under degraded voltage conditions. Surveillance test results were reviewed to ensure TS requirements were met. The inspectors reviewed the adequacy and appropriateness of design assumptions and calculations related to EDG protection and relay coordination during test mode and during emergency operation. The EDG output breaker control logic diagrams were reviewed to verify the breaker tripping and closing logic was consistent with design basis description and interlocking requirements. The inspectors reviewed applicable CRs, maintenance activities, and EDG monitoring in accordance with the station blackout (SBO) rule. The inspectors also reviewed the latest operating experience with the system engineer.

- Unit 1 Turbine-Driven AFW Pump 250 Vdc N Train Battery: To ensure that the N-train battery conformed to the design bases, the inspectors performed a walkdown of the turbine-driven (TD) AFW pump room and the N train battery room. The inspectors observed the N train battery, which supplies control power for the pump and noted the room temperature was high. The inspectors reviewed the battery load test results and completed surveillances to ensure the electrolyte levels were being maintained and the effects of the high temperature had been factored into the projections on battery life and performance. The inspectors confirmed the battery room had not approached the maximum allowable room temperature. The inspectors also reviewed the applicable operating procedures and the recent condition reports (CRs) and operability evaluations. The inspectors also reviewed the minimum voltage for the battery during the most limiting conditions of operation.

- <u>Unit 1 250 Vdc Transfer Cabinet (1-TDAB)</u>: To ensure the transfer cabinet and the breakers met the design requirements, the inspectors reviewed the vendor ratings for the Bus and breakers against the breaker specifications. The inspectors also reviewed the short circuit calculations, fuse ratings, and the coordination scheme to ensure proper sizing of protection devices. In addition, the inspectors reviewed the related operations procedures and recent condition reports and operability evaluations.

- <u>Unit 1 250 Vdc Battery and Busses (1-BATT-AB)</u>: The inspectors performed a walkdown of the battery room and the surrounding areas to inspect for discrepant conditions. The inspectors reviewed the battery sizing and loading calculation to verify all loads were accounted for, that the loads did not exceed the battery bank capacity, and that the calculated load profile bounded all accident scenarios. The inspectors also reviewed the short circuit calculation to confirm the maximum calculated short circuit current available under faulted conditions did not exceed the equipment rating and the protective fuses were adequately sized to isolate the fault and protect the affected equipment. In addition, the inspectors reviewed voltage drop calculations to verify the minimum voltage available at the equipment for the duration of the duty cycle was sufficient to ensure the proper operation of the equipment under limiting operating and environmental conditions. The review included verification that the battery tests conformed to the design and TS requirements, enveloped the calculated load profile for the duration of the duty cycle, confirmed the battery capacity exceeded the minimum capacity required under limiting conditions, and were capable of detecting battery degradation. One-line diagrams and wiring schematic diagrams were also reviewed to verify proper configuration of the 250 Vdc electrical distribution systems. The inspectors reviewed battery charger sizing calculations to verify the chargers were capable of carrying the continuous load after a design basis accident (DBA) and would be able to recharge the batteries to full capacity within the specified period. Additionally, the inspectors reviewed battery chargers tests to verify their capability of performing their intended function under design condition before their scheduled replacement. The inspectors reviewed system health reports, selected preventive and corrective maintenance history, as well as selected corrective action system documents to verify potential degradation was monitored or prevented and that corrective actions were appropriate and performed in a timely manner.

- <u>Steam Generator (SG) Power Operated Relief Valves (PORVs) (MRV-213, -223, -233,and 243)</u>: The inspectors reviewed the UFSAR, TS, applicable plant calculations, and drawings to identify the design bases requirements of the SG PORVs to determine if they were subject to common cause failure. The inspectors examined system health reports, and records of surveillance testing for the pneumatic operating components. Additionally, the inspectors reviewed station operating and off-normal response procedures to verify design bases requirements had been adequately translated into procedural instructions. The inspectors reviewed design bases documentation and drawings of the instrument air system to verify the support function provided to the SG PORVs was consistent with design requirements. The inspectors reviewed calculations for a postulated design bases steam generator tube rupture (SGTR) event to verify the ability of operators to perform required actions within the time frames assumed in the plant's design and licensing basis accident analysis. The inspectors reviewed preventative maintenance and corrective maintenance history for trends, and reviewed recent corrective action documents to ensure problems were identified and corrected. In addition, the inspectors reviewed the sources of power, and control schemes to determine consistency with the design bases and to verify intended operation during accident conditions.

- Unit 1 Train A Pressurizer PORV (1-NRV-153): The inspectors reviewed the translation of design bases into surveillance testing of the pressurizer PORV to ensure it is capable of performing as required in SGTR event. The inspectors also reviewed the design and testing of the backup pneumatic supply to the PORV actuator to ensure there is a sufficient quantity at adequate pressure to stroke the PORV when required on a loss of normal plant air. Plant alarm response procedure for low PORV accumulator pneumatic pressure was also reviewed to ensure operators have sufficient time to align additional pneumatic supply sources. The inspectors also reviewed thermal hydraulic calculations which determined the required stroke time of the PORV, and ensured testing demonstrated that stroke time requirements were met. In addition, the inspectors reviewed the sources of power, and control schemes to determine consistency with the design bases and to verify intended operation during accident conditions.

- Unit 1 TDAFW Pump Discharge Check Valve (1-135): The inspectors reviewed system design criteria, selected drawings, and maintenance and test procedures for the Unit 1 TDAFW pump discharge check valve (1-135). The inspectors also performed walkdowns and reviewed corrective action documents. Additionally, the inspectors reviewed inservice test basis documents and associated test results, including forward flow. The inspectors also reviewed calculations that provided the bases for the inservice test acceptance criteria. Recent check valve internal inspection/examination results were also reviewed.

- Unit 1 TDAFW Pump Room Coolers (1-HV-AFP-T1AC And T2AC): The inspectors reviewed analyses addressing the maximum potential TDAFW room temperatures under accident and SBO conditions and room cooler sizing calculations. The review verified the capability of required equipment in the room to perform their required functions with elevated room temperatures. The inspectors reviewed recent flow testing to ensure adequate cooling water flow to the room coolers and reviewed recent heat exchanger tube inspection results to verify heat transfer capability is maintained. The inspectors reviewed design documents which determined tube plugging limits and verified these limits were in plant maintenance procedures. The inspectors also performed walkdowns, reviewed corrective action documents and seismic qualification documents for the room coolers and structural supports.

b. Findings

(1) Non-Conservative Condensate Storage Tank (CST) Cross-Tie NPSH Calculation

Introduction: A finding of very-low-safety significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” was identified by the inspectors for the failure to ensure sufficient water volume in the condensate storage tank when both units’ AFW pumps are aligned to a single CST. Specifically, the licensee failed to perform a calculation to show that there would be adequate NPSH and vortexing issues would be prevented when a single CST is providing water to all six AFW pumps as allowed by procedures.

Description: On August 24, 2012, while reviewing calculation MD-12-CST-002-N, "Operation of the AFW System Using the Condensate Storage Tank of the Other Unit," the inspectors noted the calculation concluded that a level of seven feet above the centerline of the discharge pipe to the AFW pumps was required to provide adequate NPSH when cross-tied to the other unit's CST. The inspectors also noted this calculation assumed only three AFW pumps in a single unit in operation drawing water from the opposite unit's CST.

The inspectors also reviewed Procedure 1(2)-OHP-4022-055-003, "Loss of Condensate to Auxiliary Feedwater Pumps." The inspectors noted this procedure allowed for cross-tie operation of both units' AFW pumps (three on each unit) from a single CST whenever a CST level is less than 15 percent. The inspectors confirmed this value did not account for single CST providing water to all six AFW pumps. The 15 percent value represented operation of a single CST providing three AFW pumps in one unit.

The inspectors were concerned because a 15 percent level in a single CST could potentially result in not having sufficient NPSH and potentially introducing vortexing in the system rendering all six AFW pumps inoperable. The licensee issued AR 2012-10381 to document the inspectors' concerns.

After further assessment, the licensee performed a NPSH calculation given the configuration in question (ability to operate all six AFW pumps from one CST) and determined the required level is about 6 feet higher than the minimum level determined from calculation MD-12-CST-002-N and correlates to 38 percent. In addition, the licensee researched the minimum CST level during plant operation within the past 12 months, and determined the lowest level in any CST was 58 percent. Therefore, although the vulnerability existed, the pumps had been operable since CST level had been greater than 38 percent.

Analysis: The inspectors determined the failure to ensure sufficient water volume in the condensate storage tank when both units' AFW pumps are aligned to a single CST 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 System Cornerstone attribute of equipment performance and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to perform a calculation to demonstrate adequate NPSH and vortexing issues would be prevented when a single CST is providing water to all six AFW pumps does not ensure the availability and reliability of the AFW system to provide its accident mitigating function.

The inspectors 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 cornerstone, 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." The finding screened as of very low safety significance (Green) because it was a design deficiency confirmed not to result in loss of operability. Specifically, the licensee performed an operability determination which concluded that in the past 12 months the volume and level in each CST was adequate to allow for a single CST to supply both units' AFW pumps without affecting operability of the AFW systems.

The inspectors determined the cause of this finding did not represent current licensee performance and, thus, no cross-cutting aspect was assigned.

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

Contrary to the above, as of August 24, 2012, the licensees' design control measures failed to translate applicable design basis into procedures. Specifically, the procedure for cross-tie operation of a CST to both units' AFW pumps did not consider the minimum volume of water required in the CST to prevent NPSH and vortexing issues. Because this violation was of very low safety significance and was entered into the licensee's CAP as AR 2012-10381, this violation is being treated as a Non-Cited Violation, consistent with Section 2.3.2 of the NRC Enforcement Policy.
(NCV 05000315/2012007-01; 05000316/2012007-01, Non-conservative CST Cross-Tie NPSH Calculation).

(2) Qualification Basis for Safety-Related Relays and Motor-Starter Contactors

Introduction: The inspectors identified an unresolved item (URI) regarding the licensee's actions to maintain or extend the qualification basis for safety-related relays and motor-starter contactors used in safety-related applications greater than vendor service-life recommendations. Specifically, the licensee did not have a time based replacement program for aging active components that ensured that the SSCs were replaced or the service life was evaluated and extended before the vendor recommended service life was exceeded.

Description: As part of the review of the electrical systems and systems, structures and components (SSCs), the inspectors noted the certificate of compliance for the HFA relays stated a 41-year service life. A 21-year service life was noted for the HEA and HGA relays and a 10-year service life from the time of manufacture for the Agastat relays. The inspectors noted the licensee was not managing the replacement of the safety-related (SR) relays and contactors associated with the electrical systems and SSCs to prevent exceeding the manufacturers' recommended service lives. Based on the inspectors' questions, the licensee initiated AR 2012-9701, "Condition Not Adverse to Quality," on August 8, 2012, to investigate the impact of service life on the relay and contactor operability.

At the time of the inspection, none of the installed HFA relays had exceeded the 41-year service life. This issue is considered an unresolved item pending consultation with personnel in the Office of Nuclear Reactor Regulation (NRR) and further NRC review of the licensee response. (URI 05000315/2012007-02, 05000316/2012007-02; Qualification Basis for Safety-Related Relays and Motor-Starter Contactors).

(3) Concerns with Periodic Design Basis Testing of Installed Relays and Motor-Starter Contactors

Introduction: The inspectors identified an unresolved item (URI) regarding periodic design basis testing of installed relays and motor-starter contactors. Specifically, the licensee had neither a periodic testing program nor had any record of testing critical SSCs to ensure the SSCs continued to meet their respective design criteria (minimum pickup voltage, minimum drop-out voltage, and timing tests).

Description: As part of the review for exceeding the recommended relay service life, the inspectors determined the licensee did not periodically test the safety-related relays and motor-starter contactors to the design specifications and had not since the plant started power operations.

The inspectors noted Regulatory Guide (RG) 1.33, “Quality Assurance Program Requirements (Operation),” stated additional documents were to be included in a licensee’s Quality Assurance Program, specifically RG 1.30 “Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment” and ANSI Standard N45.2.4-1972, (also known as IEEE Std 336-1971). Section C.3 of RG 1.30 states, “Although Subdivision 1.1 of ANSI N45.2.4-1972 states that the requirements promulgated apply during the construction phase of a nuclear power plant, these requirements are also to be considered applicable for the installation, inspection, and testing of instrumentation and electric equipment during the operation phase of a nuclear power plant.” In addition, the inspectors noted Section 3.3 of IEEE 336, “Procedures and Instructions,” states “documents shall be kept current by controlled supervision so that installation, inspections, and tests are performed in accordance with the latest approved design and manufacturers’ instructions.”

The inspectors were concerned the assumptions in the degraded voltage calculations were challenged in that the licensee has not performed any testing to ensure that the design ratings for the SR relays and contactors would continue to operate at or below the manufacturers’ design ratings. Only functional testing at full voltage has been performed to check the operation. In response, the licensee initiated AR 2012-11028, “2012 CDBI – Periodic Testing of HGA Relays,” on September 6, 2012.

This issue is considered an unresolved item pending consultation with NRR personnel to confirm the testing requirements and further NRC review of the licensee response. (URI 05000315/2012007-03, 05000316/2012007-03; Concerns with Periodic Design Basis Testing of Installed Relays and Motor-Starter Contactors).

.4 Operating Experience

a. Inspection Scope

The inspectors reviewed five operating experience issues to ensure that NRC generic concerns had been adequately evaluated and addressed by the licensee. The operating experience issues listed below were reviewed and are considered inspection samples:

- GL 1979-36, "Adequacy of Station Electric Distribution Systems Voltages";
- GL 2006-02, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power”;

- IN 2004-01, “Auxiliary Feedwater Pump Recirculation Line Orifice Fouling – Potential Common Cause Failure”;

- GL 1991-13, “Request for Information Related to the Resolution of Generic Issue 130, ‘Essential Service Water System Failures at Multi-Unit Sites’ Pursuant to 10 CFR 50.54 (F)”; and

- RIS 2008-14, “Use of TORMIS Computer Code for Assessment of Tornado Missile Protection.”

b. Findings

No findings of significance were identified.

.5 Modifications

a. Inspection Scope

The inspectors reviewed one permanent plant modifications related to selected risk significant components to verify 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:

- DCP 4690, 250 Vdc Fuse Replacement Project

b. Findings

No findings of significance were identified.

.6 Operating Procedure Accident Scenarios

a. Inspection Scope

The inspectors performed a detailed review of the procedures listed below associated with the selected scenario, steam generator tube rupture. The procedures were compared to UFSAR content, design assumptions, regulatory requirements, and training materials to assure constancy. For the procedures listed, operator actions were reviewed for reasonableness, in plant actions were walked down with a licensed operator, and any interfaces with other departments were evaluated.

The following operating procedures were reviewed in detail:

- 1(2)-OHP-4023-E-0, Reactor Trip or Safety Injection;

- 1(2)-OHP-4023-E-3, Steam Generator Tube Rupture;

- 1(2)-OHP-4023-ECA-3.1, SGTR With Loss of Reactor Coolant-Subcooled Recovery Desired;

- 1(2)-OHP-4023-ES-3.1, Post-SGTR Cooldown Using Backfill; and

- 1(2)-OHP-4023-ES-3.2, Post-SGTR Cooldown Using Blowdown.

For the following selected time critical operator actions, the inspectors performed detailed walkthroughs of specific procedure steps, including observing the performance of required actions in the station's simulator and local actions in the plant by appropriate plant operators to assess operator knowledge level, adequacy of procedures, and availability of special equipment where required.

The following operator actions were assessed in detail:

- Actions to isolate AFW flow to steam generator with ruptured tube;
- Actions to isolate steam flow from steam generator with ruptured tube;
- Actions to initiate reactor coolant system (RCS) cooldown following steam generator tube rupture;
- Actions to initiate RCS depressurization following steam generator tube rupture;
- Actions to terminate safety injection following steam generator tube rupture; and
- Actions to align backup nitrogen to and locally operate the steam generator power operated relief valves on the intact steam generators.

b. Findings

(1) Concerns with Ensuring Margin to Overfill in a Ruptured SG

Introduction: The inspectors identified an URI related to the adequacy of the licensee's emergency operating procedures (EOPs) which mitigate the consequences of a SGTR. Specifically, procedures EOP 1(2) OHP-4023-E-3, "Steam Generator Tube Rupture," did not provide adequate actions to mitigate the consequences of a SGTR in sufficient time to prevent overfilling the ruptured steam generator which could lead to exceeding calculated offsite doses.

Description: During the week of July 23, 2012, the inspectors reviewed the licensing bases and plant response to a SGTR event. During the postulated design basis event, operators prevent overfill of the ruptured SG using the SG PORVs. During a SGTR, the normal source of motive force for the SG PORVs is air supplied by the compressed air system (CAS). The CAS compressors are powered from the offsite power supply and are non-safety-related. Assuming a concurrent loss of offsite power (LOOP) to the station, the only available source of immediate and remote (from the control room) motive force is air supplied by the control air compressor (CAC) since it could be powered from an EDG. The CAC is also non-safety-related. The inspectors also noted the backup Nitrogen system could be manually aligned to provide motive force for the SG PORVs.

The facility's EOPs 1(2) OHP-4023-E-3, "Steam Generator Tube Rupture," Step 7, directs the operators to perform a rapid RCS cooldown by fully opening the SG PORVs on the 3 intact SGs. The margin to overfill (MTO) analysis assumes this step occurs on demand (the SGTR PORVs open immediately as soon as the operators manipulate the valves from the control room). Once the SG PORVs are full open, the MTO analysis assumes it takes 12 minutes to complete the RCS cooldown. This is a calculated (modeled) value which takes into account plant specific characteristics.

If the SG PORVs do not open, the EOPs direct operators to establish backup Nitrogen. Because aligning Nitrogen to the SG PORVs is a manual operation requiring multiple manipulations outside the control room, the inspectors were concerned the additional time to complete these actions would result in a longer time to complete the RCS cooldown than assumed in the MTO analysis and the affected SG could overfill.

On Friday, July 27, 2012, the inspectors requested the licensee to provide a non-licensed operator to perform a walkthrough of the procedures for establishing backup Nitrogen to locally open the SG PORVs. Although no significant issues were discovered during performance of the evolution, the operator required 13 minutes to establish backup Nitrogen to the auxiliary building and 14 minutes to place the local control stations in service and open the SG PORVs, a total of 27 minutes.

The inspectors were concerned because this additional 27 minutes was not accounted for in the MTO analysis. Specifically, the MTO analysis assumes a total of 12 minutes for RCS cooldown to occur from the moment the operators get direction to open the SG PORVs (assumed to occur immediately) until commencement of RCS depressurization. The MTO analysis calculates it will take the operators and the plant a total of 52 minutes to mitigate the SGTR accident (no more RCS flow through the ruptured SG) and thus prevent overfilling the affect SG with a calculated MTO of approximated $8ft^3$. The MTO analysis has no margin to accommodate the additional 27 minutes for RCS cooldown and therefore the procedure fails to potentially prevent overfilling the ruptured SG.

In response to the inspector's concerns, the licensee implemented immediate compensatory actions to ensure (1) backup Nitrogen was continuously available to the auxiliary building and (2) an operator would be immediately dispatched to commence lining up backup Nitrogen to the SG PORVs if a SGTR with a concurrent LOOP occurred while the EDG or CAC was unavailable.

The licensee disagreed with the inspectors' assumption with respect to the initial conditions of a LOOP affecting both units. The licensee believes their licensing basis for a SGTR is a concurrent LOOP in the affected unit and the unaffected unit maintains an intact source of offsite power. In the licensee's scenario, the affected unit's SG PORVs would still have a source of immediate remote (from the control room) motive force via the compressed air system. On the contrary, the inspectors believe the licensee's design bases accident is a SGTR coincident with a LOOP to the station (both units). To resolve this issue, the inspectors requested support from NRR through a concurrence Task Interface Agreement (TIA) to determine the licensing bases for this event. In a TIA memorandum dated December 7, 2012, NRR concluded the licensing bases included a LOOP for both units (ML13011A382).

Although the inspectors received a response from NRR, the issue remained open awaiting additional licensing bases information from the licensee. This issue is considered an unresolved item (URI) pending further review this information (URI 05000315/2012007-04, 05000316/2012007-04; Concerns with Ensuring Margin to Overfill in a Ruptured SG).

(2) Concerns with Operability of SG PORVs with the Control Air Compressor Unavailable

Introduction: The inspectors identified an URI related to the operability of the SG PORVs as defined in TS 3.7.4 when the associated CAC was unavailable and incapable of providing its required support function. Specifically, the licensee may have failed to declare the affected unit's SG PORVs inoperable and complete the TS required actions when the non-safety-related CAC was made unavailable thus affecting the SG PORVs capability to mitigate the consequences of a SGTR coincident with a loss of offsite power to the station, in sufficient time to prevent overfilling the ruptured steam generator.

Description: The inspectors identified an issue related to the definition of operability of the SG PORVs as specified by the TS. The most limiting design bases accident for the SG PORVs, is a tube rupture event. During this accident, the operators prevent overfill of the ruptured steam generator using the SG PORVs. Assuming a coincident loss of offsite power (LOOP) to the station (affecting both units), the only readily available source of pneumatic motive force for the SG PORVs is the unit-specific CAC, which has the capability of being powered from onsite emergency power (EDGs). The TS bases for the SG PORVs (B 3.7.4) states the Control Air System (the system composed of the CAC) provides the normal air supply for pneumatic control. Each unit-specific CAC is not safety-related and is not subject to a TS limiting condition for operation. Therefore, these compressors could be unavailable (i.e., for maintenance) for an indeterminate length of time, consistent with the performance goals established for the maintenance rule, regardless of the unit's current mode of operation. With the exception of emergent repair maintenance, all preventive maintenance is performed on the CACs when the units are online, in Mode 1.

As defined in the facility's Unit 1 and Unit 2 TS, "A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function are also capable of performing their related support function(s)." The electrical and control air appurtenances for the SG PORVs are non-safety grade and do not have an associated TS operability requirement. However, since the electrical control power and control air system are credited to ensure the SG PORVs will operate to perform the mitigating function of cooling the RCS during the SGTR accident with a LOOP, the inspectors concluded this equipment is required to be functional for the SG PORVs to be considered OPERABLE.

The inspectors indentified several occasions when the CAC was non-functional; however, the licensee did not declare the PORVs inoperable. Specifically, a review of the facility's unavailability records for the Unit 1 and Unit 2 CACs from January 1, 2000 to August 20, 2012 identified 13 instances (5 associated with Unit 1 and 8 associated Unit 2) when the unit's CAC was unavailable for greater than 24 hours, the TS allowed outage time for two or more SG PORVs being inoperable. In three of those instances for Unit 1 (April 18, 2001, for 79.8 hrs, November 23, 2003, for 133.2 hrs, and April 7, 2008, for 108.9 hrs) and two of those instances for Unit 2 (February 12, 2003, for 71.8 hrs and January 16, 2006, for 103.5 hrs) the unavailability time of the CAC was in excess of the total TS allowed outage times of 54 hours to place the unit in a mode where the limiting condition of operation does not apply.

As stated in Section 1R21.6(b)(1), the licensee disagreed with the inspectors' initial assumption with respect to the initial conditions of a LOOP affecting both units. The licensee believes their licensing basis for a SGTR is a concurrent LOOP in the affected unit and the unaffected unit maintains an intact source of offsite power. In the licensee's scenario, the affected unit's SG PORVs would still have a source of immediate remote (from the control room) motive force via the compressed air system. On the contrary, the inspectors believe the licensee's design basis accident is a SGTR coincident with a LOOP to the station (both units). To resolve this issue, the inspectors requested support from NRR through a concurrence Task Interface Agreement (TIA) to determine the licensing bases for this event. In a TIA memorandum dated December 7, 2012, NRR concluded the licensing bases included a LOOP for both units (ML13011A382).

Although the inspectors received a response from NRR, the issue remained open awaiting additional licensing bases information from the licensee. This issue is considered an unresolved item (URI) pending further review of this information (URI 05000315/2012007-05, 05000316/2012007-05; Concerns with Operability of SG PORVs with Control Air Compressor Unavailable).

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

.1 Review of Items Entered Into the Corrective Action Program

a. Inspection Scope

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

b. Findings

No findings of significance were identified.

4OA6 Meeting(s)

.1 Exit Meeting Summary

On December 31, 2012, the inspectors presented the inspection results to Mr. M. Carlson, and other members of the licensee staff. The licensee acknowledged the issues presented. Several documents reviewed by the inspectors 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

M. Carlson, VP Site Support Services
S. Lies, Plant Manager
M. Scarpello, Regulatory Affairs and Performance Improvement Department Manager
W. Hodge, I&C Design Engineering Supervisor
S. Mitchell, Licensing Activity Coordinator (Compliance)

Nuclear Regulatory Commission

A. M. Stone, Chief, Engineering Branch 2, DRS
J. Ellegood, Senior Resident Inspector
P. LaFlamme, Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened/Closed

05000315/2012007-01; 05000316/2012007-01	NCV	Non-conservative Condensate Storage Tank (CST) Cross-Tie NPSH Calculation (1R21.3(b)(1))

Opened

05000315/2012007-02; 05000316/2012007-02	URI	Qualification Basis for Safety-Related Relays and Motor-Starter Contactors (1R21.3(b)(2))
05000315/2012007-03; 05000316/2012007-03	URI	Concerns with Periodic Design Basis Testing of Installed Relays and Motor-Starter Contactors (1R21.3(b)(3))
05000315/2012007-04; 05000316/2012007-04	URI	Concerns with Ensuring Margin to Overfill in a Ruptured SG (1R21.6(b)(1))
05000315/2012007-05; 05000316/2012007-05	URI	Concerns with Operability of SG PORVs with Control Air Compressor Unavailable 1R21.6(b)(2)

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 stated in the body of the inspection report.

CALCULATIONS

Number	Description or Title	Revision
01-E-N-ELCP-120-004	120 VAC Distribution panel Control Circuit Voltage Drop	1
01-E-N-PROT-TOL-001	600v System Continuous Duty Safety-related Motor Thermal Overload (TOL) Heater Selection	12
12-E-N-PROT-RLY-001	4KV Electrical Protection Coordination Study	1
12-E-N-PROT-TOL-MOV-001	Generic Thermal Overload Heater Sizing Calculation for AC Motor Operated Valves Within the GL 89-10 Program	1
1-EN-N-ELCP-250-008	250Vdc Battery 1N System Analysis	0
1-E-N-ELCP-600-003	600 Motor Control Center (MCC) Control Circuit Voltage Drop	5
12-E-S-ELCP-765-002	Cook Load Flow Program (CKOLF) Description and Validation	2
1-E-N-ELCP-4KV-001-Model	Unit 1 4 kV/600 V Load Control Model Calculation	10
1-E-N-ELCP-4KV-001-VOLT	Unit 1 Voltage Adequacy	4
1-E-N-PROT-RLY-002	4kV Safety-related Motors Phase Instantaneous Relay (PJC) Settings Calculation, Unit 1	0
1-E-N-PROT-RLY-002A	4kV Safety-related Motors Phase Instantaneous Relay (PJC) Settings Calculation, Unit 1	0
1-E-N-PROT-RLY-003	Degraded Grid and Loss of Voltage Relay Setting Calculation	0
1-SV-2A-1	SQUG- Steam Generator OME-3-1 Safety Valve 2A	1
2-E-N-DCP-4247-001	Fan Thermal Overload Evaluation During HELB Transient	1
2-E-N-ELCP-4KV-001	Unit 2 4 kV/600 V Load Control Calculation	4

CALCULATIONS

Number	Description or Title	Revision
DC-D-02-ESW-19	Piping and Pipe Support Analysis of Unit 2 Condensate and Essential Service Water Piping System from the Condensate Storage Tank to the three suction nozzles of the Auxiliary Feedwater Pumps.	1
DIT-0-01161-01	ESW Loads – Concurrently Supplying CTS & AFW Supply	5/15/00
DIT-B-01074-00	ESW Strainer Differential Pressure Switch and Alarm Setting	4/18/00
DIT-B-02317-05	ESW Flow Verification Test Target Flows for 01-OHP-4030-119-022FV & 02-OHP-4030-219-022FV	4/19/08
DIT-S-01153-00	Minimum Flow for the Unit 1 and Unit 2 ESW Pumps	2/9/03
DIT-S-01153-01	ESW Pump Flow Recommendation	2/10/03
DIT-S-01153-02	ESW Pump Flow Recommendation	2/10/10
MD-01-ESW-084-N	Torque Setup Calculation for 1-WMO-733 and 1-WMO-737	1
MD-12-AFW-001-N	Auxiliary Feedwater System Design Basis Analysis	2
MD-12-AFW-001-N	Aux. Feedwater System Design Bases Analysis	2
MD-12-CA-004-S	Determination of Available Pressurizer PORV Strokes Using the Auxiliary Air Supply	1
MD-12-CST-002-N	Operation of Auxiliary Feedwater System Using the CST of the Other Unit	0
MD-12-ESW-076-N	ESW Pump NPSH Available and Submergence	2
MD-12-ESW-109-N	ESW Supply to the Suction of the AFW pumps	0
MD-12-HV-018-N	Aux. Feedwater Pump Room and Hallway Heat Load Calculation	1
MD-12-RCS-021-N	Maximum Differential Pressure Calculation For Pressurizer Power Operated Relief Valves 1(2)-NRV-151, 152 and 153	0
MD-12-RCS-022-N	Actuator Capability Calculation For Pressurizer Power Operated Relief Valves 1(2)-NRV-151, 152 and 153	2
PS-4KVP-001	4 KV Safety Motor Electrical Protection	5
PS-4KVP-001	4 KV Safety Motor Electrical Protection	5
PS-4KVP-003	Ground Relay Settings 4-kV ESS and BOP Buses	1
PS-4KVP-004	Relay Setting Basis – 69kV Alt. Pwr. Supply	1
PS-4KVP-005	Unit & Reserve Feed Phase-Overcurrent Relay Settings	0
SD-990825-047	Seismic/Weak Link Torque Calculation for I-WMO-733	3
SD-990930-004	Probability of Tornado Missile Strike on Targets at D. C. Cook Nuclear Plant	5

CALCULATIONS

Number	Description or Title	Revision
SEWS-1-TK-32	SQUG Seismic Evaluation of CST 1-TK-32	0
SEWS-2-TK-32	SQUG Seismic Evaluation of CST 2-TK-32	0
SQUG 1-BATT-AB	Seismic Qualification of Station AB Batteries	0
TH-00-03	D.C. Cook Unit 2 Steam Generator Tube Rupture with Operator Actions	0
TH-00-05	AFW Pump Room Heat-Up Temperatures	0
TH-00-06	D.C. Cook Unit 1 Steam Generator Tube Rupture with Operator Actions	0
TH-04-11	Impact of New EOP Footnotes on Steam Generator Tube Rupture Overfill	0
TH-04-11-ADD	MDAFP BHP Impact on SGTR Overfill Analysis	0

CORRECTIVE ACTION DOCUMENTS GENERATED DUE TO THE INSPECTION

Number	Description or Title	Date
GT 2012-9163	Update Report No. PRA-NB-SY-MS	7/26/12
AR 2012-9209	Update Vendor Manuals in Calc 1-E-N-PROT-TOL-001	7/27/12
AR 2012-9233	SGTR DBA Analysis May Not Be Met with Some Unavailable Equipment	7/27/12
GT 2012-9592	Procedure note needs to be revised	8/6/12
GT2012-9681	1-OHP-4025-LS-4 Section LS-4-3 Enhancement	8/8/12
AR 2012-9701	CNAQ – Condition Not Adverse to Quality	8/8/12
AR 2012-10342	MOV Voltage Analysis	8/23/12
AR 2012-10362	Loss of Controlled Document	8/23/12
AR 2012-10381	2012 CDBI Enhancement to Loss of CST Inventory Response	8/23/12
AR 2012-10385	2012 CDBI – ESW to AFW Void Mod or Analysis Required	8/23/12
AR 2012-10381	2012 CDBI Enhancement to Loss of CST Inventory Response	8/23/12
AR 2012-10398	PJM Manual #3 Does Not Reflect Cook NPIRs	8/24/12
AR 2012-10425	GL 2006-2 Response	8/24/12

CORRECTIVE ACTION DOCUMENTS GENERATED DUE TO THE INSPECTION

Number	Description or Title	Date
AR 2012-10547	Application of Limitorque Technical Bulletin 93-03	8/27/12
AR 2012-10556	Potential for Inconsistent Application of LCO 3.0.6	8/27/12
GT 2012-10519	2012 CDBI: CST Tornado Qualification Question	8/27/12
AR 2012-11028	2012 CDBI Periodic Testing of HGA Relays	9/6/12

CORRECTIVE ACTION DOCUMENTS REVIEWED DURING THE INSPECTION

Number	Description or Title	Date
CR 99-02960	Procedures 01(and 02)-0HP4022.0550003 do not Alert Operations to a Change in Flow When ESW is Valved Over to Supply AFW	2/17/99
CR 05177018	The Licensing and Design Basis Documentation That Establishes The Safety Classification and Seismic Qualification Basis for the Condensate Storage Tanks is Not Easily Retrievable or Clear to Interpret.	6/26/05
AR 2008-13401	Calculation TH-95-01 Does Not Model Gap At Column Line A	5/9/07
AR 2008-13401	Calculation TH-95-01 Does Not Model Gap At Column Line A	5/9/07
AR 2008-41539 – 03	Reportability eval re: Screen House Wall/HELB Issue	11/13/08
AR 2008-49389	Vibes on East MDAFP Greater Than Alert Limits	4/9/09
AR 2008-49389	Vibes on East MDAFP greater than alert limits	4/9/09
AR 2008-49714	Unit 1 CAC Running at Higher Amps	4/13/2009
GT 2008-51617	Project Required for Identifying Alternate Method of Providing ESW Min Flow Protection	5/18/09
AR 2008-52630	Control Air Compressor Amperage is Higher Than Normal	6/5/2009
GT 2008-61464	Union Leaking in Safety Valve 1-SV-1B-4	12/2/09
AR 2008-62364	2-OME-42 Motor Elevated Vibration Level	12/18/2009
GT 2010-8756	Clarification of DBD for Transfer of Buses	8/30/10
GT 2010-9468	Revise Calculations 1(2)-E-N-ELCP-4KV-001-VOLT "Unit 1/2-Voltage Adequacy Analysis" to Better Describe the Basis for the Allowed Differences in T-bus Voltage When Connected to the UAT/RATs Versus 69KV (EP)	9/15/10

CORRECTIVE ACTION DOCUMENTS REVIEWED DURING THE INSPECTION

Number	Description or Title	Date
AR 2010-9960	TOL's Trip Prematurely	9/27/10
AR 2011-0619	Loss of ESW Header Pressure on pump swap	1/15/11
AR 2011-0632	U1 EESW Pump Test Pump DP Less Than Low Alert Limit	1/16/11
AR 2011-1788	1-WMO-733 Overtorque	2/9/11
GT 2011-2277	Evaluate Practice Of Starting All AFW pps On Loss Of A MFP.	2/21/11
GT 2011-2590	1-OME-34E, Extent Of Cause Inspection	2/28/11
GT 2011-3224	OE32888 – Voltage Cal. Failed to Include LTC Voltage	3/14/11
GT 2011-4445	Prepare Calculation for RAT LTC Control Power	4/12/11
AR 2011-5154	Air Voids in AFW Pump Emergency Suction Source from ESW	4/29/11
AR 2011-5492	When Starting Unit 1 CAC for Surveillance it Tripped and Restarted	5/6/2011
GT 2011-8565	OE-Evaluation of NRC IN 2011-14, Component Cooling Water System Gas Accumulation	7/26/11
AR 2011-9439	AFI CM.2-1 – Time Credited Operator Actions	8/18/2011
AR 2011-9779	U-1 CAC Run Scheduled for Sunday with Recording Vibrations on Monday	8/28/2011
AR 2012-10271	Evaluate Calculation 12-E-N-ELCP-250-009	8/22/12
AR 2011-10411	Air Void detected in Alt Suction line to U2 TDAFW pump	9/14/11
AR 2011-12301	T11D7 East CCP Breaker Failed Timing Per 1-OHP-4030-132-217A	10/21/11
AR 2011-13233	1-PP-3E (East Motor Driven Auxiliary Feedwater Pump) Oil	11/10/11
AR 2011-14229-02	Tube Plugging Limits	11/21/11
AR 2012-0710	Print Does Not Match Field Wiring for 2-OME-42 Motor	1/16/2012
AR 2012-6200	Potential Risk to 4kV bus From Undetected Voltage Unbalance	5/11/12
GT 2012-6873	Packing Leak 2-NMO-152	5/29/12
AR 2012-7514	1-OME-34E East ESW Strainer Gate is Leaking 1 gpm.	6/14/12

CORRECTIVE ACTION DOCUMENTS REVIEWED DURING THE INSPECTION

Number	Description or Title	Date
AR 2012-8833	1-MRV-212, No. 1 SG Stop Valve Dump Valve Lost Power	7/19/12
AR 2012-8834	1-MRV-222, No. 2 SG Stop Valve Dump Valve Failure	7/19/12
AR 2012 - 9358	Electrical Load Flow Calculations	8/8/12

DRAWINGS

Number	Description or Title	Revision
1-2-AEP-GRAV-L-24989	General Assy. And Orientation for Hardtop Floating Roof Condensate Storage Tank	0
Dresser # CP-1795	Flanged Inlet 1200 psig. Safety Valve	0
E-1300	DC Cook Station One Line Diagram	19
E-1700	DC Cook Station One Line Diagram	5
OP-1-12001-80	Main Auxiliary One-Line Diagram Bus “A” and “B” Engineered Safety System (Train “B”)	80
OP-1-12002-63	Main Auxiliary One-Line Diagram Bus “C” and “D” Engineered Safety System (Train “A”)	63
OP-1-12030-34	MCC Aux One-Line 600V Bus 11C, 11D, Engineered Safety System (Train “A”)	34
OP-1-12031-34	MCC Aux One-Line 600V Bus 11C, 11D, Engineered Safety System (Train “A”)	34
OP-1-12032-20	MCC Aux One-Line 600V Bus 11C, 11D, Engineered Safety System (Train “A”)	20
OP-1-12033-33	MCC Aux One-Line 600V Bus 11C, 11D, Engineered Safety System (Train “A”)	33
OP-1-12034-4	MCC Aux One-Line 600V Bus 11C, 11D, Engineered Safety System (Train “A”)	4
OP-1-12035-33	Distr Pnl One-Line 600V Bus 11C, 11D Engineered Safety System (Train “A”)	33
OP-1-12050-26	120/208V AC Control Room Instrumentation Cabinets “CRID-I” Thru “CRID-IV” Engineered Safety System	26
OP-1-12051-31	120V AC Critical Control Room Power Cabinets “CCRP-1” Thru “CCRP-3”	31
OP-12-5118B-46	Flow Diagram Nitrogen and Hydrogen Gas Reactor System Units 1 and 2	46
OP-1-5102D	Flow Diagram, Containment Control Air, 85 No. and 50 No. Ring Headers, Unit 1	34
OP-1-5105A-36	Flow Diagram, Main Steam, Unit 1	36
OP-1-5105A-60	Flow Diagram, Aux. Feedwater, Unit 1	60
OP-1-5105D-10	Flow Diagram, Steam Generating System, Unit 1	10
OP-1-5105D-10	Flow Diagram Steam Generating System Unit No. 1	10

DRAWINGS

Number	Description or Title	Revision
OP-1-5106A-60	Flow Diagram Aux-Feedwater Unit 1	60
OP-1-5113-92	Flow Diagram Essential Service Water	92
OP-1-5113B-1	Flow Diagram, TDAFW Pump Room Coolers, Unit 1	1
OP-1-5120A-60	Flow Diagram Compressed Air System Plant Air Turbine Room Unit 1	60
OP-1-5120R-6	Control Air System, Aux. Bldg. Tapoffs, Unit 1	6
OP-1-5120S-14	Control Air System Auxiliary Bldg. Tapoffs Unit 1	14
OP-1-5120V-6	Control Air System Auxiliary Bldg. Tapoffs Unit 1	6
OP-1-5120X-7	Control Air System Auxiliary Bldg. Tapoffs Unit 1	7
OP-1-5128-29	Flow Diagram, Reactor Coolant, Unit 1	29
OP-1-98002-4	Transformer 101 AB Temperature and Cooling Elementary Diagram	4
OP-1-98003-2	Transformer 101 AB load Tap Changer Motor Drive Control Elementary Diagram	2
OP-1-98043-56	4kV Diesel Generator 1AB A.C.B. Elementary Diagram	56
OP-1-980461-6	4Kv 600V Auxiliary Transformers 11B and 11D Elementary Diagram Sh 2 of 2	6
OP-1-98046-33	4Kv 600V Auxiliary Transformers 11B and 11D Elementary Diagram Sh 1 of 2	33
OP-1-98050-28	Reserve Bus Tran. and Auxiliary Buses Low Voltage Protection Elementary Diagram	28
OP-1-98062-14	120 V.A.C. Distribution Cabinet “CCRP-3” Elementary Diagram	14
OP-1-98063-29	15KVA Static Inverter 1-CCRP-INV and 120VAC Distribution Cab. “CCRP-1” Elementary Diagram	29
OP-1-98063-29	15KVA Static Inverter 1-CCRP-INV & 120VAC Dist Elementary Diagram	29
OP-1-98214-46	Motor Driven Aux. Feedwater Supply Sys. Sheet No. 1 Elementary Diagram	46
OP-1-98273-54	Chemical and Volume Control System Reactor Coolant Charging Elementary Diagram	54
OP-1-98281-49	Emergency Core Cooling (Safety Injection) Sheet No. 1 Elementary Diagram	49
OP-1-982841-21	Emergency Core Cooling (Residual Heat Removal) Sheet #2 Elementary Diagram	21
OP-1-982851-12	Containment Spray System Elementary Diagram Sheet #2	12
OP-1-98404-35	Component Cooling Water System (East) Sh 1 of 3 Elementary Diagram	35
OP-1-98416-24	Essential Service Water System East Sheet No. 2 Elementary Diagram	24
OP-1-98538-17	Steam Line and Feedwater Isolation Elementary Diagram	17
OP-1-98538-17	Steam Line and Feedwater Isolation Elementary Diagram	17
OP-1-98583-11	Steam Generator Relief Valve Position Indicators Elementary Diagram	11
OP-1-985841-4	Steam Pressure Protection Channel 3 Elementary Diagram	4

DRAWINGS

Number	Description or Title	Revision
OP-1-985842-3	Steam Pressure Protection Channel 4 Elementary Diagram	3
OP-1-98701-29	Compressed Air Control Elementary Diagram Sheet No. 1 Unit 1 CIA-42373	29
OP-1-98701-29	Compressed Air Control Elementary Diagram Sheet No. 1 Unit 1	29
RSC1-4097	Relay Diagram 4kV Bus T11D Standby Feed Overload Protection	2
RSC1-4104	Relay Diagram 4kV Bus T11D Emergency Diesel Gen. Overload Protection	1
RSC1-4106	Relay Diagram 4kV Bus T11D Essential Service Water Pump “1E” Motor Protection	3
RSC1-4107	Relay Diagram 4kV Bus T11D Motor Driven Aux. Feedwater Pump “1E” Motor Protection	3
RSC1-4108	Relay Diagram 4kV Bus T11D Standby Feed Overload Protection	2
RSC1-4281	Relay Diagram 4kV Bus T11D Under voltage Protection	5

MISCELLANEOUS

Number	Description or Title	Date or Revision
	Letter from NRC to J. Dolan, D. C. Cook, Subject: Response to 06/13/86 letter	6/25/86
	Maintenance Rule Scoping Document – Compressed Air System	1
12-E-N-ELCP-250-009	Auxiliary Relays – Hand Reset Type HEA	0
2584	Commitment Number Report 2584	10/05/88
AEP Report No. NED-2000-536 REP	Seismic Qualification Test Report for NLI Room Coolers	1
AEP: NRC: 0976	Letter from M. P. Alexich, D. C. Cook Plant to NRC, Subject: Auxiliary Feedwater System Low Suction Pressure Pump Trip	6/13/86
AEP:NRC:00268	Response to NRC GL 79-36	12/17/79
AEP:NRC:00268 C	Grid Degraded Voltage	1/27/81
AEP:NRC:00486 A	Seismic Qualification of Auxiliary Feedwater Systems: GL 81-14	8/28/81
AEP:NRC:0981	Letter Documenting Commitment 3870	7/10/86
AEP:NRC:1104C	Generic Letter 91-13, “Request for Information Related to the Resolution of Generic Issue 130, ‘Essential Service Water System Failures at Multi-Unit Sites’ Pursuant to 10 CFR 50.54 (F)”	3/13/92
AEP:NRC:1246A	DC Cook Response to Generic Letter(GL) 96-01, Testing of Safety-related Logic Circuits	10/28/97

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Number	Description or Title	Date or Revision
AEP:NRC:1246B	DC Cook Response to Generic Letter(GL) 96-01, Testing of Safety-Related Logic Circuits	01/30/99
AEP:NRC;1246	DC Cook Response to Generic Letter(GL) 96-01, Testing of Safety-Related Logic Circuits	04/17/96
AEP-NRC-2011-75	LER 315/2011-002-00	12/20/11
AEP-NRC-2012-61	Enforcement Discretion Regarding Engineered Safety Feature Actuation System Steam Line Isolation Automatic Actuation Logic and Actuation Relays for Steam Generator Stop Valve Dump Valve	7/23/12
C1 101-03	Technical Specification Change Request 4kV Bus Undervoltage Setpoint	11/16/01
C1299-17	DC Cook Supplement to the Generic Letter (GL) 96-01 Response	12/17/99
DB- 12-4KV	Design Basis Document for the 4kV System	3
DB- 12-ESW	Design Basis Document for the ESW System	7
DB-12-AFWS	Design Basis Document for AFW	4
DIT-B-00176-02	Design Input Transmittal; Aux. Bldg. Seismic Response Spectra	1/14/00
DIT-B-00435-00	Updated MOV Data for the Voltage Drop Calculations	0
DIT-B-00621-06	AC Powered Motor Operated Valves (MOV's)	6
DIT-B-00980-01	Unit 2 SGTR Supplemental Analysis Input Assumptions	01
DIT-B-009801-01	Design Input Transmittal; Main Steam and Reactor Coolant System	3/31/00
DIT-B-01061-14	EOP Operator Action Times from Accident Analyses	14
DIT-B-01093-07	Minimum Switchyard Voltage Requirements	7
DIT-B-01399-01	Unit 1 SGTR Supplemental Analysis Input Assumptions	1
DIT-B-02600-00	Minimum Allowable Switchyard Voltage Swings With pre-LOCA Mode 1 Loading on RATs	0
DIT-B-03072-00	Trip Times for Slow Trip Class 30 Eutectic Cutler Hammer Thermal Overload Relay	0
DIT-B-03106-01	Trip Times for Eutectic and Bimetallic Standard Trip Class 20 Cutler Hammer Thermal Overload Relays	1
ECG-EA-5339	Technical Evaluation Report, Adequacy of Station Electric Distribution Systems Voltages, D. C. Cook Nuclear Station, Units 2 and 3, Docket Nos. 50-315 and 50-316	1/31/81
ECP-1-2-O0-14	Emergency Operation Procedure Footnotes	18
EHI-5202	Gas Accumulation Condition Monitoring	3
Flowserve Report No. 10 CFR 21-67	Potentially Reduced Flow Capacity of WKM PORVs	6/7/12
GEH-2024A	Multicontact Auxiliary Relay, Type HFA51	0
GEI-10190G	DC Auxiliary Relays Type HGA	0
GQE No. 101.7	Generic Qualification Guide for ASCO Model 206-832-3RF	0

MISCELLANEOUS

Number	Description or Title	Date or Revision
LER 315/1999-022-01,	Electrical Bus Degraded Voltage Too Low For Safety-related Loads	3/23/00
	Minimum flow requirements for Johnston Service Water Pumps Model 30CC/2 Stage	6/14/90
	Pump and Valve Inservice Test Program for Donald C. Cook Nuclear Plant Fourth Ten Year Interval	1
PJM Manual M03	Transmission Operations	40
PJM Manual M39	Nuclear Plant Interface Coordination	4
PO 01530931	Receipt No. 95518, TYCO Relay AGASTAT Relay Quality Certificate	02/13/09
PO 01545871	Receipt No. 117063, Relay HEA Type Quality Certificate	02/25/11
PO 01551747	Receipt No. 126896, Relay Auxiliary HFA Quality Certificate	04/30/12
PRA-NB-SY-ESW	Essential Service Water System Electrical Power Support Systems	3
RQ-S-2012-CDBI-1	2012 CDBI Scenario No. 1	0
Trend Data	1-FW-135, Turbine Driven Auxiliary Feed Pump PP-4 Discharge Check Valve Trend Data	6/2009 - 6/2012
Trend Data	1-NRV-153 Pressurizer OME-4 Train 'A' Pressure Relief Valve Trend Data	6/2009 - 6/2012
VTD-AGAS-0001	ASCO Engineering Information for Solenoid Valves In Nuclear Power Plants	0
VTD-AGAS-0011	AGASTAT E7000 Series Nuclear Qualified Time Delay Relays [PUB. # E70-1]	1
VTD-AREV-0007	AREVA NP, Inc. HK-VR Instruction Book for DC COOK	
VTD-ASCO-0001	ASCO Engineering Information for Solenoid Valves in Nuclear Power Plants	0
VTD-ATWD-0034	Atwood & Morrill Co., Inc. Instruction Manual For Check Valves	2
VTD-CDBA-0001	C&D Technologies Standby Battery Vented Cell Installation and Operating Instructions	4
VTD-CDBA-0005	Condensed Installation and Operating Instructions for C and D Standby Batteries (Flooded)	0
VTD-CHAM-0021	Cutler-Hammer Data Sheets for CN15 and CN55 A-C Contactors	0
VTD-CHAM-0022	Cutler-Hammer Parts and Instruction Publication for NEMA Size “2” 3-Pole Starter with Standard Trip Eutectic Overload Relay	0
VTD-CHAM-0056	Cutler-Hammer Instruction Manual For Unitrol Motor Control Center [Pub. No. 15412]	2
VTD-CHAM-0087	Cutler-Hammer Technical Information Publication for NEMA Contactors and Starters (Pub. No. 8231)	0

MISCELLANEOUS

Number	Description or Title	Date or Revision
VTD-CHAM-01 03	Cutler-Hammer (Eaton Corporation) Instructions for VR-Series Replacement Breakers for ITE Type 5HK and 15HK [PUB. No. I.B. 94A5990R17]	0
VTD-GENE-0013	General Electric Instructions for Type HFA151 Multi-Contact Auxiliary Relay	3
VTD-GENE-1182	General Electric Instructions for Type HEFA61and HEA62 t Auxiliary Relay	1
VTD-LIMIT-0023	Limitorque Technical Update on Reliance 3-Phase Actuator Motors [Pub. No. 93-03]	0
VTD-MASN-0076	Masoneilan 20000 Series Control Valve Instructions	2
VTD-TATE-0001	ESW Pump Discharge Strainer Vendor Manual	0

MODIFICATIONS

Number	Description or Title	Date or Revision
DCP 4690	250 Vdc Fuse Replacement Project	0
2-DCP-4786	Installation of Flange Sets and Orifice Plates in ESW Lines for AFW Pump Room Coolers	10/2/00

PROCEDURES

Number	Description or Title	Revision
01-IHP-6030-IMP-309	4KV Bus Loss Of Voltage And 4KV Bus Degraded Voltage Relay Calibration	7
01-OHL-4030-SOM-039	Annunciator No. 108, Drop 28	10
1-0HP-4030-119-022FV	ESW Flow Verification	16
12-EHP-4075-TCA-001	Operator Time Critical Actions	0
12-EHP-5030-CAR-001	Characterization Testing Program	7
12-EHP-5043-EDC-001	Evaluation of Discrepant Conditions	16
12-IHP-5021-EMP-080	Eaton/Cutler-Hammer 4kV Circuit Breaker Maintenance	13
12-IHP-5030-EMP-006	MCCB/TOLR Testing and Electrical Enclosure Maintenance	029
12-MHP-5021-056-001	Motor Driven and Turbine Driven Auxiliary Feed Pump Maintenance	12

PROCEDURES

Number	Description or Title	Revision
12-OHP-4021-019-001	Operation of the Essential Service Water System	49
12-OHP-4022-082-004	Degraded Offsite AC Voltage Response	006
12-OHP-4023-E-0	Reactor Trip or Safety Injection	29
12-OHP-4023-E-3	Steam Generator Tube Rupture	12
12-OHP-4023-ECA-3.1	SGTR With Loss of Reactor Coolant-Subcooled Recovery Desired	9
12-OHP-4023-ES-3.1	Post-SGTR Cooldown Using Backfill	4
12-OHP-4023-ES-3.2	Post-SGTR Cooldown Using Blowdown	6
1-OHP-4022-001	Earthquake	15
1-OHP-4022-002-021	Steam Generator Tube Leak	13
1-OHP-4022-055-003	Loss of Condensate to Auxiliary Feedwater Pumps	9
1-OHP-4022-056-001	Steam Binding in Auxiliary Feedwater System	5
1-OHP-4022-064-001	Control Air Malfunction	8
1-OHP-4022-064-002	Loss of Control Air Recovery	12
1-OHP-4023-E-0	Reactor Trip or Safety Injection	33
1-OHP-4023-E-3	Steam Generator Tube Rupture	15
1-OHP-4023-ECA-3.1	SGTR With Loss of Reactor Coolant-Subcooled Recovery Desired	12
1-OHP-4023-ES-0.1	Reactor Trip Response	26
1-OHP-4023-ES-3.1	Post-SGTR Cooldown Using Backfill	10
1-OHP-4023-ES-3.2	Post-SGTR Cooldown Using Blowdown	10
1-OHP-4023-FR-H.1	Response to Loss of Secondary Heat Sink	21
1-OHP-4024-108	Annunciator No. 108 Response: Pressurizer	20
1-OHP-4025-LS-3	Steam Generator 2/3 Level Control	3
1-OHP-4025-LS-4	Steam Generator 1/4 Level Control	3
1-OHP-4025-R-12	Component Restoration	5
1-OHP-4025-R-7	Restore Control Air	0a
12-IHP-4030-082-006	AB, CD, and N-Train Battery Yearly Surveillance and Maintenance	8
1-OHP-4030-114-021	Inoperable Power Supply	19
1-OHP-4030-114-031	Breaker Alignment	20

PROCEDURES

Number	Description or Title	Revision
1-OHP-4030-114-031	Operations Weekly Surveillance Checks	19
1-OHP-4030-119-022FV	ESW Flow Verification	16
1-OHP-4030-132-027CD	CD Diesel Generator Operability Test (Train A)	20
1-OHP-4030-156-017E	East Motor Drive Auxiliary Feedwater System Test	7
1-OHP-4030-156-017T	Turbine Driven Auxiliary Feedwater System Test	10
1-OHP-5030-119-003	ESW to AFW Pump Cleanout Connection Flush	4
12-IHP-5030-EMP-013	Electrical Enclosure 10 Year Preventive Maintenance	22
12-IHP-6030-RLY-022	General Electric Type HFA51 and HFA151 Multi-Contact Auxiliary Relay Adjustment and Maintenance	14
2-OHP-4023-E-0	Reactor Trip or Safety Injection	37
2-OHP-4023-E-3	Steam Generator Tube Rupture	16
2-OHP-4023-ECA-3.1	SGTR With Loss of Reactor Coolant-Subcooled Recovery Desired	11
2-OHP-4023-ES-3.1	Post-SGTR Cooldown Using Backfill	9
2-OHP-4023-ES-3.2	Post-SGTR Cooldown Using Blowdown	9
AE-O-E232	Local Operation of U1 Steam Generator PORV	8
OHI-4023	Abnormal/Emergency Procedure User's Guide	31
PMI-2010	Written Instructions	36
PMI-4075	Operator Time Critical Actions	1
PMP-2010-PRC-002	Procedure Alteration, Review, and Approval	38
PMP-2291-OLR-001	On-Line Risk Management	22
PMP-4075-TCA-001	Time Critical Action Validation and Verification	1 & 2
PMP-6065-EIC-001	Electrical and Instrumentation and Controls Engineering Information Control Process	1
PMP-7030-SFD-001	Safety Function Determination Program	2

WORK ORDERS

Number	Description or Title	Date
1-OHP-4030-156-017T	Turbine Driven AFW Pump System Test	9/20/11, 06/21/12
01-OHL-4030-102-60	Pressurizer PORV Testing	10/16/11

WORK ORDERS

Number	Description or Title	Date
01-OHL-4030-114-034	Pressurizer PORV Position Indication Test	10/17/11
JO 3135065	1-WMO-733, Replace Valve	4/11/05
55225736 01	I-WMO-733-ACT, As-Found/As-Left Diagnosis	12/22/10
55231048 01	1-T11D, Inspect/Clean Switchgear Bus	10/7/06
55237882 03	Clean/Inspect/Test BKR 1-T11D11	7/8/09
55240935 01	1-87X-t11A Trip Test Lockout Relay	3/12/10
55244511 01	1-PP-3E-MTR, Lube Bearings and Clean	7/8/09
55245339 01	Clean/Inspect/Test BKR 1-T11D10	2/11/09
55245436 01	I-OME-34E, Disassemble /Inspect Strainer	12/16/08
55265288	1-S V-1A-1: Perform Setpoint Verification	9/19/11
55270942 01	Remove/Refurbish/Replace 1-PP-7E-MTR	3/5/08
55292664 03	MTM, I-FW-124, Remove/Replace Check Valve	1/31/11
55358182 01	Flush and Replace Oil in the Upper Bearing Reservoir on 1-PP-7E-MTR	2/9/11
55358183 01	Clean The Air Screens on 1-PP-7E-MTR, "East Essential Service Water Pump PP7E Motor"	2/9/11
55366415 01	Perform Infrared Inspection on 1-PP-3E-MTR	7/7/11
55366416 01	Perform Characterization Testing on 1-PP-3E-MTR (East Motor Driven Auxiliary Feedwater Pump PP-3E Motor) in Accordance With 12-EHP-5030-CAR-001	7/7/11
55366416 02	Clean Motor Screens on I-PP-3E-MTR, (East Motor Driven Auxiliary Feedwater Pump PP-3E MOTOR)	7/7/11
55366416 03	Support Characterization Testing on 1-PP-3E-MTR (East Motor Driven Auxiliary Feedwater Pump PP-3E Motor), at Breaker Cubicle I-TIIDT1.	7/7/11
55366421 04	Sample/ Drain /Refill Pump Brg and Clean 1-PP-3E-MTR	7/7/11
55378968 01	Lube Lower Bearing: (1-PP-7E-MTR)	2/8/12
55378969 01	Obtain an Oil Sample From 1-PP-7E-MTR, "East Essential Service Water Pump PP-7E MOTOR" for Analysis	2/9/12
55379811-01	1-FW-135; Perform Non-Intrusive Turbine Driven AFW Pump PMTM	8/31/2011
55379811	1-FW-135; Perform Non-intrusive Test	8/31/11
55380536 01	1-PP-7E-MTR; Perform Motor Characterization Test	2/7/12
55386933 01	Perform Surveillance Procedure 01-IHP-6030-IMP-309 (4KV Bus Loss of Voltage and 4kV Bus Degraded Voltage Relay Calibration).	1/6/12
55397158 01	Perform Surveillance Procedure 01-IHP-6030-IMP-309 (4KV Bus Loss of Voltage and 4kV Bus Degraded Voltage Relay Calibration).	7/6/12
55407760 01	1-MRV-222, No. 2 SG Stop Valve Dump Valve	7/21/12
PM 00109769	1-87X-t11A Trip Test Lockout Relay	1
PM 00110736	U1 CD EDG, Replace Relays	1
PM 00114641	1-5A6-LCTA, Replace HFA Relay	1

LIST OF ACRONYMS USED

AFW	Auxiliary Feedwater
ASME	American Society of Mechanical Engineers
CAC	Control Air Compressor
CAS	Compressed Air System
CCW	Closed Cooling Water
CDBI	Component Design Basis Inspection
CFR	Code of Federal Regulations
CR	Condition Report
CST	Condensate Storage Tank
DBA	Design Basis Accident
DRS	Division of Reactor Safety
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
ESW	Emergency Service Water
GL	Generic Letter
HELB	High Energy Line Break
IEEE	Institute of Electrical and Electronics Engineers
IMC	Inspection Manual Chapter
IN	Information Notice
LTC	Load Tap Changer
MDAFW	Motor Driven Auxiliary Feedwater
MOV	Motor Operated Valve
MTO	Margin to Overfill
NCV	Non-Cited Violation
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
PARS	Publicly Available Records System
PORV	Power Operated Relief Valve
PRA	Probabilistic Risk Assessment
RCS	Reactor Coolant System
RIS	Regulatory Issue Summary
RG	Regulatory Guide
SBO	Station Blackout
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SR	Safety-related
SSC	Structures, Systems or Components
TDAFW	Turbine Driven Auxiliary Feedwater
TIA	Task Interface Agreement
TOL	Thermal Overload
TS	Technical Specification
TSO	Transmission System Operator
UFSAR	Updated Final Safety Evaluation Report
URI	Unresolved Item

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Sincerely,

/RA/

Ann Marie Stone, Chief
Engineering Branch 2
Division of Reactor Safety

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