UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
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November 30, 2010

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

SUBJECT: DONALD C. COOK NUCLEAR POWER PLANT, UNITS 1 AND 2
COMPONENT DESIGN BASES INSPECTION (CDBI) INSPECTION REPORT
05000315/2010006; 05000316/2010006(DRS)

Dear Mr. Weber:

On October 21, 2010, the U.S. Nuclear Regulatory Commission (NRC) completed a component design bases inspection at your Donald C. Cook Nuclear Power Plant, Units 1 and 2. The enclosed report documents the results of this inspection, which was discussed on September 17, 2010, with you, and on October 21, 2010, 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 the issue as 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 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 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 05000315/2010006; 05000316/2010006(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:	05000315/2010006; 05000316/2010006(DRS)
Licensee:	Indiana Michigan Power Company
Facility:	Donald C. Cook Nuclear Power Plant, Units 1 and 2
Location:	Bridgman, MI
Dates:	February 8 – 12, 2010 (Heat Sink) August 16 through October 21, 2010 (CDBI)
Inspectors:	A. Dunlop, Senior Reactor Engineer, Lead G. Hausman, Senior Reactor Engineer, Electrical R. Langstaff, Senior Reactor Engineer, Operations J. Corujo-Sandin, Reactor Engineer, Mechanical O. Mazzoni, Electrical Contractor B. Sherbin, Mechanical Contractor G. O'Dwyer, Reactor Engineer (Heat Sink) L. Jones, Reactor Engineer (Heat Sink)
Observer:	N. Melly, Office of Nuclear Regulatory Research
Approved by:	Ann Marie Stone, Chief Engineering Branch 2 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000315/2010006(DRS)/05000316/2010006(DRS); 08/16/2010 – 10/21/2010; Donald C. Cook Nuclear Power Plant; Component Design Bases Inspection (CDBI).

The inspection was a 3-week onsite baseline inspection that focused on the design of components that are risk-significant and have low design margin. In addition, the inspection included a 1-week onsite baseline inspection of the Heat Sink Performance. 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 most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. 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 having very low safety significance and an NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's failure to establish inspection procedures that were appropriate for the circumstances. Specifically, the licensee's heat exchanger inspection guidance and acceptance criteria could potentially result in the design basis tube plugging limit being exceeded due to the accumulation of macro fouling and as a result the heat exchanger would not be able to meet the design basis heat removal capability. This finding was entered into the licensee's corrective action program and a review of the heat exchanger tube plugging analysis identified additional margin to remain within its design basis heat removal capability.

  The finding was more than minor because if left uncorrected, it would have the potential to lead to a more significant safety concern. This finding was of very low safety significance (Green) because the licensee was able to demonstrate adequate margin and therefore there was not a design or qualification deficiency, did not represent a loss of system safety function, and did not screen as potentially risk-significant due to a seismic, flooding, or severe weather initiating event. The finding had a cross-cutting aspect in the area of human performance because the licensee did not use conservative assumptions in decision making when developing the inspection guidance and acceptance criteria. [H.1(b)] (Section 1R21.3.b.(1))

REPORT DETAILS

1. **REACTOR SAFETY**

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

1R07 Heat Sink Performance (71111.07T)

.1 Triennial Review of Heat Sink Performance

a. Inspection Scope

The inspectors reviewed completed surveillances, vendor manual information, associated calculations, performance test results, and cooler inspection results associated with the component cooling water (CCW) heat exchanger (2-HE-15E) and the emergency diesel generator (EDG) CD lube oil cooler (2-QT-110-CD). These heat exchangers were chosen based on their risk-significance in the licensee's probabilistic safety analysis, their important safety-related mitigating system support functions, their operating history, and relatively low margin.

For the CCW heat exchanger and the EDG lube oil cooler, the inspectors reviewed the methods and results of heat exchanger performance inspections. The inspectors reviewed the methods and established acceptance criteria used to inspect and clean heat exchangers to verify consistency with industry standards and the as-found results were recorded, evaluated, and appropriately dispositioned such that the as-left condition was acceptable.

In addition, the inspectors reviewed the condition and operation of the CCW heat exchanger and the EDG lube oil cooler to verify consistency with design assumptions in heat transfer calculations and as described in the Updated Final Safety Analysis Report (UFSAR). This included verification that the number of plugged tubes was within pre-established limits based on capacity and heat transfer assumptions. The inspectors reviewed the licensee's evaluation for a potential water hammer to ensure appropriate measures were in place to address it. The controls and operational limits were reviewed to verify they were adequate to prevent heat exchanger degradation due to excessive flow induced vibration during operation. In addition, eddy current test reports and visual inspection records were reviewed to verify the structural integrity of the heat exchanger.

The inspectors reviewed the performance of the ultimate heat sink (UHS), the essential service water (ESW) system to ensure availability and accessibility to the inplant cooling water system. The focus was on the attributes listed in IP 71111-07, Section 02.02.d.2. This included reviewing the inspection results of the screenhouse's safety-related and non-safety-related intake bays to verify that identified settlement or movement indicating loss of structural integrity and/or capacity was appropriately evaluated and dispositioned by the licensee. In addition, the review was to ensure that reservoir capacity was sufficient by removing debris or sediment buildup in the screenhouse's intake forebay and inspection results were being adequately trended. The inspectors performed a walkdown of accessible portions of the screenhouse. The inspectors also reviewed the programs for biotic control and macro fouling to verify they were implemented as

required and the results were monitored, trended, and evaluated to prevent clogging of heat exchangers and other components within the ESW system.

In addition, the inspectors reviewed corrective action program documents related to the heat exchangers/coolers and heat sink performance issues to verify that the licensee had an appropriate threshold for identifying issues and to evaluate the effectiveness of the corrective actions. The documents that were reviewed are included in the Attachment to this report.

These inspection activities associated with the CCW heat exchanger and the EDG lube oil cooler constituted two heat sink inspection samples as defined in IP 71111.07-05.

b. Findings

No findings of significance were identified; however, a finding was identified during the component design bases inspection associated with the CCW heat exchanger as discussed in Section 1R21.3.b(1) of this report.

1R21 Component Design Bases Inspection (71111.21)

.1 Introduction

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

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

.2 Inspection Sample Selection Process

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

The inspectors performed a margin assessment and detailed review of the selected risk-significant components to verify that the design bases have been correctly implemented and maintained. This design margin assessment considered original design reductions

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

This inspection constituted 25 samples as defined in Inspection Procedure 71111.21-05, which included 18 components, 5 operating experience reviews and 2 operator actions.

.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 NRC 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 that the selected components would function as designed when required and support proper operation of the associated systems. The attributes that were needed for a component to perform its required function included process medium, energy sources, control systems, operator actions, and heat removal. The attributes to verify that the component condition and tested capability was consistent with the design bases and was appropriate may include installed configuration, system operation, detailed design, system testing, equipment and environmental qualification, equipment protection, component inputs and outputs, operating experience, and component degradation.

For each of the components selected, the inspectors reviewed the maintenance history, 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 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 18 components were reviewed:

- Unit Auxiliary Transformer (TR2CD): The inspectors reviewed applicable calculations, system one-line diagrams, and loading requirements to determine the adequacy of the transformer to supply the 4.16kV Train A power demand requirements. The inspectors reviewed the licensee's engineering guide for selection of surge arrester protection for the transformer and upstream

equipment in order to evaluate the surge protection for downstream system. The protective relay setpoint calculations were reviewed to assess the adequacy of electrical protection and that trip setpoints would not unduly interfere with the transformer performing its design function during transients occurring upon transformer energization and through-faults, and at transformer maximum loading conditions. The inspectors reviewed the dynamic calculations for auto transfer scheme to ensure that no undue transient torque issues were introduced during transfer of loads, as well as the breaker control diagrams for the associated breakers.

- Auxiliary 4.16kV Engineered Safety System Bus (T-21D): The inspectors reviewed the one-line diagram, load flow calculations, short circuit calculations, and protective relay trip setpoints, to evaluate the adequacy of the switchgear's voltage, current, and interrupting ratings as well the adequacy of electrical protection coordination with upstream and downstream breakers. The inspectors reviewed calculations and engineering guides to evaluate the surge protection afforded for the bus and connected equipment when supplied by the reserve auxiliary transformer, via the 34.5kV switchyard, and transformer TR4 or TR5, as well as the surge arrester protection when the bus was supplied by the 69kV source. The inspectors also reviewed the licensee's evaluation of switching surges to verify adequacy of the recently installed vacuum circuit breakers. The recently installed back fit modification for the degraded voltage relay settings was reviewed to ensure that adequate voltage was maintained at the terminals of the safety loads under the different available plant power sources. The backfit was required by NRC letter, "Donald C. Cook Nuclear Power Plant, Units 1 and 2 - Imposition of Facility-Specific Backfit Re: Degraded Voltage Protection System, dated November 9, 2005, (ML050680057). The inspectors reviewed the capability and availability of the 69kV source to supply bus T-21D, and the modification for the recently installed voltage regulator, to ensure that the 69kV source would deliver adequate voltage to loads connected under loss-of-coolant-accident (LOCA) conditions. The bus tie breakers closing and opening control circuits were reviewed to verify that breaker tripping and closing logic was consistent with design basis description and interlocking requirements.

- Engineered Safety System 600V Bus (21D): The inspectors reviewed the calculations for short circuit, loading, and voltage regulation to assess that adequate voltage was available at the bus under all postulated plant operating conditions and that the bus interrupting equipment was properly rated for the imposed service conditions. The control drawings for the normal supply breaker 21D1 from transformer TR21D and for the alternate supply breaker 21BD tie to Train B were reviewed to verify that breaker tripping and closing logic was consistent with its design basis and interlocking requirements.

- Emergency Diesel Generator (EDG) (2-CD): The inspectors reviewed the EDG dynamic loading calculation to ensure the adequacy of voltage, frequency, current, and loading sequence during the loss of offsite power and LOCA scenarios. Short circuit calculations were reviewed to ensure that the ratings of the connected equipment were adequate for the calculated available short circuit duties. The electrical drawings and calculations applicable to the generator

output breaker T21D8 control logic and interlocks were reviewed to determine whether the breaker opening and closing control circuits were consistent with design basis documents. Protective relay setpoint calculations were reviewed to assess adequacy of protection during test mode and during emergency operation. The generator grounding scheme was also reviewed to verify the adequacy of the grounding scheme and ground over-current relay coordination. The inspectors performed a review of the EDG system field conditions, including the generator nameplate, to assess the adequacy of the ratings utilized in the calculations. The inspectors reviewed the licensee response to IN 2007-36, "Emergency Diesel Generator Voltage Regulator Problems," to assess the licensee's evaluation with respect to the plant's components.

- Supplemental Diesel Generators (SDGs) (12-OME-250-SDG1/SDG2): The inspectors reviewed the modification package for the addition of two new SDGs to the emergency power system to allow for the extension of the allowed outage time for the safety-related EDGs. Calculations addressing fuel consumption and tank volume for 24-hour operation were also reviewed to verify adequate onsite fuel inventory. The inspectors performed a review of the SDGs normal operating procedures and test procedures to assess whether component operation and alignments were consistent with design and licensing bases assumptions. In addition, the inspectors reviewed the SDGs' operational testing and emergency operation (i.e., sequence of loads) during a loss of 69kV offsite power.

- 250Vdc Transfer Cabinet (2-TDCD): The inspectors reviewed 250Vdc elementary and schematic diagrams, voltage drop and coordination calculations, and fuse ratings to confirm that sufficient coordination existed between various interrupting devices. In addition, the inspectors verified that sufficient power and voltage was available to safety-related direct current equipment to perform their safety function.

- 250Vdc Plant Battery and Busses (2-BATT-CD): The inspectors reviewed 250Vdc battery and charger sizing calculations, TS surveillance requirements, and completed surveillances to confirm that sufficient capacity exists for the battery as well as the charger to perform their safety function and were being adequately maintained. The ventilation calculations were reviewed to verify that the temperature rise in the battery and charger rooms during station blackout and post-LOCA conditions would not adversely affect the performance of the battery and its charger. In addition, the inspectors reviewed the battery room's hydrogen concentration calculation and mitigation procedures to verify that the battery room's hydrogen concentration would be maintained below 2 percent and that if ventilation was ever lost, there would be adequate time to respond and take compensatory actions (i.e., install temporary ventilation) to preclude reaching the 2 percent concentration level.

- Control Room Instrument Distribution-II (CRID-II) 7.5kVA Static Inverter: The inspectors reviewed 120Vac elementary and schematic diagrams, voltage drop and coordination calculations to confirm that sufficient coordination existed between various interrupting devices. In addition, the inspectors verified that

sufficient power and voltage was available to safety-related alternating current equipment to perform their safety function.

- Residual Heat Removal (RHR) Pump (2-PP-35E): The inspectors reviewed design basis documents, including hydraulic calculations, accident analyses, and drawings to verify that the RHR pump was capable of meeting system functional and design basis requirements. The inspectors reviewed operating and emergency operating procedures and refueling water storage tank (RWST) level setpoints to verify pump suction swap-over occurred before the onset of vortexing at the RWST outlet piping. The inspectors also reviewed RHR pump inservice tests (IST) results to verify acceptance criteria were met and performance degradation could be identified. The inspectors reviewed the licensee's Bulletin 88-04, "Potential Safety-Related Pump Loss," evaluation for pump minimum flow and pump to pump interaction. The inspectors also reviewed RHR room ventilation and flooding analyses to ensure environmental conditions were maintained in the pump room.

- Containment Recirculation Sump Main Strainer (2-STN-320): The inspectors reviewed the strainer design basis function to provide adequate suction flow to the emergency core cooling system (ECCS) pumps and to stop foreign particles larger than its mesh openings to prevent pump and piping system damage. The inspectors reviewed calculations including net positive suction head, debris generation and transport, pressure drop across the strainer, structural loading, hydrodynamic loading, and vortexing to verify that the strainer would perform as required. Recent inspection reports were reviewed to ensure physical condition of the strainer was being maintained.

- Pressurizer Power Operated Relief Valve (PORV) (2-NRV151/152): The inspectors reviewed design calculations that were performed to determine the lift settings and required stroke timing of the PORVs while in the Low-Temperature Overpressure Protection (LTOP) mode of operation. This included reviewing the licensee's response to IN 1993-58, "Nonconservatism in Low-Temperature Overpressure Protection for Pressurized-Water Reactors." The inspectors reviewed instrument setpoint and uncertainty calculations for pressure instruments that would be relied upon to open the PORV when required. The inspectors also reviewed the PORV air-operated valve actuator capability calculation, and backup air bottle sizing calculations and air bottle testing to ensure the valve would perform the required number of strokes for LTOP operation. The inspectors also reviewed analysis assumptions of PORV stroke time and reactor coolant system (RCS) pressurization rate due to the operation of safety injection pumps and reactor coolant pumps to ensure they were accounted for in plant operating and surveillance procedures. In addition, the inspectors reviewed 250Vdc elementary and schematic diagrams, solenoid vendor specification data, solenoid load voltage drop, and environmental qualification requirements to confirm that the pressurizer PORVs' solenoid valves would perform their safety function.

- Recirculation Sump to RHR/Containment Spray Pumps Suction Containment Isolation Valve (2-ICM-305): The inspectors reviewed motor-operated valve (MOV) calculations and analysis to ensure the valve was capable of functioning under design conditions. These included calculations for required thrust, maximum differential pressure, pressure locking, degraded voltage, and valve weak link analysis. Diagnostic testing and IST surveillance results, including stroke time and available thrust, were reviewed to verify acceptance criteria were met and performance degradation could be identified. The inspectors reviewed the valve's bonnet relief valve to ensure it was properly set and tested to overcome a pressure locking scenario.

- East RHR Heat Exchanger to Charging Pumps Suction Shutoff Valve (2-IMO-340): The inspectors reviewed MOV calculations and analysis to ensure the valve was capable of functioning under design conditions. These included calculations for required thrust, maximum differential pressure, degraded voltage, and valve weak link analysis. Diagnostic testing and IST surveillance results, including stroke time and available thrust, were reviewed to verify acceptance criteria were met and performance degradation could be identified.

- RCS Loop No. 2 Hot Leg to RHR Pumps Suction Containment Isolation Valve (2-IMO-128): The inspectors reviewed MOV calculations and analysis to ensure the valve was capable of functioning under design conditions. These included calculations for required thrust, maximum differential pressure, degraded voltage, and valve weak link analysis. Diagnostic testing and IST surveillance results, including stroke time and available thrust, were reviewed to verify acceptance criteria were met and performance degradation could be identified.

- East Component Cooling Water (CCW) Pump (2-PP-10E): The inspectors reviewed the CCW pump design to verify its capability to meet design basis flow and pressure requirements. The inspectors reviewed procedures, pump curves, and IST analyses to verify assumptions were accurate and justified. The inspectors reviewed the equipment's seismic classification and environmental requirements to ensure the pump would function under design basis conditions. Analysis and test results were reviewed to verify pump performance and capability to provide the required flow rates was acceptable without going into run-out conditions. The inspectors reviewed the licensee's evaluation of NRC Bulletin 88-04 for pump to pump interaction.

- East CCW Heat Exchanger (2-HE-15E): The inspectors reviewed the operating and emergency procedures, heat transfer calculations, and flow requirements associated with the CCW heat exchanger for conformance with design basis requirements. The inspectors reviewed the heat exchanger's capability to remove design basis heat loads based on the allowed tube plugging limit. The licensee's process for inspecting the as-found condition of the heat exchanger's internals was reviewed to verify acceptable limits had been established to ensure the heat exchanger heat removal capability remained within its design limits. The inspectors reviewed the two most recent inspect and clean activities, and eddy current testing results performed on the heat exchanger as part of the GL 89-13, "Service Water System Problems Affecting Safety-Related Equipment," program.

- EDG CD Lube Oil Cooler (2-QT-110-CD): The inspectors reviewed the operating and emergency procedures, heat transfer calculations, and flow requirements associated with the EDG lube oil cooler for conformance with design basis requirements. The inspectors reviewed the most recent inspect and clean activities performed on the heat exchanger as part of the GL 89-13 program. The inspectors reviewed the heat exchangers' capability to remove design basis heat loads based on the allowed tube plugging limit. As part of the inspection, the inspectors reviewed IN 1997-41, "Potentially Undersized Emergency Diesel Generator (EDG) Oil Coolers" to determine any applicable aspects to the lube oil cooler.

- East ESW Pump Discharge Strainer Outlet Check Valve (2-ESW-102E): The inspectors reviewed the maintenance records associated with the check valve, including the last disassemble and inspection to verify the valve was being adequately maintained. The inspectors reviewed the IST to ensure the valve was capable to perform its design functions.

b. Findings

(1) Inadequate As-Found Heat Exchanger Inspection Guidance and Acceptance Criteria.

Introduction: A finding of very low safety significance (Green) and associated Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified by the inspectors for the failure to establish inspection procedures that were appropriate for the circumstances for the GL 89-13 program heat exchangers. Specifically, the licensee's heat exchanger inspection guidance and acceptance criteria could potentially result in the design basis tube plugging limit being exceeded due to the accumulation of macro fouling and as a result the heat exchanger would not meet its design basis heat removal capability.

Description: The CCW system is a safety-related closed loop cooling system which serves as an intermediate system between potentially radioactive heat sources and the ultimate heat sink. The CCW system removes heat from the reactor coolant system (RCS), in addition to other loads, and transfers the heat to the ESW system. The CCW heat exchangers are only one of various parallel loads (i.e., other safety-related heat exchangers) supplied by the ESW system. There is one CCW heat exchanger per train and there are two independent CCW trains per reactor unit. The CCW heat exchanger is a two pass flow heat exchanger, with ESW (lake water) flowing though the tube side and CCW water flowing though the shell side.

While reviewing documentation, including pictures from the last inspect and clean of the 2-HE-15E CCW heat exchanger, the inspectors noted that large segments of the tubesheet were covered by macro fouling. The macro fouling was composed mainly of blue-green algae, silt, and sand. The algae, although not growing in the system, was small enough to bypass the ESW strainer and accumulate, primarily on the second pass of the heat exchanger's tubesheet. Once a sufficient amount of algae attached itself to the tubesheet, the sand and silt would be trapped developing a mat like substance that covered a number of tubes, which then would prevent or limit flow through the affected tubes. The worst as-found conditions, from the pictures provided, showed approximately

¾ of the tubesheet entrance to the second pass had its tubes either partially or fully plugged by the algae and silt accumulation. The inspectors reviewed pictures from previous inspect and clean activities for the 2-HE-15E CCW heat exchanger and pictures from the other three CCW heat exchangers, and noted that similar conditions had been observed on the other heat exchangers as well.

The discovery of macro fouling during the inspection and cleaning activities conducted in 2009 was not considered to be outside of the analyzed design limit by the licensee. Specifically, 12-EHP-8913-001-002, "Heat Exchanger Inspection," required the development of visual inspection acceptance criteria prior to each inspection. This procedure included a note that stated "100 percent flow blockage is required to remove a tube from service." The estimation of partial versus full-tube fouling was based on the qualitative judgment of the licensee personnel performing the inspection, not on any industry or plant guidance. The inspectors noted the licensee established an acceptance criteria in 2009 consistent with the guidance of 12-EHP-8913-001-002 and confirmed through interviews with the licensee that partial blockage of tubes was viewed as acceptable because the tube surface area remained active as long as flow existed. In addition, the licensee believed that the tubes that where completely covered with the algae and silt accumulation would have been flushed cleaned by a design basis flow had the system been called upon to mitigate an accident. As such, no tubes were considered fully plugged by the licensee during their inspection of the heat exchanger.

The applicable design document, mechanical design standard MDS-607, "Heat Exchanger Tube Plugging," indicated the 2-HE-15E CCW heat exchanger could accommodate a total of 2 percent of its tubes to be plugged, 45 out of 2260, and still meet its design basis heat removal capability. The heat exchanger already had 39 mechanically plugged tubes when it was inspected and cleaned during the spring refueling outage of 2009. After cleaning, inspecting, and eddy-current testing the heat exchanger, the licensee increased the number of mechanically plugged tubes to 44. As a result, since April 2009 the heat exchanger had only a 1 tube margin based on the licensee's plugging analysis.

In addition, during inspect and clean activities, the licensee used a light test to identify apparently blocked tubes and a compressed air test to determine if tubes that failed the light test where actually blocked. However, further discussions with the licensee revealed that the algae and silt accumulation on the tubesheet was removed prior to performing the light and compressed air tests. Therefore, while these tests supported the licensee's identification of any blockage further down the length of the tubes, it could not verify flow though the algae and silt accumulation that blocked the tubes.

The as-found inspection guidance and acceptance criteria used by the licensee for identifying tubes potentially fully blocked by macro fouling were considered inadequate. The acceptance criteria also did not assess the potential impact that partially blocked tubes could have on the heat transfer capabilities of the heat exchanger. The inspectors were concerned that a heavily fouled CCW heat exchanger would result in higher flow resistance. Therefore, the design basis flow required for mitigating an accident could potentially not be met, because water would flow though the path of least resistance (i.e., the other parallel heat exchangers). The licensee only performs as-left flow verification of the ESW system such that it was unclear whether there would be sufficient

flow through a heavily fouled CCW heat exchanger under accident conditions. These deficiencies with the inspection guidance and acceptance criteria could prevent an adequate evaluation of the operability of the CCW system.

The inspectors were concerned because the as-found conditions of the CCW heat exchanger did not appear to be within the established tube plugging design limits. If the heat removal capability of the heat exchanger cannot be verified, the frequency of inspect and clean would need to be reevaluated. Currently the licensee scheduled inspection and cleaning of the CCW heat exchangers during every refueling outage.

The licensee initiated Action Request (AR) 2010-8974 to address the inspectors' concerns. A review of the heat exchanger tube plugging analysis identified that the heat exchanger could have up to 3 percent, 68 out of the 2260, of its tubes plugged and still remain within its design basis heat removal capability. Although administratively the CCW heat exchanger only had 1 tube margin, analytically it had an additional 23 tubes of margin.

The licensee's evaluation in AR 2010-8974 also stated that before restarting from a Unit 1 forced outage, the licensee performed flow verification on both trains of Unit 1 ESW system. At the time the forced outage occurred, Unit 1 had been operating for almost a year since the last inspect and clean on the CCW heat exchangers. Both tests were within specifications with only a minor (2 to 3 gpm) adjustment being made to one train. The licensee stated these as-found tests provided confidence macro fouling would not substantially affect the system flow balance. However, when reviewing the CCW heat exchangers' as-found inspection results performed subsequent to the flow verification, the inspectors noted the fouling conditions were significantly less, compared to the results from other CCW heat exchangers. Therefore, while the flow verification could be considered an example of as-found for the Unit 1 CCW heat exchangers, it is not a representative sample for comparison with all CCW heat exchanger, since macro fouling appears to be highly dependent on lake conditions.

As part of their corrective actions the licensee intends to conduct a self-assessment of their GL 89-13 program and perform an in-depth apparent cause evaluation to assess this issue. Based on the information obtained and considering the additional margin determined by calculation, the inspectors had no further operability issues with the heat exchanger.

Analysis: The inspectors determined that the failure to establish inspection procedures with adequate inspection guidance and acceptance criteria for the GL 89-13 program heat exchangers was a performance deficiency. The performance deficiency was determined to be more than minor because if left uncorrected, it has the potential to lead to a more significant safety concern. Specifically, the licensee could potentially exceed their allowed design basis tube plugging limit due to the accumulation of macro fouling (i.e., blue-green algae and silt) and as a result, the heat exchanger would not be able to meet the design basis heat removal capability.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Table 4a for the Mitigating System

cornerstone. The finding screened as very low safety significance (Green) because the finding was not a design or qualification deficiency, did not represent a loss of system safety function, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. In addition, the licensee provided an analysis that showed that the CCW heat exchanger had higher analytical design margin for tube plugging than the one tube administrative margin currently in place.

The inspectors determined that the finding had a cross-cutting aspect in the area of human performance because the licensee did not use conservative assumptions in decision making. Specifically, the licensee failed to establish appropriate procedures because the licensee made non-conservative assumptions that tubes fully and partially blocked by algae and silt were acceptable without further evaluation when establishing the acceptance criteria for the heat exchanger inspection in 2009. [H.1(b)]

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

Contrary to the above, as of September 17, 2010, the procedures used for the inspection of GL 89-13 program heat exchangers were not appropriate to the circumstances. Specifically, procedure 12-EHP-8913-001-002 required the licensee to remove a heat exchanger tube from service only when 100 percent flow blockage was identified; however, the procedure did not contain appropriate acceptance criteria or guidance for identifying a fully block tube. Additionally, there was no adequate guidance for evaluating the effects partially blocked tubes could have on the heat transfer capability of the heat exchangers. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program as AR2010-8974, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000315/2010006-01; 05000316/2010006-01, Inadequate As-Found Heat Exchanger Inspection Guidance and Acceptance Criteria).

(2) Non Conservative Analysis Used to Determine LTOP Setpoint for the PORV

Introduction: The inspectors identified an unresolved item (URI) concerning a non-conservative analysis. Specifically, the design calculation performed to determine the pressurizer PORV lift setting while in the LTOP mode of operation failed to consider the instrument uncertainty associated with the pressure instrumentation that actuates the PORV to open when required.

Description: In order to limit RCS pressure at low temperatures, LTOP is established so the integrity of the RCS pressure boundary would not be compromised by violating the pressure and temperature (P-T) limits of 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements." The reactor vessel is the limiting RCS pressure boundary component for demonstrating such protection, and is less tough at low temperatures than at normal operating temperature. The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only while shutdown, because a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P-T limits by a significant amount could cause brittle

cracking of the reactor vessel. While in the LTOP mode of operation, vessel protection for overpressure is provided by the PORVs, and, in some cases, in combination with RHR system relief valves. Based on the current Appendix G methodology, the P-T limit was determined to be 621 psig.

The LTOP PORV setpoint of 435 psig was determined by Westinghouse in 1989 based on the 10 CFR Part 50, Appendix G limit curves, and LTOP setpoint methodology at the time. Although there were several updates to the P-T limits since 1989, Westinghouse determined there was sufficient margin between the maximum allowable LTOP setpoint and the current PORV setpoint of 435 psig, such that no setpoint change was warranted.

Information Notice 93-58, "Nonconservatism in Low-Temperature Overpressure Protection for Pressurized-Water Reactors," documented LTOP setpoint errors identified by Westinghouse. These errors resulted in a pressure increase of 88.7 psi to the LTOP setpoint for D. C. Cook. The inspectors reviewed ECP-12-N1-05, "LTOP Setpoint Calculation," which documented the LTOP setpoint basis and reconciled the errors mentioned in the IN. This calculation had previously determined the contribution from overshoot (78 psi), the pressure the reactor vessel would be subjected to above the setpoint based on the relief rate of the PORV's size and stroke time. Considering the adjustments for the errors and the contribution from overshoot, the margin to the Appendix G limits was about 19.3 psig (621 psig - 435 psig - 88.7 psig - 78 psig).

However, the inspectors determined that the licensee did not account for instrument uncertainty in the PORV lift setting of 435 psig. The IN recommended method to compensate for the pressure increase was to demonstrate that the available margin in the LTOP calculation, taking into account instrumentation uncertainty, was sufficient to offset the plant-specific pressure differences. The Westinghouse letter, AEP-93-208, dated March 10, 1993, contained a similar recommended action to offset the pressure increase by using the instrument uncertainty used in the development of the P-T curves. However, the D. C. Cook P-T curves did not include instrument uncertainty in their development such that it was not available to offset the plant-specific pressure differences. Based on the licensee's interpretation of the Westinghouse letter, they did not believe instrument uncertainty needed to be accounted for in the setpoint calculation, especially since they believed it was not within their license basis.

In Calculation 1-2-UNC-211 Calc 2, "RCS Wide Range Pressure," the licensee had determined there was about 75 psi pressure uncertainty associated with the PORV lift setting, which had not been incorporated in the basis for the current setpoint of 435 psig. After further evaluation during this inspection, the licensee concluded the pressure uncertainty associated with the PORV lift setting was about 68.6 psi. The inspectors reviewed the calculation, and noted there was no seismic contribution to instrument uncertainty in the calculation. Upon review of NUREG-0933, "Resolution of Generic Safety Issues," Issue 70, "PORV and Block Valve Reliability (Rev. 3)," the inspectors determined that the seismic contribution to instrument uncertainty needed to be addressed. If instrument uncertainty, without the seismic consideration, was included in the calculation, the results would be a negative margin of 49.3 psi (19.3 psi – 68.6 psi) to the Appendix G limit of 621 psig.

The inspectors were concerned that the existing PORV setpoint would not limit an overpressure condition such that Appendix G limits could be exceeded. In summary, based on the above, the available margin to the Appendix G limit was as follows:

- 435 psig, the existing PORV setpoint;
- + 88.7 psi error determined in the response to Westinghouse/IN 93-58;
- + 68.6 psi instrument uncertainty (not including the seismic contribution);
- +78 psi maximum overshoot applied to the LTOP setpoint (12-N1-05, Rev. 13); and
- = 670.3 psig which exceeded the Appendix G limit of 621 psig.

In response to inspectors concerns, the licensee's initiated AR 2010-10197 to further evaluate the margin issue. Westinghouse performed an evaluation that was documented in letter, AEP-10-127, dated September 24, 2010. This evaluation identified approximately 50 psi of additional margin based on changes in P-T limits calculation methodology or removal of excessive calculational conservatisms (e.g., use of ASME Boiler and Pressure Vessel Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves, Section XI, Division 1"). Applying the 50 psi margin available via Code Case N-640, the Appendix G limit would increase to 671 psig (621 psig + 50 psi). The Westinghouse letter also identified an analytical margin of approximately 35 psi based on the margin between the current PORV setpoint and the maximum allowable setpoint and conservatisms in the LOFTRAN computer code analyses. Therefore, the overall margin would be approximately 35.7 psi (- 49.3 psi + 50 psi + 35 psi). The licensee also concluded that any seismic contribution to instrument uncertainty would be small and would not exceed the margin determined by Westinghouse. The licensee concluded that based on the above information from Westinghouse, the PORV setpoint was acceptable and the PORVs were operable but non-conforming.

With the apparent 35 psi of margin, there was reasonable assurance that the existing LTOP setpoint of 435 psig will provide adequate overpressure protection, even when considering the seismic contribution to instrument uncertainty. The inspectors concluded that the above analysis was acceptable from an operability standpoint.

Although there was no immediate technical concern based on the licensee's analysis, the licensee disagreed with the NRC's position on whether the inclusion of instrument uncertainty in the LTOP setpoint was within their license basis. The licensee did provide the inspectors with correspondence during the upgrade to Improved Technical Specifications where they informed the NRC that instrument uncertainty was not included when the LTOP setpoint was established. However, since the setpoint was not being revised, it was not clear whether Office of Nuclear Reactor Regulations (NRR) reviewed and agreed with the licensee's position on instrument uncertainty. Since the inspectors were unable to verify whether the use of instrument uncertainty was part of the license basis and the perceived difference between the recommended actions in the IN and the Westinghouse letter, this issue is considered an unresolved item (URI 000315/2010006-02; 000316/2010006-02) pending further discussions with NRR to resolve these two concerns.

.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 as part of this inspection:

- Bulletin 1988-04, “Potential Safety-Related Pump Loss”;
- IN 1993-58, “Nonconservatism in Low-Temperature Overpressure Protection For Pressurized-Water Reactors”;
- IN 1997-41, “Potentially Undersized Emergency Diesel Generator (EDG) Oil Coolers”;
- IN 2007-36, “Emergency Diesel Generator Voltage Regulator Problems”; and
- RIS 2005-29, “Anticipated Transients That Could Develop into More Serious Events.”

b. Findings

No findings of significance were identified.

.5 Modifications

a. Inspection Scope

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

- 12-MOD-45617, Supplemental Diesel Generators;
- EC-0000047412, Degraded Voltage Back-Fit Modification; and
- EC-0000048552, Voltage Regulation for EP Alternate Offsite Power Source.

b. Findings

No findings of significance were identified.

.6 Risk-Significant Operator Actions

a. Inspection Scope

The inspectors performed a margin assessment and detailed review of two risk significant, time critical operator actions. These actions were selected from the licensee’s PRA rankings of human action importance based on risk achievement worth values. Where possible, margins were determined by the review of the assumed design basis and UFSAR response times and performance times documented by job performance measures results. For the selected operator actions, the inspectors performed a detailed review and walk through of associated procedures, including observing the performance of some actions in the station’s simulator and in the plant for other actions, with an appropriate plant operator to assess operator knowledge level, adequacy of procedures, and availability of special equipment where required.

The following operator actions were reviewed:

- Mitigation of a Loss of Coolant Accident While In Mode 4; and
- Restoration of 4kV Power Using One Supplemental Diesel Generator.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

.1 Review of Items Entered Into the Corrective Action Program

a. Inspection Scope

The inspectors reviewed a sample of the selected component problems that were identified by the licensee and entered into the 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

The inspectors presented the CDBI inspection results to Mr. L. Weber, and Mr. M. Carlson, and other members of the licensee staff on September 17, 2010, and on October 21, 2010, respectively. The licensee acknowledged the issues presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. 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.

.2 Interim Exit Meeting

Triennial Heat Sink inspection results were discussed with Mr. J. Gebbie, Site Vice President, on February 12, 2010. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
L. Weber, Site Vice President-Chief Nuclear Officer
M. Carlson, Site Support Vice President
J. Gebbie, Site Vice President
S. Lies, Plant Manager
G. Curten, Mechanical Design Engineering
R. Ebright, Engineering Director
D. Etheridge, Operations Support Manager
R. Hall, ISI Program Engineer
W. Hodge, I&C Design Supervisor
B. Mammoser, Mechanical Design Supervisor
D. Naughton, Design Engineering Manager
K. O'Conner, Compliance Manager
J. Phelan, Design Engineering Support Team
R. Pickard, Engineering Program Supervisor
J. Ross, Operations Director
M. Scarpello, Regulatory Affairs Manager
G. Truini, ENSE Supervisor
C. Vanderzwaag, System Manager
P. Vandevisse, Design Engineering
R. West, Licensing Activity Coordinator
J. Whitten, Electrical Design Engineering
T. Woods, Performance Assurance Director

Nuclear Regulatory Commission
J. Lennartz, Senior Resident Inspector
P. LaFlamme, Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened/Closed

000315/2010006-01; 000316/2010006-01	NCV	Inadequate As-Found Heat Exchanger Inspection Guidance and Acceptance Criteria
000315/2010006-02; 000316/2010006-02	URI	Non Conservative Analysis Used to Determine LTOP Setpoint for PORV

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
12-E-S-ELCP-4KV-002	Bus Transfer Calculation	4
1-2 UNC-211 Calc2	RCS Wide Range Pressure	2
12-E-N-4KV-RPT2	Surge Protection for 4 kV Vacuum Breakers	0
12-E-N-ELCP-4KV-002	Unit 1 and Unit 2 Aggregate Load	1
12-E-N-ELCP-EDG-05	Diesel Generator Transient Analysis	0
12-E-S-EPEDG-001	Allowable Outage Time (AOT) EDG ETAP Analysis	3
1-2-I9-03 Calc 7	RWST Minimum Tech Spec Volume	1
1-2-UNC-339, Calc 3	Setpoint Calc for RWST Level Alarms, RHR Pump Trip Interlock, and Operations Points	1
2-E-N-ELCP-120-002	120Vac Control Room Instrument Distribution (CRID) Loading and Voltage Drop Analysis – Unit 2 Calculation	3
2-E-N-ELCP-250-0001	Unit 2 250 VDC Coordination Study	0
2-E-N-ELCP-250-006	250Vdc Battery 2CD System Analysis	6
2-E-N-ELCP-4KV-001-FAULT	Unit 2 4kV/600V Fault Calculation	0
2-E-N-ELCP-4KV-001-MODEL	Unit 2 4KV/600V Load Control Model	7
2-E-N-ELCP-4KV-001-VOLT	Unit 2 Equipment Loading Analysis	0
2-E-N-ELCP-4KV-001-VOLT	Unit 2 Voltage Adequacy Analysis	1
2-E-N-PROT-RLY-010	Bus Overvoltage Relay Setting	0
2-E-N-PROT-RLY-02A	4kV Safety Related Motors Phase Instantaneous Relay (PJC) Settings	0
2-O-0-45	Assessment of Unit 2 EDG Capability	1
3-SA-096.026	CCI Calculation-Structural Analysis of Wall Strainer Cartridge	
ALION-CAL_AEP-3085-16	DC Cook Units 1 and 2: Summary of Debris Transport and Generation	1
ECP No. 12-N1-05	LTOP Setpoint Calculation	13
ENSM 970606JJR	RWST Vortexing	2
MD-02-ECCS-003-N	DC Cook Unit 2 ECCS & CVCS Proto-Flo Model	0
MD-02-ECCS-005-N	Unit 2 ECCS Pumps NPSH Analysis	5
MD-02-RH-033-N	Torque Limits for Motor Operated Valves 2-IMO-350/340	3
MD-02-RH-119-N	Analysis of Thrust & Torque Limits for MOV 2-ICM-305 and 2-ICM-306	2

CALCULATIONS

Number	Description or Title	Revision
MD-02-RH-195-N	Analysis of Thrust and Torque Limits for MOV 2-ICM-129/ 2-IMO-128	3
MD-12-CA-004-S	Determination of Available Pressurizer PORV Strokes Using the Auxiliary Air Supply	1
MD-12-CCW-001-N	CCW Heat Exchanger Tube Plugging Criteria	0
MD-12-CCW-011-N	U1 & 2 CCW System Analysis to Relate Accident, Testing and Minimum Operability Requirements	0
MD-12-DG-010-S	Determination of Minimum Fuel Oil Level to Allow for 24 Hours SDG Operation at 55% Load and Maximum Run Time of SDG Operation at 100% Load	1
MD-12-DG-011-N	Effect Of EDG Frequency On Pump BHP	0
MD-12-ECCS-016-N	Centrifugal Charging, RHR Deadheading and Emergency Boration Analysis	1
MD-12-ESW-078-N-ADD	EDG Cooler Tube Plugging Allowance	D
MD-12-ESW-101-N	CCW HX Partition Plate Modification	0
MD-12-ESW-106-N	Assessment of Increased Lake Water Temperature on Safety Related and Non-Safety Related Systems	4
MD-12-HV-013-N	AB & CD Battery Rooms Hydrogen Evolution	1
MD-12-HV-021-N	Switchgear & Battery Rooms Heat Gain Calculation	4
MD-12-HV-026-N	CD Battery Room Steady State and Station Blackout Transient Temperatures-Units 1 & 2	2
MD-12-MSC-041-N	MOV Parameter Calculation for Valves 1/2 IMO-316/326/ 360/361/362, 1/2 ICM-305/306/311/321, and 1/2QMO-200/201/225/226, 1-QMO-410, 2-QMO-420	5
MD-12-MSC-043-N	D.C. Cook Generic Letter 89-10 Program Determination of Water Inertia Loads for Valves 1-ICM-305/306, 1-IMO-310/320, 2-ICM-305/306, 2-IMO-310/320	2
MD-12-MSC-056-N	MOV Parameter Calculation, Flow Rates, Temperatures and Pressures for Valves 1/2 IMO-261, 390, 310/320, 312/322, 314/324, 340/350	4
MD-12-MSC-068-N	Tube Plugging Allowances for Safety-Related HXs	1
MD-12-MSC-076-N	IST Program Pump Minimum Operability Limits (MOL)	1
MD-12-RCS-021-N	Maximum Differential Pressure Calculation for Pressurizer Power Operated Relief Valves 1(2)-NRV-151, 152, & 153	0
MD-12-RCS-022-N	Actuator Capability	0
MD-12-RH-040-N	Maximum Differential Pressure During Operation of RHR Valves 1-IMO-310, 1-IMO-312, 1-IMO-314, 1-IMO-320, 1-IMO-322, 1-IMO-324, 1-IMO-340, 1-IMO-350, 2-IMO-310, 2-IMO-312, 2-IMO-314, 2-IMO-320, 2-IMO-322, 2-IMO-324, 2-IMO-340, & 2-IMO-350	4
MD-12-RH-138-N	Maximum Differential Pressure During Operation of RHR Recirculation Sump Valves 1/2-ICM-305 & 306	1

CALCULATIONS

Number	Description or Title	Revision
MD-12-RH-210-N	MOV Parameter Calculation for RHR Valves 1/2 IMO-128 & 1/2 ICM-129	2
MD-12-RH-211-N	Maximum D/P During Operation of RHR Shutdown Cooling Suction Isolation Valves 1/2-IMO-128, 1/2-ICM-129	2
MD-12-RH-212-N	Determination of Water Inertia Loads for Miscellaneous ECCS Valves - 1/2-IMO-255, IMO-256, IMO-314, IMO-324, IMO-330, IMO-331, IMO-340, IMO-350, IMO-390	6
MD-12-RHR-004-N	EPRI PPM Evaluation of 1/2-ICM-129 and 1/2-IMO-128	1
MD-12-RHR-105-N	EPRI PPM Evaluation of 1/2-ICM-305 and 1/2-ICM-306	0
MD-12-RHR-110-N	EPRI PPM Evaluation of 1/2-IMO-340 and 1/2-IMO-350	1
MD-12-RHR-901-N	RHR and CCW HX UA Determination	0
MD-12-RHR-904-N	Pressure Locking Evaluation for MOV'S 1/2-ICM-305 and 1/2-ICM-306	0
MD-12-RWST-001-N-ADD	Maximum Differential Pressure for RWST Vent Path	0
MD-12-SUMP-001-N	Containment Recirculation Sump Function Margins	0
MD-CW-005-N	Flood Protection Features	1
PS-CRID-004	120Vac Vital Bus System(CRID) Voltage Study	0
SD-010405-001	Structural Qualification of Unit 1/2 CCW Heat Exchangers	0
SD-070104-00	Qualification for Anchorage for Main Strainer	1

CORRECTIVE ACTION DOCUMENTS GENERATED DUE TO THE INSPECTION

Number	Description or Title	Date
10042008	Oil Leak on 2-CMO-410	02/10/10
10042010	Power Cables Exposed Conductor Improperly Abandoned in Place	02/10/10
10042019	Erroneous References Listed in CCW DBD	02/10/10
10042024	Go Calculation MD-12-RHR-901-N, Revision 0 was No Longer Valid	02/11/10
10042037	Go Calculation in System of Records Missing Significant Page	02/11/10
10042043	Bent and Tilted Fire Sprinkler Deflector Plate at 2-WMO-734 J	02/11/10
2010-8257	Valve 2-IMO-340 Local Label Missing	08/17/10
2010-8298	Calculation MD-12-DG-011-N, Rev. 4 Cannot Be Found	08/18/10
2010-8301	Plastic Sleeve on Drain to U2 East RHR Pump Room Sump	08/17/10
2010-8334	OP-2-98046-32 Drawing Discrepancies	08/19/10
2010-8336	Review Basis for LTOP PORV Stroke Times	08/19/10
2010-8347	Load Changes on CRID II Not Update; Update Table in 8.1.1	08/19/10
2010-8756	Clarification Needed in DBD-12-4KV	08/30/10
2010-8812	ARs Not Initiated for Trending Purposes	08/31/10
2010-8865	CDBI Identified Issues for 2-IMO-340	09/01/10
2010-8974	Need for GL 89-13 Program Assessment Identified	09/03/10
2010-8983	Operator Response Time for Mode 4 LOCA Exceeded Time Limit	09/03/10
2010-8996	Timeliness of Approval of Heat Exchanger Inspection Reports	09/03/10

CORRECTIVE ACTION DOCUMENTS GENERATED DUE TO THE INSPECTION

Number	Description or Title	Date
2010-8997	Review of IN 2008-02 Not Completed in Timely Manner	09/03/10
2010-8998	Timeliness of Submittal of Heat Exchanger Inspection to NDM	09/03/10
2010-9096	CR Inappropriate Eval & Identified Condition Not Addressed	09/08/10
2010-9468	Allowable Voltage for 69 KV(EP)	09/15/10
2010-9515	RHR Pump Mini-Flow During SBLOCA and IE 88-04 Response	09/16/10
2010-9596	Improper Application of Inst Uncertainty for RHR Testing	09/17/10
2010-9603	Training Has Not Used Validated Times for Operator Actions	09/17/10
2010-10197	LTOP Analysis Does Not Apply Instrument Uncertainty	10/01/10

CORRECTIVE ACTION DOCUMENTS REVIEWED DURING THE INSPECTION

Number	Description or Title	Date
AR00089572	Why NRC-Identified ESW Nonconforming Condition before Cook	04/15/04
AR00800287	Unverified Assumption in Debris Loading in Sump Strainer	10/18/07
AR00807351	Create Recurring Task for Partial Retubing of 1-HE-15W	01/01/07
AR00820569	Goodway Cable and Brush Became Lodged in 2-HE-15W	10/12/07
AR00822377	Evaluation of IN 2007-36: EDG Voltage Regulators	12/19/07
AR00826662	TR5 Lightning Arrester Failure	02/24/08
AR00830143	1-HE-18W South Side Bottom End Cover Leaking 60 Dpm	04/21/08
AR00831630	Additional Cells Weeping Electrolyte	05/14/08
AR00833190	WO for CCW EAST CCW HX Performance Test Has Been Repeatedly Scoped Into and Dropped From Unit 2 Outages	06/13/08
AR00835211	PRB is Not Intrusive Enough for NRC OE Products	07/18/08
AR00849840	Sand Removed From 2-HE-15E was Found Contaminated	04/13/09
AR00851615	Replacement of the U2 CCW Heat Exchangers 2-HE-15E/W Need to be Added to the Long Range Plan	05/18/09
AR00851616	Replacement of the U1 CCW Heat Exchangers 1-HE-15E/W Needs to be Added to Long Range Plan	05/18/09
AR00857912	Modification Package May be Required to Re-Tubing of the HX	09/23/09
AR00858746	U1 West CTS HX ESW Outlet Piping is Peeling and Flaking	10/06/09
AR00860154	Self-Assessment for 2010 Triennial Heat Sink Inspection	01/04/10
AR2010-3532	Xfmr 201AB Neutral/Ground Overcurrent Fault	10/03/09
CR01033017	TS 3.4.4 LCO for Pzr Water Level is Potentially Non-Conservative	02/01/01
CR04104047	NRC Information Notice 2004-07, Evaluation	04/13/04
CR05168031	DP Concern With Containment Sump Recirculation Valves	06/21/05
GT00005944	Calculation Does Not Exist to Determine Station Battery Life	03/02/99
P-00-05883	Tracking CR: SRRB Question Regarding Operator Action Time to Preclude Pressurizer Overfill From Stuck Open Spray Valve	04/20/00
P-98-04610	NSAL 98-007 Modeling Issues Related to Pressurizer Filling	09/01/98
P-99-16531	The Time to Overfill Pressurizer due to an Inadvertent Safety Injection Needs to be Verified	06/24/99
P-99-27057	Pressurizer Overfill Caused by FWLB Needs Analysis	11/09/99

DRAWINGS

Number	Description or Title	Date or Revision
2-84851-9	Pressurizer Power Relief Valves NRV-151 & NRV-152	9
2-AEP-MASN-011498-516	Pressurizer PORV Control Valve-2" Model 38-20771	7
8504-243500	MLW-Worthington Limited Drawing for CCW Heat Exchangers Shell and Channel Details	03/08/73
CS-12-91001-0	Supplemental Diesel Generator 4160V CAT SWGR	06/30/06
CS-12-91002-0	Supplemental Diesel Generator SDG Generator 1	06/30/06
CS-12-91003-0	Supplemental Diesel Generator SDG Generator 2	06/30/06
CS-12-91004-0	SDG Communications and Main Control Room	06/30/06
E-1000A-34	Donald C. Cook Nuclear Plant Unit 1 One Line Diagram	34
E-1300	345/34.5KV One Line Diagram	16
E-2847-8	69/4kV Elementary Diagram DC Cook Station	8
OP-12-12007-14	Miscellaneous Aux System One Line Offsite Plant Service	14
OP-2-12001-44	Main Auxiliary One-Line Diagram Bus A & B Engineered Safety System (Train B)	44
OP-2-12002-38	Main Auxiliary One-Line Diagram Bus C & D Engineered Safety System (Train A)	38
OP-2-12003-28	250Vdc Main One Line Diagram Engineered Safety System (Train "A, B, N & BOP")	28
OP-2-12006-31	Miscellaneous Aux System One-Line Offsite Plant Service	31
OP-2-12007-14	Miscellaneous Aux System One-Line Offsite Plant Service	14
OP-2-12030-32	Misc Aux One-Line 600V bus 21C, 21D, Engineered Safety System (Train "A")	32
OP-2-12031-27	Misc Aux One-Line 600V bus 21C, 21D, Engineered Safety System (Train "A")	19
OP-2-12032-14	Misc Aux One-Line 600V bus 21C, 21D, Engineered Safety System (Train "A")	14
OP-2-12033-19	Misc Aux One-Line 600V bus 21C, 21D, Engineered Safety System (Train "A")	19
OP-2-12050-26	120V AC Control Room Instrument Distribution Cabinets "CRID-I" Thru "CRID-IV" Engineered Safety System	26
OP-2-12060-25	DC Aux One Line 250Vdc Bus AB Engineered Safety System	25
OP-2-12070-26	DC Aux One Line 250Vdc Bus CD Engineered Safety System	26
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OP-2-98050-26	Reserve Bus Tran. & Auxiliary Buses Low Voltage Protection	26
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PS-2-92530-12	Input/Output Cabinets Computer Power Distribution	12
PS-2-925421-1	Plant Process Computer (PPC) and Site LAN Split Unit 2	1
QB-12847	CCW Heat Exchanger Channel	A
QB-12857	CCW Heat Exchanger Gaskets	08/11/69

MISCELLANEOUS

Number	Description or Title	Date or Revision
	2-HE-15E CCW Eddy Current Test	10/04
	Eddy Current Inspection AEP-DC Cook Unit 2 U2C18	04/17/09
	U2 CCW Heat Exchanger ECT Done During U2C18 Outage by Intech, Inc	04/17/09
02-OHP-4022-022-15	EOP Verification and Validation Report, Mode 4 LOCA	04/10/03
2-DCP-4392, Att 3	250Vdc Fuse Replacement	0a
2-Figure 19.8	Safety Related Throttle Valves; CCW	37
680-41400	Containment Sump Strainer Replacement: Large Size Head Loss Test Report	0
Cable Pull Cards	33003-1, 33004-1, 33005-1, 8669R-2, 8674R-2, 8755R-2, 8760R-2, 8847R-2, 8872R-2, 8873R-2, 9705CR-2, 9705PR-2, 9706CR-2, 9706PR-2, 9707R-2	2006
CEEQ File EQ-0557	ASCO Solenoid Valves 1,2-XSO-503	3
DIT B-00621-09	Units 1 and 2 AAC Powered GL 89-10 Operated Valves	0
DIT-B-00175-00	AB & CD Battery Rooms Hydrogen Generation	09/14/99
DIT-B-00175-02	AB & CD Battery Rooms Temperatures	03/13/00
DIT-B00459-01	Subject: RWST Flow Rates	03/14/00
DIT-B-00760-05	Design Basis Performance Parameter for the CCW Pump	01/16/08
DIT-B-00802-10	CCW Flow Balance Criteria for Procedure 01-EHP-4030-116-248 & 02-EHP-4030-216-248	09/23/03
DIT-B-01061-08	EOP Operator Action Times from Accident Analyses	04/22/03
DIT-B-02706-01	Provide Testing Methodology for ESW Served HXs	04/26/03
DIT-S-00945-03	250Vdc Battery Qualified Life and 4-Hour Discharge Rate	09/29/07
EC-0000049530	Replace SDG's Flat Screen Controller 2-SDG-FSC	0
ECP No. 1-2-19-03	RWST Level Transmitters Scaling and Volume Setpoint	19
ECP-1-2-O0-14	Emergency Operating Procedure Footnotes	18
EHI-5071	IST Program Implementation	11
Figure 2-15.1	Technical Data Book: IST Action Setpoints for CCW Pump	101

MISCELLANEOUS

Number	Description or Title	Date or Revision
ICP-01102	Improved Tech Spec Changes to LTOP Admin. Controls	06/01/05
Ltr. AEP:NRC:1065	DC Cook Response to NRC IEB 88-04	07/08/88
Ltr. AEP:NRC:4321	Degraded Voltage Protection	12/10/04
Ltr. AEP:NRC:7321	Degraded Voltage Protection	03/30/07
Ltr. AEP:NRC:90309	American Electric Power Review of NRC IN 97-41	07/30/01
Ltr. AEP-00-249	RHR Pump Minimum Flow (Westinghouse)	10/17/00
Ltr. AEP-10-127	Conservatisms in Pressure-Temperature Limit Curves and LTOPS PORV Setpoint (Westinghouse)	09/24/10
Ltr. AEP-92-096	LTOP Evaluation (Westinghouse)	05/22/92
Ltr. AEP-93-208	Infograms: IG93003 and IG 93004	03/10/93
Ltr. AEP-97-239	Reduced RHR Spray Flow (Westinghouse)	12/09/97
Ltr. NRC to AEP	Degraded Voltage Protection	09/26/07
MDS-607	Heat Exchanger Tube Plugging	7
Order 31603	MLW-Worthington Form U-1 Manufacturer Report	06/19/70
Order 31760	MLW-Worthington Limited Specification Sheet for CCW Heat Exchangers	09/05/72
PMP-4030-001-001	Impact of Safety Related Ventilation on the Operability of TS	9
Report R20029OP	Aquatic Sciences Project Report of Sonar Inspection of Cook Nuclear Plant Screenhouse Forebay	08/14/09
SS740001	Engineering Guide, AEP Surge Arrester Application	2
TS-O-3020	Perform Heat Exchanger Inspection	2
VTD-INDR-0001	Pump Curve for Unit 2 East CCW Pump	06/03/71
VTD-INDR-0012	Unit 2 RHR Pump Curve	06/13/71

MODIFICATIONS

Number	Description or Title	Revision
12-MOD-45617	Addition of Supplemental Diesel Generators (SDGs)	1
EC- 0000047742	RHR Open Crosstie Modification for Unit 2	0
EC-0000047412	Degraded Voltage Back-Fit Modification	0
EC-0000048552	Voltage Regulation for EP Alternate Offsite Power Source	0

PROCEDURES

Number	Description or Title	Revision
12-EA-6090-ENV-101	Zebra Mussel Sampling and Analysis	3
12-EA-6090-ENV-106	Bio-Monitoring Equipment Installation Control	4
12-EHP-4030-001-001	Check Valve Examination Surveillance	5
12-EHP-4030-216-248	CCW Flow Balance	4
12-EHP-8913-001-001	Program for Implementing GL 89-13 Inspections	1
12-EHP-8913-001-002	Heat Exchanger Inspection	3
12-IHP-4030-082-001	AB, CD and N-Train Battery Weekly Surveillance and Maintenance	17

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12-IHP-4030-082-002	AB, CD and N-Train Battery Quarterly Surveillance and Maintenance	18
12-IHP-4030-082-003	AB, CD and N-Train Battery Discharge Test and 24 Month Surveillance	21
12-IHP-4030-082-006	AB, CD and N-Train Battery Yearly Surveillance and Maintenance	7, 8
12-IHP-5021-EMP-008	Battery Connection Maintenance	9
12-MHP-4030-031-001	Containment Recirculation Sump Inspection	1
12-MHP-5030-016-001	Component Cooling Water Heat Exchanger Inspection, Cleaning and Tube Plugging	7
12-OHL-4030-SOM-020	Unit 12 Tours - Outside TS Tour - Days.pdf	8
12-OHL-4030-SOM-046	Unit 12 Tours - Outside TS Tour - Nights.pdf	3
12-OHP-4021-033-001	Supplemental Diesel Generator Operations	4
12-OHP-4021-082-037	Operation of 120V UPS for EP Voltage Regulators	000
12-OHP-4024-033-SDG-1 Composite	Annunciator #033 Response: SDG 1	2
12-OHP-4024-033-SYS Composite	Annunciator #033 Response: System	3
12-OHP-4030-033-001	Supplemental Diesel Generator Testing	13
12-THP-6020-CHM-304	Essential Service Water	4
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2-OHP-4021-082-006	Operation of 2AB & 2CD Battery Chargers	12
2-OHP-4022-002-015	Mode 4 LOCA	13
2-OHP-4022-016-001	Malfunction of the CCW System	8
2-OHP-4022-016-004	Loss of Component Cooling Water	18
2-OHP-4022-082-02CD	Loss of Power to 250 Vdc Bus 2CD	5
2OHP-4023-ECA-0.0,	Loss of All AC Power	24
2-OHP-4023-ES-1.2	Post LOCA Cooldown and Depressurization	13
2-OHP-4023-SUP-009	Restoration of 4kV Power from EP	6
2OHP-4023-SUP-009	Restoration of 4kV Power from EP	6
2-OHP-4024-220	Annunciator #220 Response: Station Auxiliary CD	16
2-OHP-4024-221	Annunciator #221 Response: Generator	27
2-OHP-4030-216-020V	CCW Valve Actuation	3
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EHI-8913	Program for Implementing Generic Letter 89-13	5
ENVI-8913	Zebra Mussels Monitoring and Control Program	7
PMP-4030-EIS-001	Event-Initiated Surveillance Testing	21

SURVEILLANCES (COMPLETED)

Number	Description or Title	Date
12-EHP.SP.133.002	Auxiliary Building Ventilation System Air Balance	06/12/00
2-OHP-4030-202-060-A2	PZR Power Operated Relief Valve Testing, NRV-152 and 2-CA-711	07/27/09
2-OHP-4030-216-020E	East CCW Group A Pump Test and Quarterly Valve Test	06/02/10
2-OHP-4030-217-050E	East Train RHR Operability Test, Modes 1-4	07/08/10

WORK DOCUMENTS

Number	Description or Title	Date
JO03135015-06	SWSIP Internal Visual Examination Checklist	05/29/03
JO05088013-14	12-MOD-45617, Closeout/Cleanup, Operation Mode Check	11/03/05
JOR0101841	2-EHP-4030-216-248 – CCW Flow Balance	06/06/03
WO55226851 01	2-BATT-CD Performance Test (60 Month Surveillance)	10/20/07
WO55234570-05	2-ESW-102E, Disassemble and Inspect	11/17/09
WO55234870-03	2-ESW-102E, Remove/Install Valve	11/23/09
WO55238049-02	2-HE-15E Plug HX Tubes Per Engineering Analysis	05/24/09
WO55247485 02	2-TDAB & 2-TDCD, Clean Inspect Fuses	10/04/07
WO55255498-03	2-HE-15E CCW Hx Visual Inspection and Cleaning	10/07/07
WO55259596-01	Perform Sonar Mapping of Forebay West of Trash Racks	12/20/06
WO55273132-04	2-ESW-102E, Remove Install Valve 86-03 Inspection	12/08/03
WO55285068	Intake and Discharge Structure Inspection and Cleaning	12/21/07
WO55287662-01	Perform Sonar Mapping of Forebay West of Trash Racks	09/16/09
WO55288313 01	2-BATT-AB Performance Test (60 Month Surveillance)	10/04/07
WO55303837	Intake and Discharge Structure Inspection and Cleaning	02/09/09
WO55308244-15	2-HE-15W CCW Heat Exchanger Inspection	03/28/09
WO55309466-05	2-HE-15E CCW Hx Visual Inspection and Cleaning	04/08/09
WO55309853 01	2-BATT-AB Service Test (24 Month Surveillance)	04/05/09
WO55309854 01	2-BATT-CD Service Test (24 Month Surveillance)	04/14/09
WO55335787-01	Sonar Inspection on East Pump Side of U-1 Traveling Screen	10/30/09
WO55335789-01	Sonar Inspection between Traveling Screens & Trash Racks	09/16/09
WO55354056 01	2-BATT-AB 92 Day Surveillance	02/25/10
WO55358476 01	2-BATT-CD 92 Day Surveillance	05/13/10
WO55363561 01	2-BATT-CD 92 Day Surveillance	08/12/10
WO55366899 01	2-BATT-CD 7 Day Surveillance	07/21/10
WO55366900 01	2-BATT-AB 7 Day Surveillance	07/21/10
WO55367257 01	2-BATT-CD 7 Day Surveillance	07/28/10
WO55367258 01	2-BATT-AB 7 Day Surveillance	07/28/10
WO55367591 01	2-BATT-DCD 7 ay Surveillance	08/04/10
WO55367592 01	2-BATT-AB 7 Day Surveillance	08/04/10
WO55367931 01	2-BATT-CD 7 Day Surveillance	08/01/10
WO55367932 01	2-BATT-AB 7 Day Surveillance	08/01/10

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access Management System
AR	Action Request
ASME	American Society of Mechanical Engineers
CCW	Component Cooling Water
CDBI	Component Design Bases Inspection
CFR	Code of Federal Regulations
DRS	Division of Reactor Safety
EC	Engineering Change
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
ESW	Essential Service Water
GL	Generic Letter
gpm	Gallons per Minute
IEEE	Institute of Electrical & Electronic Engineers
IMC	Inspection Manual Chapter
IN	Information Notice
IP	Inspection Procedure
IR	Inspection Report
IST	Inservice Test
kV	Kilovolt
LOCA	Loss of Coolant Accident
LTOP	Low Temperature Overpressure Protection
MOV	Motor-Operated Valve
NCV	Non-Cited Violation
NPSH	Net Positive Suction Head
NRC	U.S. Nuclear Regulatory Commission
PARS	Publicly Available Records
PORV	Power Operated Relief Valve
PRA	Probabilistic Risk Assessment
psi	pounds per square inch
psig	pounds per square inch gauge
P-T	Pressure and Temperature
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RIS	Regulatory Issue Summary
RWST	Refueling Water Storage Tank
SDG	Supplemental Diesel Generator
SDP	Significance Determination Process
SPAR	Standardized Plant Analysis Risk
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
V	Volts
Vac	Volts Alternating Current
Vdc	Volts Direct Current
WO	Work Order

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Sincerely,

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

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