April 16, 2007

Mr. Mano K. Nazar
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 NRC COMPONENT DESIGN BASES INSPECTION (CDBI) REPORT 05000315/2007002(DRS); 05000316/2007002(DRS)

Dear Mr. Nazar:

On March 2, 2007, the U .S. Nuclear Regulatory Commission (NRC) completed an inspection at your Donald C. Cook Nuclear Power Plant, Units 1 and 2. The enclosed report documents the inspection findings which were discussed on March 2, 2007, with Mr. Mark Peifer and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety, and to compliance with the Commission's rules and regulations, and with the conditions of your license. The inspectors reviewed selected calculations, design bases documents, procedures, and records; observed activities; and interviewed personnel. Specifically, this inspection focused on the design of components that are risk significant and have low design margin.

Based on the results of this inspection, three NRC-identified findings of very low safety significance were identified, all of which involved violations of NRC requirements. However, because these violations were of very low safety significance and because they were entered into your corrective action program, the NRC is treating the issues as Non-Cited Violations (NCV) in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

If you contest the subject or severity of a NCV, you should provide a response with a basis for your denial, within 30 days of the date of this inspection report, 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 NRC Resident Inspector at the Donald C. Cook facility.

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), accessible from the NRC Web site 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 05000315/2007002; 05000316/2007002(DRS)
w/Attachment: Supplemental Information

cc w/encl: M. Peifer, Site Vice President
L. Weber, Plant Manager
S. Simpson, Regulatory Affairs Manager
G. White, Michigan Public Service Commission
L. Brandon, Michigan Department of Environmental Quality - Waste and Hazardous Materials Division
Emergency Management Division
MI Department of State Police
State Liaison Officer, State of Michigan

component of NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-315; 50-316
License Nos:	DPR-58; DPR-74
Report No:	05000315/2007002(DRS); 05000316/2007002(DRS)
Licensee:	Indiana Michigan Power Company
Facility:	Donald C. Cook Nuclear Power Plant, Units 1 and 2
Dates:	January 29 through March 2, 2007
Location:	Stevensville, MI
Inspectors:	G. Hausman, Senior Engineering Inspector (Lead) C. Baron, Mechanical Contractor J. Chiloyan, Electrical Contractor J. Jacobson, Senior Engineer Inspector B. Jose, Engineering Inspector D. Passehl, Senior Reactor Analyst D. Reeser, Operations Inspector
Observer:	M. Jones, Engineering Inspector (Training)
Approved by:	Ann Marie Stone, Chief Engineering Branch 2 Division of Reactor Safety (DRS)

SUMMARY OF FINDINGS

IR 05000315/2007002(DRS); 05000316/2007002(DRS); 02/2/2007 - 03/02/2007;
Donald C. Cook Nuclear Power Station, Units 1 and 2; Component Design Bases Inspection.

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. The inspection was conducted by regional engineering inspectors and two consultants. Three findings of very low safety significance were identified with three associated Non-Cited Violations (NCVs). 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 3; dated July 2000.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding having very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action" for the licensee's failure to promptly identify that the Unit 1 Train A (1-CD) emergency diesel generator (EDG) would exceed its capacity rating. Specifically, the 1-CD EDG's capacity rating would have been exceeded if the 1-CD EDG was allowed to run at the upper frequency band of 61.2 Hz as allowed by Technical Specifications. As a result, the licensee performed corrective action calculations to assess the finding and on March 1, 2007, imposed an operational upper frequency limit of $\leq$60.5Hz on the station's Unit 1 EDGs. This finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program because the licensee did not take appropriate corrective action to address the safety issue in a timely manner commensurate with its safety significance and complexity.

  This finding was more than minor because the 1-CD EDG would have exceeded its design load rating at the maximum TS allowed frequency of 61.2Hz. Without the evaluation and imposing an administrative limit, the licensee could not ensure that the 1-CD EDG would reliably perform its safety related function. The finding was of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." (Section 1R21.3b)

Cornerstone: Barrier Integrity

- Green. The inspectors identified a finding having very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action" for failure to promptly identify and correct a condition adverse to quality regarding inadequate safety analysis dose calculations. Specifically, the licensee failed to address the aggregate effect of various nonconforming conditions on containment leakage rates for offsite dose and control room calculations to ensure that accurate and adequate margin remained available for offsite dose analyses and control room habitability. The finding was entered into the licensee's corrective action program and an operability determination evaluation (ODE) was initiated during the inspection. The primary cause of this violation was related to the cross-cutting area of problem identification and resolution because the licensee did not thoroughly evaluate known discrepant conditions.

This finding was more than minor because the licensee did not verify the capability of containment to maintain the offsite and control room dose within required limits under post-accident conditions to the values assumed in the analyses. The finding was of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." (Section 1R21.4b.1)

- Green. The inspectors identified a finding having very low safety significance and an associated NCV of 10 CFR Part 50.36, "Technical Specifications." Specifically, the licensee failed to maintain previously imposed administrative limits (i.e., compensatory measures) required by non-conforming updated final safety analysis report (UFSAR) offsite and control room dose analyses. The station operated from April 25, 2003, through February 28, 2007, based on analyses that included assumed containment leakage values that were not bounded by the licensee's TS 5.5.14, "Containment Leakage Rate Testing Program." Once the finding was identified by the inspectors, the licensee re-imposed the required compensatory measures during the inspection. The primary cause of this violation was related to the cross-cutting area of human performance because the licensee failed to communicate decisions with respect to containment leakage and the basis for those decisions to personnel.

  The finding was more than minor in accordance with IMC 0612, Appendix B because the finding was associated with the configuration control (containment design parameters maintained) attribute of the Barrier Integrity Cornerstone and affected the cornerstone's objective of maintaining the functionality of containment. Specifically, the licensee did not re-impose compensatory measures to limit the maximum allowable containment leakage rate to the values assumed in the analyses. The finding was of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." (Section 1R21.4b.2)

B. <u>Licensee-Identified Violations</u>

None.

REPORT DETAILS

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 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 inspectible 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.

In addition, the inspectors reviewed several licensee audits and self-assessments to assess how effective licensee personnel were at self-identifying problems. The assessment was accomplished by comparing licensee-identified problems with problems that the inspectors identified during this inspection. The sample included a self-assessment in preparation for the inspection and selected assessments of the engineering design control program.

.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 Donald C. Cook Standardized Plant Analysis Risk Model, Revision 3P. In general, the selection was based upon the components and operator actions having a risk achievement worth of greater than 2.0. The operator actions selected for review included actions taken by operators both inside and outside of the control room during postulated accident scenarios.

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 failed performance test results, significant corrective action, 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.

.3 Component Design

a. Inspection Scope

The inspectors reviewed the UFSAR, 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 Code, the Institute of Electrical and Electronics Engineers Standards and the National Electric Manufacturers Association, to evaluate acceptability of the systems' design. 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 and licensee corrective action program documents (action requests--ARs). 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 Unit 1 and Unit 2 components were reviewed (18-inspection samples):

- Unit 1 Switchgear 1-T11A and Tie Breaker 1-T11A9: The inspectors reviewed electrical diagrams, specifications for the original and recently installed 4160 Volt (4.16kV) switchgear 1-T11A vacuum breakers, system short circuit calculations, protective relay trip setpoints, circuit breaker coordination, recently completed surveillance and relay calibration test results to assess the adequacy of the switchgear and tie breaker 1-T11A9 to meet the connected bus loading and short circuit duty requirements. The inspectors reviewed the loss of voltage protection on safety bus 1-T11A and reviewed the offsite voltage profile and the protocols between the plant operators and offsite power system operations to ensure that the loss of voltage relays would not actuate spuriously during certain offsite electrical system disturbances. The inspectors also reviewed the degraded voltage relay settings to ensure that adequate voltage was maintained at the terminals of the safety loads. The inspectors interviewed plant engineers to discuss the electrical distribution system configuration under all modes of operating conditions. The inspectors reviewed tie breaker 1-T11A9 closing and opening control circuits to verify that breaker tripping and closing logic was consistent with design basis description. The inspectors also reviewed recently completed plant preventive maintenance, surveillance testing and relay calibration test procedures to verify that calibrations were within the calculated limits. The inspectors performed a visual inspection of the 1-T11A switchgear to verify that breaker position indication lights, control switches, relay trip setpoints and equipment alignment were consistent with electrical calculations and drawings.

- Unit 1 Unit Auxiliary Transformer (UAT) TR1AB: The inspectors reviewed the UAT's vendor specifications, nameplate data, system one-line diagrams, protective relay setting calculations, 4.16kV buses 1-1A and 1-1B feeder cable ampacity calculations and loading requirements to determine the adequacy of the transformer to supply the 4.16kV Train B power demand requirements. The inspectors also performed independent relay setpoint calculations to verify the adequacy of electrical protection and that trip setpoints would not spuriously interfere with the transformer performing its designed function during energization, through-faults, and at maximum loading conditions. The relay settings review included the transformer overall differentials and the ground overcurrent relays. The inspectors also reviewed the adequacy of the transformer neutral grounding resistor rating. The inspectors reviewed the results of several recently completed transformer preventive maintenance and relay setpoint calibration tests to verify that the test results were within the allowable limits. Finally, the inspectors performed a visual inspection of the observable portions of Unit 1 UAT and the neutral grounding resistor to assess the installation configuration, material condition, and potential vulnerability to external hazards.

- Unit 1 Auxiliary 4.16kV Bus 1-1A and Supply Breaker 1-1A7: The inspectors assessed the components performance requirements through a selective review of one-line diagrams, load flow calculations, short circuit currents, protective relay trip setpoints, and system descriptions 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 also performed independent short circuit and relay trip setpoint calculations to verify adequacy of the ratings for the switchgear and that of the recently installed vacuum circuit breaker when the power supply to bus 1-1A is switched from the UAT to the reserve auxiliary transformer (RAT).

- Unit 1 Engineered Safety System (ESS) Train B: The inspectors reviewed calculations and drawings for supply breaker 1-T11A10 (4.16kV), transformer 1-TR11A (4.16kV/600V), supply breaker 1-11A11 (600V) and bus 1-11A (600V) to determine whether the loading of the components were within equipment ratings. The inspectors reviewed supply breaker 1-T11A10 control circuit voltage drop calculations to ensure adequate breaker control voltage was maintained. The inspectors reviewed the appropriateness of design assumptions and calculations related to short circuit currents, voltage and protective relay settings associated with transformer 1-TR11A and bus 1-11A. On a sample basis, the inspectors reviewed completed maintenance and functional validation test results to verify that transformer 1-TR11A was capable of supplying adequate power to bus 1-11A during normal and accident conditions. Cable routing drawings were reviewed to determine whether adequate separation was maintained between trains.

- Unit 1 Reactor Coolant Pump Supply Breaker 1-1B9: The inspectors assessed the component performance requirements through a review of electrical drawings and calculations describing the RCP motor power 4.16kV supply breaker, feeder and breaker control requirements during normal and degraded voltage operating conditions to evaluate the adequacy of the RCP 1-1B9 supply breaker, including the adequacy of the power feeder cable ampacity as-well-as that of the containment electrical penetrations. The protective relay setting calculations and coordination curves associated with the RCP motor circuit were also reviewed to determine the adequacy of relay trip settings. Specifically, the review included the relay setpoint calculations of the differential, phase and ground overcurrent relays. The containment electrical penetration ratings were reviewed to determine if they were adequate to withstand the available electrical and mechanical loadings during motor starting and in the event of electrical faults at the RCP motor terminals.

- Unit 1 Train B (1-AB) Emergency Diesel Generator (EDG): The inspectors reviewed the EDG loading calculations including voltage, frequency, current and loading sequence during loss of offsite power and loss of coolant accident. Short circuit calculations were reviewed to ensure that the ratings of the generator output breaker were adequate for the available short circuit duties. 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 determine the adequacy of the grounding scheme and ground overcurrent relay coordination. The electrical drawings and calculations that describe the generator output breaker 1-T11A11 control logic and interlocks were reviewed to determine whether the breaker opening and closing control circuits were consistent with design basis documents. The inspectors also reviewed electrical calculations and drawings to evaluate the capability of the 600V motor circuit center 1-ABD-A to supply the control and power requirements to the EDG's fuel oil transfer pump motor. The inspectors reviewed the Diesel Room Heat Up calculations, assessing the validity of assumptions, design inputs, and results. The assessment included fan flow rate margin and fan blade adjustments to maximize heat removal. The inspectors also interviewed the EDG System Engineer regarding the 2003 replacement of the jacket and lube oil coolers. Calculations addressing fuel consumption and tank volumes were also reviewed to verify adequate onsite fuel inventory. The inspectors performed a review of system normal operating procedures and surveillance test procedures to assess whether component operation and alignments were consistent with design and licensing bases assumptions.

- Unit 1 Heat Exchanger 1-HE-15W Component Cooling Water (CCW) Outlet Motor Operated Shutoff Valve 1-CMO-420: The inspectors reviewed the motor-operated valve (MOV) calculations including required thrust, weak link, and maximum differential pressure, to ensure the valve was capable of functioning under design conditions. Periodic Verification Diagnostic Test results were reviewed to verify acceptance criteria were met and performance degradation would be identified. Associated electrical calculations were reviewed to confirm that the design basis minimum voltage at the MOV motor terminals was consistent with the design inputs used in the MOV thrust calculations, and that the thermal overload heaters protecting the motors would not prematurely trip. The inspectors reviewed motor data, electrical control and schematic diagrams, degraded voltage calculations, thermal overload settings, voltage drop calculations, etc., to confirm that the motor operated shutoff valve 1-CMO-420 would have sufficient voltage and power available to perform its safety function at worst case degraded voltage and ambient conditions. The inspectors also reviewed operator actions requiring throttling of this valve to ensure that the thermal overload selected for this valve would not spuriously actuate due to frequent throttling. The inspectors also performed a review of system normal operating procedures to assess whether component operation and alignments were consistent with design and licensing bases assumptions.

- Unit 2 Heat Exchanger 2-HE-15W (Train B): The inspectors reviewed the CCW heat exchanger specifications and heat removal calculations to ensure that design basis heat removal requirements were met. The review included heat exchanger capacities, flow rates, fouling factors, and limiting service water temperatures.

- Unit 1 Train A CCW Pump 1-PP-10E: The inspectors reviewed the licensing and TS basis for the CCW pump. The inspectors reviewed the system hydraulic and net positive suction head (NPSH) analysis, the basis for the pump in-service test acceptance criteria, and a sample of actual in-service test results to verify the capability of the pump to perform its design function under accident conditions. In addition, the inspectors reviewed the pump control logic, and the system low pressure and low tank level setpoints to verify the availability of the pump. A sample of recent condition reports and operating procedures associated with the pump were also reviewed. The inspectors reviewed associated electrical drawings and calculations to confirm that the design basis minimum voltage at the pump motor terminals would be adequate for starting and running the motor under design basis conditions. The inspectors also reviewed the adequacy of the electrical power supply, feeder cable ampacity, T11D3 breaker opening and closing control logic and the protective relaying associated with the pump motor feeder circuit. The inspectors performed a review of system normal operating procedures and maintenance procedures, associated with use of the “spare” CCW pump, to assess whether component operation and alignments were consistent with design and licensing bases assumptions.

- Unit 1 250V direct current (dc) Transfer Cabinet 1-TDCD (Train A): The inspectors reviewed 250Vdc elementary and schematic diagrams, fuse ratings, 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 direct current equipment to perform their safety function.

- Unit 1 250 Vdc Plant Battery 1-BATT-CD and Busses (Train A): The inspectors reviewed 250Vdc battery and charger sizing calculations, TS surveillance requirements, the 7-day, 92-day, yearly and 60-month (load/discharge test) surveillances to confirm that the 250Vdc system health and sufficient capacity exists for the battery as well as the charger to perform their safety function. The inspectors also reviewed the ventilation calculations to verify that the temperature rise in the battery and charger rooms specifically during station black out and post-LOCA conditions would not adversely affect the performance of the battery and its charger.

- Unit 1 Condensate Storage Tank (CST) 1-TK-32: The inspectors reviewed the licensing and TS basis for the CST. The inspectors reviewed the analyses associated with the tank capacity and level setpoints, including potential vortexing concerns. The inspectors' review also included the temperature limits of the tank, the instrument uncertainty analyses, and the capacity of the tank during a station blackout event. These reviews verified the capability of the tank to perform its required function. A sample of recent condition reports and operating procedures associated with the tank were also reviewed.

- Unit 2 Turbine Driven Auxiliary Feedwater (TDAFW) Pump 2-PP-4: The inspectors reviewed the licensing and TS basis for the TDAFW pump. The inspectors reviewed the system hydraulic and NPSH analysis, the basis for the pump in-service test acceptance criteria, and a sample of actual in-service test results to verify the capability of the pump to perform its design function under accident conditions. In addition, the inspectors reviewed pump control logic to verify the availability of the pump. The inspectors reviewed the setpoints for the pump runout flow, low suction pressure, and suction strainer pressure differential to verify availability. In addition, the inspectors reviewed the design of the pump oil cooler, the design provisions for a high energy line break, the design of the essential service water (ESW) backup supply, and the performance of the pump during a station blackout event. A sample of recent condition reports and operating procedures associated with the pump were also reviewed. The inspectors reviewed the Unit 2 N-battery sizing and voltage drop calculations, battery discharge testing and routine TS surveillances to confirm that sufficient battery capacity and voltage existed to support the satisfactory operation of the TDAFW dc motor operated valves.

- Unit 1 Heat Exchanger 1-HE-15W ESW Outlet Motor Operated Shutoff Valve 1-WMO-737: The inspectors reviewed the MOV calculations including required thrust, weak link, and maximum differential pressure, to ensure the valve was capable of functioning under design conditions. Periodic Verification Diagnostic Test results were reviewed to verify acceptance criteria were met and performance degradation would be identified. Associated electrical calculations were reviewed to confirm that the design basis minimum voltage at the MOV motor terminals was consistent with the design inputs used in the MOV thrust calculations, and that the thermal overload heaters protecting the motors would not prematurely trip. The inspectors reviewed MOV motor data, electrical control and schematic diagrams, degraded voltage calculations, thermal overload settings, voltage drop calculations etc., to confirm that the MOV would have sufficient voltage and power available to perform its safety function at worst case degraded voltage and ambient conditions. The inspectors also reviewed operator actions requiring throttling of this valve to ensure that the thermal overload selected for this valve will not spuriously actuate due to frequent throttling. In addition, the inspectors performed a review of system normal operating procedures to assess whether component operation and alignments were consistent with design and licensing bases assumptions.

- Unit 1 ESW Pump 1-PP-7W (Train B) and Unit 2 ESW Pump 2-PP-7W (Train B): The inspectors reviewed piping and instrumentation diagrams, pump line up, pump capacities, and in-service testing data for the ESW pumps. Design calculations related to pump head, flow, NPSH were reviewed to ensure the pumps were capable of providing their accident mitigation function during all ambient conditions. Design change history was reviewed to assess potential component degradation and impact on design margins. The water supply (forebay) condition was also reviewed (recent sonar inspection report) to ensure that the water source design basis was maintained. The inspectors reviewed associated electrical drawings and calculations to confirm that the design basis minimum voltage at the pumps' motor terminals would be adequate for starting and running the motors under design basis conditions. The inspectors also reviewed the motors' nameplate data, the adequacy of the electrical power supply, feeder cable ampacity, T11A5 (Unit 1) and T21A5 (Unit 2) breaker ratings, opening and closing control logic including the protective relaying associated with the pumps' motor feeder circuits.

- Unit 1 Refueling Water Storage Tank (RWST) 1-TK-33: The inspectors reviewed the licensing and TS basis for the RWST. The inspectors reviewed the analyses and test data associated with the tank capacity and level setpoints, including potential vortexing concerns. The inspectors' review also included the temperature limits of the tank and the instrument uncertainty analyses. These reviews verified the capability of the tank to perform its required function. In addition, the inspectors reviewed issues associated with post-accident leakage from the emergency core cooling system (ECCS) into the tank to verify the potential impact on the control room and offsite dose analyses. A sample of condition reports and operating procedures associated with the tank were also reviewed.

- Unit 1 Residual Heat Removal Pump 1-PP-35W (Train B): The inspectors reviewed associated pump design calculations to ensure that design requirements were properly determined (e.g., pump pressures, flows and required NPSH) and that design basis requirements were correctly translated into test acceptance criteria. The inspectors also reviewed completed tests to ensure the pump's capability to perform its required design basis functions could be accomplished. The inspectors reviewed associated electrical drawings and calculations to confirm that the design basis minimum voltage at the pump motor terminals would be adequate for starting and running the motor under design basis conditions. The inspectors also reviewed the adequacy of the electrical power supply, feeder cable ampacity, T11A4 breaker ratings, opening and closing control logic and the protective relaying associated with the pump's motor feeder circuit.

b. Findings

The inspectors identified one finding of very low safety significance with one associated NCV.

Failure to Identify and Correct a Condition Adverse to Quality

Introduction: The inspectors identified a NCV of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action" having very low safety significance (Green) for failure to promptly identify and correct a condition adverse to quality regarding the Unit 1 Train A (1-CD) EDG's capacity rating until prompted by the NRC. Specifically, the 1-CD EDG's capacity rating could have been exceeded if the 1-CD EDG was allowed to run at the upper frequency band of 61.2 Hertz (Hz) as permitted by TS. As a result, the licensee performed corrective action calculations to assess the finding and on March 1, 2007, imposed an operational upper frequency limit of ≤60.5Hz on the station's Unit 1 EDGs. The primary cause of this violation was related to the cross-cutting area of problem identification and resolution.

Description: The inspectors reviewed Calculation 1-E-N-ELCP-4KV-001, "Unit 1, 4.16kV/600V Load Control Calculation," Revision 1, Change Sheet #6, dated January 22, 2002, and noted the 1-CD EDG's maximum loading was 3465 kW at a frequency of 60Hz for a "LOOP/LOCA with Containment Spray Initiated" scenario. With a design load rating of 3500 kW, this represented a very small margin (3500-3465/3500 or 1 percent) which should have prompted the licensee to evaluate the EDG assuming the maximum allowed frequency of 61.2 Hz. The inspectors determined that the 1-CD EDG would likely exceed its design load rating at the maximum upper frequency limit.

The inspectors discussed this concern with the licensee. The licensee indicated that the 1-CD EDG loading with respect to frequency was being evaluated as part of AR00124406, "Effects of EDG Frequency at 61.2Hz on Safety Related Loads," dated March 30, 2006. The AR documented that while replacing the internal assembly of the Unit 2 east charging pump, the licensee identified that the pump would develop 726 break horse power (BHP) versus 690 BHP of the pump motor when power to the pump motor was supplied by the EDG at 61.2Hz. Based on this discovery, the licensee evaluated all safety related pumps in both units, including the Unit 1 charging pumps and determined that only the Unit 2 east charging pump would exceed its motor BHP rating when supplied by the EDGs at 61.2Hz. Subsequently, the licensee performed a quick frequency analysis for Unit 2 and determined that at 60.5Hz, the Unit 2 east charging pump would not exceed its motor BHP rating and administratively limited the Unit 2 EDG operation to ≤60.5Hz as allowed by NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications That Are Insufficient to Assure Plant Safety." As part of the resolution of AR00124406, the licensee assigned a corrective action (CA #14) with a due date of February 2007, to review the Unit 1 and Unit 2 EDG load rating with respect to the upper frequency limit. The inspectors noted that the licensee's corrective action consisted of a review and re-analysis of the calculation to gain additional margin and to determine the highest frequency at which the Unit 1 EDGs could run safely without exceeding their design load rating. The licensee did not use the best information available at the time (i.e., Change Sheet #6 which showed very little margin) and therefore, did not place an administrative limit of 60Hz on the Unit 1 EDGs.

Until questioned by the inspectors on February 14, 2007, the licensee had not completed this evaluation. On February 15, 2007, the licensee issued AR00809059, “EDG Steady State Frequency Limits Contained in Technical Specifications,” and stated that under worst case loading conditions at 61.2 Hz, the 1-CD EDG would produce 3600kW, exceeding the EDG’s design load rating of 3500 kW by 100 kW. As a result, on March 1, 2007, the licensee imposed a Unit 1 operational upper frequency limit of ≤60.5Hz on the station’s 1-AB and 1-CD EDGs. The inspectors concluded the licensee allowed the 1-CD EDG to remain in a condition for a period of 11-months (from March 30, 2006, to March 1, 2007), where the 1-CD EDG design load rating could have been exceeded at the maximum TS allowed frequency of 61.2 Hz.

Analysis: The inspectors determined that the failure to promptly identify that the 1-CD EDG would exceed its capacity rating until prompted by the NRC constituted a performance deficiency and a finding. The 1-CD EDG’s capacity rating would have been exceeded if the 1-CD EDG was allowed to run at the upper frequency band of 61.2 Hz as allowed by TS. Furthermore, the inspectors determined that it was reasonably within the licensee’s ability to have identified this issue on January 22, 2002 with the approval of Change Sheet #6 to Calculation 1-E-N-ELCP-4KV-001 and on March 30, 2006, when the Unit 2 east charging pump issue was identified.

The inspectors determined that the finding was more than minor in accordance with IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Screening” because the finding was associated with the equipment performance (reliability) attribute of the Mitigating Systems Cornerstone and affected the cornerstone’s objective of ensuring the availability, reliability, and capability of the Unit 1 EDGs. Specifically, the 1-CD EDG would have exceeded its design load rating at the maximum TS allowed frequency of 61.2Hz. Without the evaluation and imposing an administrative limit of ≤60.5Hz, the licensee could not ensure that the 1-CD EDG would perform its safety related function.

The inspectors evaluated the finding using IMC 0609, “Significance Determination Process,” Appendix A, Phase 1 screening. The finding screened as Green because it was not a design issue, did not represent an actual loss of a system safety function, did not result in exceeding a TS allowed outage time, was not an actual loss of safety related equipment and did not affect external event mitigation.

This finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program because the licensee did not take appropriate corrective action to address the safety issue in a timely manner commensurate with its safety significance and complexity. Specifically, the licensee failed to identify the 1-CD EDG would exceed its capacity rating because the licensee’s corrective action to re-perform calculations was not implemented in a timely manner and allowed the condition to exist for 11 months.

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

Contrary to the above, from January 22, 2002, to March 1, 2007, the licensee failed to promptly identify and correct a condition adverse to quality related to the 1-CD EDG capacity rating until prompted by the NRC. Specifically, the licensee failed to identify that the 1-CD EDG's capacity rating of 3500 kW would have been exceeded by 100 kW had the 1-CD EDG operated at the upper frequency band of 61.2 Hz as permitted by TS. Because the finding was determined to be of very low safety significance, and the licensee entered the finding into their corrective action program as AR00809805, "Potential Criterion XVI Violation," this violation is being treated as a NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000315/2007002-01(DRS)).

.4 Operating Experience

a. Inspection Scope

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

- IN 91-56 — Potential Radioactive Leakage to Tank Vented to Atmosphere;
- IN 2005-21 — Plant Trip and Loss of Preferred AC Power from Inadequate Switchyard Maintenance;
- IN 2006-21 — Operating Experience Regarding Entrainment of Air into Emergency Core Cooling and Containment Spray Systems;
- IN 2006-22 — New Ultra-Low-Sulfur Diesel Fuel Oil Could Adversely Impact Diesel Engine Performance; and
- OpESS FY2007-01 — Review of Operating Experience Smart Sample (OpESS) FY2007-01, related to Information Notice 2006-20.

b. Findings

The inspectors identified two findings of very low safety significance with two associated NCVs.

1. **Failure to Correct Inadequate Safety Analysis Dose Calculations**

Introduction: The inspectors identified a NCV of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action” having very low safety significance (Green) for failure to promptly identify and correct a condition adverse to quality regarding inadequate safety analysis dose calculations. Specifically, the licensee failed to address the aggregate effect of various nonconforming conditions on containment leakage rates for offsite dose and control room calculations to ensure that accurate and adequate margin remained available for offsite dose analyses and control room habitability. The finding was entered into the licensee’s corrective action program and an operability determination evaluation (ODE) was initiated during the inspection. The primary cause of this violation was related to the cross-cutting area of problem identification and resolution.

Description: The inspectors reviewed the licensee’s evaluation for NRC Information Notice (IN) 91-56, “Potential Radioactive Leakage to Tank Vented to Atmosphere” during this inspection. The IN addressed the potential radiological consequences of fluid leakage from the ECCS into the vented RWST under post-accident conditions and the leak testing of system isolation valves in that leakage path. As a result of this review, the inspectors questioned the impact of the potential leakage paths on both the offsite and control room dose analyses. Several condition reports and analyses addressed the impact of ECCS leakage and other concerns on the offsite and control room dose analyses for various postulated accidents, including the following:

- On June 29, 1998, Condition Report 98-03076 was initiated to identify that although UFSAR Chapter 6 and Chapter 14 analyses were performed for up to 10-gpm ECCS leakage, these analyses were not introduced into the licensing basis with a supporting 10 CFR 50.59 evaluation. The corrective action associated with the UFSAR change was closed on April 2, 2004, without an appropriate resolution.

- On February 19, 1999, Condition Report 99-03135 was initiated to identify that the 1997 update for the Unit 2 UFSAR did not include a dose contribution from ECCS leakage. The condition report documented a basis for operability that was based on calculation RD-94-01 for ECCS leakage and the Unit 2 UFSAR for other contributors to dose. On October 20, 1999, the corrective action associated with the ECCS leakage was closed based on the incorporation of 0.2-gpm ECCS leakage into control room and offsite dose calculations using the alternative source term methodology.

- On November 11, 1999, Calculation CN-CRA-99-78, “D. C. Cook (AEP/AMP) TID-14844 Source Term LOCA Radiation Dose Analyses” was approved. This calculation was an operability type evaluation that was intended to support restart and it included a 0.2-gpm ECCS leakage assumption. A new offsite dose accident analysis using the alternative source term methodology was intended to be submitted to the NRC for review and approval after restart. The offsite dose accident analysis using the alternative source term methodology was not submitted due to other technical issues.

- On December 8, 2005, Condition Report 05342040 (AR00119229) was initiated to identify that several corrective actions associated with UFSAR offsite and control room dose accident analyses were inappropriately closed. The condition report documented a basis for operability that was based on Calculation RD-94-01 for ECCS leakage and the more conservative dose results in the Unit 1 UFSAR. There was no ODE implemented as a result of this condition report. This condition report was open at the time of the inspection.

- On June 12, 2006, Condition Report 06163008 (AR 00127854) was initiated to identify that non-conservative values were used for containment free air volumes in previous calculations. This condition report documented a new basis for operability, based on calculation RD-94-01 for ECCS leakage and calculation CN-CRA-99-78, Revision 2. There was no ODE implemented as a result of this condition report. This condition report was open at the time of the inspection.

The inspectors noted that the licensee had not assessed the aggregate effect these conditions had on containment leakage rates for offsite dose and control room calculations. The inspectors were concerned that without this aggregate review, the licensee could not ensure that the remaining margin for the offsite dose and control room habitability analyses was adequate. In addition, the inspectors questioned the lack of established compensatory measures for TS to ensure that the maximum allowable containment leakage rate and maximum allowable ECCS leakage rates were maintained below the values assumed in the operability analyses as outlined in NRC Administrative Letter 98-10 (see Section 1R21.4b.2).

In response, the licensee initiated an “Operability Determination Evaluation (ODE) for the Aggregate Effects of Non-Conservative Values Impacting Control Room Habitability and Offsite Dose Analyses” (ODE for AR00809145) during this inspection. The ODE addressed the known non-conservative values used in the various dose analyses. In addition, as discussed in Section 1R21.4b.2 of this report, the licensee initiated compensatory measures on February 28, 2007, to ensure that the maximum allowable containment leakage rate and maximum allowable ECCS leakage rate were maintained below the values assumed in the operability analyses.

<u>Analysis</u>: The inspectors determined that the licensee’s failure to perform an operability evaluation to determine the aggregate effect several discrepant conditions had on containment leakage rates for offsite dose and control room habitability calculations constituted a performance deficiency and a finding. Furthermore, the inspectors determined that it was reasonably within the licensee’s ability to have corrected this finding as indicated by several condition reports on the subject.

The inspectors determined the finding was more than minor in accordance with IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Screening” because the finding was associated with the configuration control (containment design parameters maintained) attribute of the Barrier Integrity Cornerstone and affected the cornerstone objective of maintaining the functionality of containment. Specifically, the licensee did not verify the capability of containment to maintain the offsite and control room dose within required limits under post-accident conditions to the values assumed in the analyses.

The inspectors evaluated the finding using IMC 0609, "Significance Determination Process," Appendix A, Phase 1 screening. The finding screened as Green because the inspectors answered no to the three questions in the Containment Barriers Cornerstone Column. Specifically, the finding did not represent an actual open pathway in the physical integrity of reactor containment.

This finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program because the licensee did not thoroughly evaluate the condition such that, corrective actions addressed the causes and extent of conditions, and that effectiveness reviews of those corrective actions ensured that the problems were resolved in a timely manner. Specifically, the licensee failed to perform an operability evaluation to assess the aggregate effect several known discrepant conditions because the licensee failed to thoroughly evaluate several identified conditions, inappropriately closed corrective actions and did not recognize the potential impact on the calculated margins in offsite dose and control room habitability analyses.

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

Contrary to the above, from February 19, 1999, to February 28, 2007, the licensee failed to promptly identify and correct conditions adverse to quality regarding inadequate safety analysis dose calculations, which supported operability of containment.

Specifically,

a. On December 8, 2005, the licensee identified that previously on April 2, 2004, Condition Report 98-03076 was inappropriately closed without resolving the issue. The licensee failed to provide a basis (a 10 CFR 50.59 safety evaluation) for a UFSAR change which allowed up to 10-gpm of ECCS leakage either into the auxiliary building or back to the RWST under post-accident conditions. As of March 2, 2007, the licensee failed to take prompt corrective actions, in that, this evaluation (or an evaluation supporting a different leakage rate) had not been completed.

b. On February 19, 1999, the licensee identified that the 1997 Unit 2 UFSAR update did not include a dose contribution from ECCS leakage. Subsequent condition reports identified other deficiencies in the analyses and operability type evaluations were performed to address individual issues. However, the licensee did not identify the aggregate effect various nonconforming conditions had on containment leakage rates for offsite dose and control room calculations. This evaluation was not completed and compensatory actions were not implemented until February 28, 2007.

Because the finding was determined to be of very low safety significance, and because the licensee subsequently entered the finding into their corrective action program as AR00809806, "Potential Criterion XVI Violation," dated March 1, 2007, this violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000315/2007002-02(DRS); 05000316/2007002-02(DRS)).

2. **Failure to Maintain Previously Imposed Compensatory Measures**

Introduction: The inspectors identified a NCV of 10 CFR Part 50.36, “Technical Specifications” having very low safety significance (Green) regarding the failure to maintain previously imposed compensatory measures for inadequate safety analysis dose calculations. Specifically, the licensee previously imposed administrative limits (i.e., compensatory measures) as a result of non-conforming UFSAR offsite and control room dose analyses. However, the licensee stopped those actions and operated between April 25, 2003, and February 28, 2007, with assumed containment leakage values that were not bounded by the licensee’s TS 5.5.14, “Containment Leakage Rate Testing Program.” Once identified by the inspectors, the licensee re-imposed the required compensatory measures.

Description: As discussed in Section 1R21.4b.1 of this report, the licensee initiated numerous condition reports and analyses which addressed non-conforming UFSAR offsite and control room dose analyses for postulated accidents. As a result, the inspectors were concerned that adequate compensatory measures had not been established to address the impact of the non-conforming conditions on containment leakage. The inspectors identified the following:

- On May 20, 2000, a Unit 2 administrative limit was imposed by the licensee’s Administrative Technical Requirements (ATR) Manual Number 2-CNTMT-1, “Containment Systems - Containment Leakage,” on Unit 2's Procedure 2-EHP-4030-001-001, “Unit 2 Primary Containment Leak Rate Running Total.” The administrative limit was imposed based on Condition Report P-00-01069, “Impact Assessment for Westinghouse Letter Report AEP-00-004 Identified Changes to Plant Procedures,” dated January 20, 2000. The administrative limit imposed a maximum containment leakage rate which was half the TS allowed value.

- On November 6, 2000, a Unit 1 administrative limit was imposed by the licensee’s ATR Manual Number 1-CNTMT-1, “Containment Systems - Containment Leakage,” on Unit 1's Procedure 1-EHP-4030-001-002, “Unit 1 Primary Containment Leak Rate Running Total.” The administrative limit was imposed based on Condition Report P-00-01069, “Impact Assessment for Westinghouse Letter Report AEP-00-004 Identified Changes to Plant Procedures,” dated January 20, 2000. The administrative limit imposed a maximum containment leakage rate, which was half the TS allowed value.

- On April 25, 2003, the Unit 2 administrative limit imposed by ATR Manual Number 2-CNTMT-1 was removed. The licensee stated that the administrative limit was removed based on the NRC’s approval of Unit 2's TS Amendment 252 for alternate source term which eliminated the need for the administrative restriction.

- On September 30, 2003, the Unit 1 administrative limit imposed by ATR Manual Number 1-CNTMT-1 was removed. The licensee stated that the administrative limit was removed based on the NRC's approval of Unit 1's TS Amendment 271 for alternate source term which eliminated the need for the administrative restriction.

- On December 8, 2005, Condition Report 05342040 (AR00119229), was initiated to document that several corrective actions associated with the offsite and control room dose analysis were inappropriately closed and relied upon referenced calculations as a basis for operability. Among the assumptions made in the referenced calculations were maximum containment leakage rates which were half the TS 5.5.14 allowed leakage rate. This condition report was still open at the time of this inspection.

- On June 12, 2006, Condition Report 06163008 (AR00127854) was initiated to identify that non-conservative values were used for containment free air volumes in previous calculations. To provide a reasonable assurance of operability associated with this condition report, credit was taken for containment leak rates which were half of the TS 5.5.14 value, as supported by actual leak test results. Administrative limits were not imposed to ensure that the assumed leakage rates were not exceeded. This condition report was still open at the time of this inspection.

As a result of this information, the inspectors questioned the lack of compensatory measures established to ensure that containment leakage rates assumed in these operability analyses would not be exceeded. At the time of this inspection, neither TS 5.5.14 nor plant procedures included administrative leakage limits that bounded the analyses.

In response, the licensee initiated compensatory measures on February 28, 2007, to ensure that the maximum allowable containment leakage rate and maximum allowable ECCS leakage rate were maintained below the values assumed in the operability analyses.

<u>Analysis</u>: The inspectors determined that the failure to maintain previously imposed compensatory measures to ensure that containment leakage rates assumed in various operability analyses would not be exceeded constituted a performance deficiency and a finding. Without the administrative limits (i.e., compensatory measures) maintained and/or imposed, the station could have operated up to the non-conservative and/or deficient TS values. The inspectors further determined that the finding was within the licensee's ability to foresee and correct, and that it could have been prevented had the licensee re-imposed the required compensatory measures in 2005.

The inspectors determined the finding was more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening" because the finding was associated with the configuration control (containment design parameters maintained) attribute of the Barrier Integrity Cornerstone and affected the cornerstone's objective of maintaining the functionality of containment. Specifically, the licensee did not re-impose compensatory measures to limit the maximum allowable containment leakage rate to the values assumed in the analyses.

The inspectors evaluated the finding using IMC 0609, "Significance Determination Process," Appendix A, Phase 1 screening. The finding screened as Green because the inspectors answered no to the three questions in the Containment Barriers Cornerstone Column. Specifically, the finding did not represent an actual open pathway in the physical integrity of reactor containment. This determination was based on an ODE performed by the licensee during the inspection, and on the licensee's review of actual ECCS and containment leakage rate test data.

This finding has a cross-cutting aspect in the area of human performance associated with the decision-making component because the licensee failed to communicate decisions and the basis for those decisions to personnel who had the need to know the information in order to perform work safely. Specifically, the licensee failed to maintain previously imposed compensatory measures because the need to limit the maximum allowable containment leakage rate was not communicated to station personnel.

Enforcement: Title 10 of the CFR Part 50.36, "Technical Specifications," requires, in part, that each TS limiting condition for operation specify, at a minimum, the lowest functional capability or performance level of equipment required for the safe operation of the facility.

TS 5.5.14 c. states that the maximum allowable containment leakage rate, La, at the calculated peak containment internal pressure stated in TS 5.5.14.b, shall be 0.25 percent of containment air weight per day.

Contrary to the above, from April 25, 2003, to February 28, 2007, the licensee operated Unit 1 and Unit 2 without restriction to the maximum allowable containment leakage rate defined by TS 5.5.14.c. However, as a basis for containment operability, the licensee relied upon calculations which assumed half the TS 5.5.14 allowed leakage rate as documented in at least two condition reports (Condition Report 05342040 and 06163008). The licensee failed to recognize the TS were non-conservative and did not imposed administrative limits to ensure that the assumed leakage rates were not exceeded until February 28, 2007. Because the finding was determined to be of very low safety significance and because the licensee subsequently entered the finding into their corrective action program as AR00809878, "Potential Violation of 10 CFR Part 50.36," dated March 1, 2007, this violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000315/2007002-03(DRS); 05000316/2007002-03(DRS)).

.5 Modifications

a. Inspection Scope

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

- 1-DCP-4894 Modify "Standby Readiness" Position of TDAFW Pump Discharge Valves (1-FMO 211, 221, 231 and 241);

- 1-MOD-35003 4kV Motor Current Transformer (CT) Saturation Resolution;
- 1-MOD-55348 1-T11A9, Install New 4kV Breaker;
- 12-LDCP-5260 Essential Service Water Pump Upgrades for Reliability; and
- EC-MOD-ECC47442 MCC Molded Case Circuit Breaker Replacement.

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 four risk significant, time critical operator actions (four samples). 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:

- Actions, as directed by the station's emergency operating procedures, to transfer to cold leg recirculation when the RWST level reaches 30 percent;
- Actions, as directed by the station's emergency and abnormal operating procedures, to cross-tie the chemical and volume control system (CVCS) to the unit affected unit's CVCS upon a loss of the CCW system;
- Actions, as directed by the station's emergency and abnormal operating procedures, to restore reactor coolant system inventory following recovery from a loss of CCW; and
- Actions, as directed by the station's emergency and abnormal operating procedures, to mitigate and recover from a station blackout.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA2 Problem Identification and Resolution

.1 Review of Condition Reports

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. The specific corrective action documents that were reviewed by the inspectors are listed in the attachment to this report.

b. Findings

Two findings of very low safety significance were identified during this review and are discussed in Sections IR21.3b and IR21.4b.1

4OA6 Meetings, Including Exits

Exit Meeting Summary

- The inspectors presented the inspection results to Mr. Mark Peifer and other members of licensee management at the conclusion of the inspection on March 2, 2007. Proprietary information was reviewed during the inspection and was handled in accordance with NRC policy.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
J. Anderson, Program Owner
D. Badgero, Operations
B. Bradley, System Engineering
J. Chong, Design Engineer - I & C
R. Crane, Regulatory Affairs
D. Fadel, Design Engineering Director
A. Feliciano, Design Engineering - Mechanical
T. Fisher, System Engineering
J. Gebbie, Plant Engineering Director
R. Gray, Design Engineering - Mechanical
R. Hackman, Regulatory Affairs
J. Jensen, Site Vice President
G. Kilpatrick, Design Engineering - Electrical
J. Kovarik, Design Engineer Manager - I & C
M. Ma, Probability Risk Assessment
S. Macey, Instrument and Control Technician
M. Madigan, Design Engineering - Electrical
E. Malle, System Engineering
B. Mammoser, Design Engineering, Mechanical
P. Mangan, Configuration Control Manager
R. Meister, Regulatory Affairs
T. Mottl, Administration
M. Peifer, Site Support Vice President
J. Phelan, Design Engineering - Electrical
M. Radocha, System Engineering
P. Schoepf, Design Engineering Manager
Y. Shen, PRA Supervisor
S. Simpson, Regulatory Affairs Manager
G. Smith, Design Engineering - Mechanical Contractor
C. Vanderzwaag, System Engineering
W. Wah, System Engineering
L. Weber, Plant Manager
J. Wicks, Operations Support Manager
V. Woods, Performance Assurance Manager

NRC
B. Kemker, Senior Resident Inspector
J. Lennartz, Resident Inspector
C. Lipa, RIII DRP Branch 4, Chief
D. Passehl, RIII Senior Reactor Analyst
A. Stone, RIII DRS Engineering Branch 2, Chief

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Opened and Closed

05000315/2007002-01(DRS); 05000316/2007002-01(DRS)	NCV	Failure to Identify and Correct a Condition Adverse to Quality (Section 1R21.3b.1)
05000315/2007002-02(DRS); 05000316/2007002-02(DRS)	NCV	Failure to Correct Inadequate Safety Analysis Dose Calculations (Section 1R21.4b.1)
05000315/2007002-03(DRS); 05000316/2007002-03(DRS)	NCV	Failure to Maintain Previously Imposed Compensatory Measures (Section 1R21.4b.2)

Closed and Discussed

None

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	Date or Revision
1-2-F2-01	Low Suction Press Alarm SP AFW Pumps	8
1-2-UNC-338 CALC3	CST Level Loop Uncertainty Calc	2
1-2-UNC-338 CALC4	CST Level Loop Uncertainty Calc	0
1-2-UNC-339 CALC3	SP Calc for RWST Level Alarms, RHR Pump Trip Interlock and Operation Points	0
1-E-N-ELCP-250-001	U1 250Vdc System Coordination Study	0
1-E-N-ELCP-250-006	U1 250Vdc Battery and Charger Sizing Calc	0
1-E-N-ELCP-4KV-001	4kV/600V Load Control Calc	1
1-E-N-ELCP-600-003	600V Motor Control Control Voltage Drops	1
1-E-N-PROT-PEN-001	Electrical Containment Penetration Protection	0
1-E-N-PROT-RLY-003	Degraded Grid and LOV Relay Setting Calc	0
1-E-N-PROT-TOL-001	600V System Continuous Duty SR Motor Thermal Overload Heater Selection	5
2-E-N-ELCP-250-008	250Vdc Batt 2N System Analysis	0
12-E-N-SBO-COP-001	Station Blackout Required Coping Duration	0
DIT-B-00174-00	RWST Vortexing Values	0
DIT-B-792-00	Provide Input to Support Selection of AFW Strainer Press Alarm SP	0
DIT-B-01061-10	EOP Operator Action Times from Accident Analyses	0
DIT-B-01150-00	Determine Alarm SPs for AFW Pump Suction Strainers Using New 1/8” Mesh Size Strainer Baskets	0
DIT-B-01543-00	D. C. Cook (AEP/AMP) TID-14844 Source Term LOCA Radiation Dose Analysis	2
ENSM990305AF	Determine CCW Heat Exchanger UA during Recirculation Operation	0
MD-01-AFW-004-N	TDAFW Pump FMO Vlv Position Determination	1
MD-01-CCW-024-N	Torque Setup for 1-CMO-420	1
MD-01-ECCS-004-N	U1 ECCS Pumps NPSH Analysis	2
MD-01-ECCS-041-N	U1 ECCS Pumps NPSH Analysis	2
MD-01-ESW-084-N	Torque Setup Calc for 1-WMO-737	1
MD-02-CCW-011-N	U2 CCW Sys Analysis to Relate Accident, Testing and Min Operability Requirements	0
MD-02-ECCS-005-N	U2 ECCS Pumps NPSH Analysis	1
MD-02-ESW-077-N	U2 ESW Sys Analysis for as Left March 29, 2000 Flow Balance Conditions to Determine Allowable Min Operability Requirements	1

CALCULATIONS

Number	Description or Title	Date or Revision
MD-12-AFW-001-N	AFW Sys Design Basis Analysis	1
MD-12-AFW-046-N	Actuator Capability Calc for East MDAFW and TDAFW Pump Test Valves and Emergency Leakoff Valves 1(2)-FRV-255, 256, 257, and 258	1
MD-12-AFW-047-N	Actuator Capability Calc for West MDAFW Pump Test Valves and Emergency Leakoff Valves 1(2)-FRV-245 and 247	1
MD-12-CCW-005-N	Valve and Pump Seal Leakage from Misc Trains of the CCW Sys	0
MD-12-CCW-809-N	GL89-10 Program Press, Temp, Flows for CCW Sys Valves	1
MD-12-CST-001-N	CST Usable Volume and Vortexing	0
MD-12-CST-001-N-ADD	CST Usable Volume and Vortexing	0
MD-12-CST-002-N	Operation of AFW Sys Using CST of Other Unit	0
MD-12-CTS-012-N	RWST Level	0
MD-12-DG-004-N	Diesel Fuel Oil Consumption Rate, Verification of DG Fuel Oil Storage and Day Tank Volumes, and Transfer Pump and Diesel Exhaust Line Sizing	2
MD-12-ECCS-007-N	Leakage through a CCP Mini-Flow during the Recirculation Phase of a LOCA	0
MD-12-ESW-076-N	ESW Pump NPSH Available and Submergence	0
MD-12-HV-020-N	Heat Gain and Max/Min Temp Determination for the EDG Rooms	3
MD-12-HV-021-N	Switchgear and Batt Rooms Heat Gain Calc	4
MD-12-RWST-001-N	Max dp for RWST Vent Path	2
MD-12-RWST-001-N-ADD	Max dp for RWST Vent Path	0
MD-12-RWST-002-N	RWST Vortex Model Test Results Eval	0
MD-CCW-812-N	EPRI PPM Eval of 1/2 - CMO-420	0
PS-4KVP-003	Grd Relay Settings for 4kV ESS and BOP Buses	0
PS-4KVP-005	Unit and Reserve Feed Phase OC Relay Setting	0
PS-4KVP-006	4kV BOP Motor Electrical Protection	2
PS-4KVP-014	4kV Breaker Cooling Fan Start Relay Settings	0
PS-EDGP-002	EDG Grd Relay	0
RD-94-01	Offsite Dose Due to ECCS Leakage	0
SD-990825-013	Seismic/Weak Link Torque Calc for MOVs	6
TH-99-13	CST Inventory	0

CORRECTIVE ACTION PROGRAM DOCUMENTS ISSUED DURING INSPECTION

Number	Description or Title	Date or Revision
AR00808316	Errors in UFSAR Table 14.4.2-1A (eSAT 07030015)	January 30, 2007
AR00808367	CDBI Question Regarding DB-12-AFWS R2, Sect 5.1.9.3 (eSAT 07030068)	January 30, 2007
AR00808997	Work Order Detail Planning Needs Enhancement (eSAT 07045021)	February 14, 2007

CORRECTIVE ACTION PROGRAM DOCUMENTS ISSUED DURING INSPECTION

Number	Description or Title	Date or Revision
AR00809005	CDBI Identified Several Enhancements to ECA-0.0 (eSAT 07045057)	February 14, 2007
AR00809059	EDG Steady State Frequency Limits Contained in Technical Specifications (eSAT 07046078)	February 15, 2007
AR00809128	2-PP-7W Pump Baseline Sheet Typographical Error (eSAT 07046027)	February 15, 2007
AR00809145	Inadequate Calc Related to Offsite Dose from ECCS Leakage (eSAT 07047028)	February 16, 2007
AR00809194	U2 UFSAR, Sect 14.3.5, Offsite Dose Accident Analyses, Should Not Be Used as Stated in U1 UFSAR, Sect 14.3.5 (eSAT 07047029)	February 16, 2007
AR00809195	CDBI Item EDG Freq/Loading Extent of Condition Concern (eSAT 07047033)	February 16, 2007
AR00809213	LOOP LOCA Procedures Do Not Doc Return to Ready to Load (eSAT 07047056 closed to AR805607 and AR805608)	February 16, 2007
AR00809215	CDBI Identified Enhancements to LOOP/LOCA Procedure (eSAT 07047030)	February 16, 2007
AR00809656	RCP Motor Nameplate Anomaly (eSAT 07058011)	February 27, 2007
AR00809659	U1 UFSAR Sect 14.3.5.20.2.2, Effectiveness of Spray Sys for Activity Removal (eSAT 07057023)	February 26, 2007
AR00809748	Breaker Operating Time (eSAT 07059049)	February 28, 2007
AR00809805	Potential Criterion XVI Violation (eSAT 07060059)	March 1, 2007
AR00809806	Potential Criterion XVI Violation (eSAT 07060060)	March 1, 2007
AR00809878	Potential Violation of 10 CFR 50.36 (eSAT 07060061)	March 1, 2007

CORRECTIVE ACTION PROGRAM DOCUMENTS ISSUED PRIOR TO INSPECTION

Number	Description or Title	Date or Revision
CR98-03076	A 10 CFR50.59 Eval CR Dose Analysis with Operational Aspects; Loss of Inventory from the Sump Needs to Be Performed	June 29, 1998
CR99-03135	UFSAR Chap 14 Dose Analysis of the Offsite Consequences of a LOCA Did Not Include Contribution of the Recirculation Fluid in the Aux Bldg That Bypassed Containment	February 19, 1999
CR99-04235	CCW Surge Tank Vacuum Breaker Check Valve CCW-215 Is Not Currently in the IST Program	March 2, 1999
CR99-29181	Operability Evaluation Should Be Developed for the U1 CR Ventilation Sys	December 15, 1999
CR P-00-01069	Impact Assessment for Westinghouse Letter Report AEP-00-004 Identified Changes to Plant Procedures	January 20, 2000
CR01244041	2-PP-7W Differential Pressure in the Alert Range	August 31, 2001
CR03020009	NRC IN 2003-02, Recent Experience with RCS Leakage and Boric Acid Corrosion	January 20, 2003
CR03158021	1E and 2E ESW Pump Bearing Damage	June 7, 2003

CORRECTIVE ACTION PROGRAM DOCUMENTS ISSUED PRIOR TO INSPECTION

Number	Description or Title	Date or Revision
CR03161056	NRC Bulletin 2003-01, Potential Impact of Debris Blockage on Emergency Sump Recirculation at PWRs	June 10, 2003
CR03234018	NRC Bulletin 2003-02, Leakage From RX Press Vessel Lower Head Penetrations and RX Coolant Boundary Integrity	August 22, 2003
CR04022018	NRC IN 2004-01, AFW Pump Recirculation Line Orifice Fouling Potential Common Cause Failure	January 22, 2004
CR04160070 / AR00092892	Determine a Better Method of External Valve Position Indication	June 8, 2004
CR05231027	NRC IN 2005-23, Vibration-Induced Degradation of Butterfly Valves	August 19, 2005
CR05266069	Changes Implemented to The U1 TDAFW Pump Discharge FMO Valves Standby Readiness Position	September 23, 2005
CR05342040 / AR00119229	14 CRAs Involving CAQS Related to UFSAR Offsite Dose Accident Analyses Were Closed with No Actions Taken and No Basis Provided for Doing So, Leaving the CAQS Without Adequate Resolution	December 8, 2005
CR06017027	Freedom Series Starter Coils Do Not Have the Same Inrush and Hold Characteristics as the Original Equipment (Citation Series)	January 11, 2006
CR06039045	NRC IN 2006-03, Motor Starter Failures Due to Mech Interlock Binding	February 8, 2006
CR06067042	NRC IN 2006-05, Possible Defect in Bussmann KWN-R and KTN-R Fuses	March 8, 2006
AR00120662	Engineering Evals Replacing Original Cutler-Hammer Starters with Freedom Series Cutler-Hammer Starters Reveal Issues	January 17, 2006
AR00122570	600Vac MCC Breaker Replacement and 4.16kV Vacuum Breaker Replacement	
AR00124406	Effects of EDG Freq at 61.2Hz on SR Loads	March 30, 2006
CR06163008 / AR00127854	Non-Conservative Values for Containment Free Air Volumes are Used in Offsite Dose Accident Analysis in AEP-00-004 and DIT-B-00356-01	June 12, 2006
AR00804911	NRC IN 2006-22, New Ultra Low Sulfur Diesel	October 31, 2006

DRAWINGS

Number	Description or Title	Date or Revision
1-1313	General Arrangement and Details Containment Penetrations Power Circuiting and Conductor Information	23
5-030-02-008-001	02008 BCF Exchanger - 1 Pass	5
12-1237	Electric Heat Tracing for RWST	15
12-5684	Ventilation of AFW Pump Room - Sheet 1	8
OP-1-2001	Main Aux One-Line Diagram ESS Bus A and B (Train B)	77
OP-1-2002	Main Aux One-Line Diagram ESS Bus C and D (Train A)	62
OP-1-5129	Flow Diagram CVCS RX Letdown and Charging U1	55
OP-1-5129A	Flow Diagram CVCS RX Letdown and Charging U1	34

DRAWINGS

Number	Description or Title	Date or Revision
OP-1-5142	Flow Diagram Emergency Core Cooling (SIS)	43
OP-1-5143	Flow Diagram Emergency Core Cooling (RHR) U1	67
OP-1-5144	Flow Diagram Containment Spray U1	41
OP-1-12001	Main Aux One-Line Diagram Bus A and B ESS (Train “B”)	77
OP-1-12002	Main Aux One-Line Diagram Bus C and D ESS (Train “A”)	62
OP-1-12003	Train A, B, N and BOP 250Vdc One-Line Diagram	31
OP-1-98013	Diesel Generator 1AB and Auxiliaries Elementary Diagram	34
OP-1-98041	4kV Aux Transformers 1AB and 101AB Elem Diagram	29
OP-1-98043	Diesel Generator 1AB A.C.B. Elementary Diagram	50
OP-1-98045	4kV/600V Aux Transformers 11A and11C Elem Diagram	26
OP-1-98046	4kV 600V Aux Transformers 11B and 11D Elem Diagram	28
OP-1-98284	Emergency Core Cooling (RHR) Elem Diagram, Sheet 1	48
OP-2-5129	Flow Diagram CVCS RX Letdown and Charging U2	49
OP-2-5129A	Flow Diagram CVCS RX Letdown and Charging U2	34
OP-2-98033	DG 2CD Excitation and Regulation and Misc Elem Diagram	43
OP-2-98035	Diesel Generator 2CD Control Elementary Diagram	33
OP-2-98044	Diesel Generator 2CD A.C.B. Elementary Diagram	48
OP-2-98215	TDAFW Supply Sys Elementary Diagram, Sheet 1	56
OP-2-98216	TDAFW Supply Sys Elementary Diagram, Sheet 2	21
OP-12-5148B	Flow Diagram Misc SR Ventilation Sys	17
OP-12-12007	Misc Aux Sys One-Line Offsite Plant Services, Sheet 2	10
SOD-00800-001	Emergency Core Cooling Injection Phase	4
SOD-00800-002	Emergency Core Cooling Recirculation Phase	3
SOD-01600-001	CCW Sys	3a

EVALUATIONS (50.59 and ODEs)

Number	Description or Title	Date or Revision
2000-2234-01	Modify Standby Readiness Position of TDAFW Pump Discharge Valves(1-FMO-211, -221, -231, and -241)	December 5, 2000
2005-0469-00	Revision to U1 EOP OHP 4023 ECA 0.0, Step 4	September 29, 2005
AR00809145-12	ODE for Aggregate Effects of Non-Conservative Values Impacting CR Habitability and Offsite Dose Analyses	March 2, 2007

MODIFICATIONS

Number	Description or Title	Date or Revision
1-DCP-4894	Modify Standby Readiness Position of TDAFW Pump Discharge Valves (1-FMO-211,-221,-231,-241)	0
1-MOD-35003	4kV Motor Current Transformer Saturation Resolution	0
1-MOD-55348	Installation of New 4kV Breakers	0
12-LDCP-5260	ESW Pump Upgrades for Reliability	0
DC-12-073	Add Flow Controllers to Limit MD and TD Pump Flow	May 7, 1973
DC-12-2912	Remove Auto Pump Trip on Low Suction Press in AFW Sys	January 2, 1987

MODIFICATIONS

Number	Description or Title	Date or Revision
EC-MOD-ECC-47442	MCC Molded Case Circuit Breaker Replacement	0
ICP-00559	AFW Strainer Modification	May 13, 2000

PROCEDURES

Number	Description or Title	Date or Revision
1/2-OHP-4021-016-001	Filling and Venting the CCW Sys	18
1/2-OHP-4021-016-002	Interchanging Spare CCW Pump with East or West CCW Pump	13/9
1/2-OHP-4021-082-001	4kV Buses Power Source Transfer and De-Energizing and Re-Energizing a Safeguards Bus	17/12
1/2-OHP-4022-016-001	Malfunction of CCW Sys	5
1/2-OHP-4022-016-004	Loss of CCW	11/15
1/2-OHP-4022-019-001	ESW Sys Loss/Rupture	6
1/2-OHP-4023-E-0	Reactor Trip or Safety Injection	31/32
1/2-OHP-4023-E-1	Loss of Reactor or Secondary Coolant	14/16
1/2-OHP-4023-ECA-0.0	Loss of All AC Power	20/18
1/2-OHP-4023-ECA-0.1	Loss of All AC Pwr Recovery W/O SI Required	12
1/2-OHP-4023-ECA-0.2	Loss of All AC Pwr Recovery With SI Required	11/13
1/2-OHP-4023-ECA-1.1	Loss of Emergency Coolant Recirculation	10/11
1/2-OHP-4023-ECA-1.3	Sump Blockage Control Room Procedure	0
1/2-OHP-4023-ES-1.3	Transfer to Cold Leg Recirculation	10
1/2-OHP-4023-ES-1.4	Transfer to Hot Leg Recirculation	4
1/2-OHP-4023-SUP-002	Restoration of Reserve Power to 4kV Buses	6
1/2-OHP-4023-SUP-009	Restoration of 4kV Power from EP	4
1/2-OHP-4023-SUP-012	Restoring DG Power	½
1-EHP-4030-001-002	U1 Pri Cont Leak Rate Running Total, Rev 0	September 30, 2003
1-EHP-4030-001-002	U1 Pri Cont Leak Rate Running Total, Rev 1 (Superceded by 1-EHP-4030-134-001)	February 2, 2005
1-EHP-4030-134-001	U1 Pri Cont Leak Rate Running Total, Rev 2	March 6, 2007
1-OHP-4021-016-003	CCW Sys Operation	5
1-OHP-4021-032-001AB	DG 1-AB Operation	13
1-OHP-4021-032-001CD	DG 1-CD Operation	13
1-OHP-4021-032-008AB	Operating DG 1-AB Subsystems	8
1-OHP-4021-032-008CD	Operating DG 1-CD Subsystems	9
1-OHP-4022-016-001	Malfunction of the CCW Sys	5
1-OHP-4022.055.003	Loss of Condensate to AFW Pumps	8
1-OHP-4024-104	Annunciator No. 104 Response: ESW and CC	22
1-OHP-4030-132-027AB	AB Diesel Generator Operability Test (Train B)	0
1-OHP-4030-132-027CD	CD Diesel Generator Operability Test (Train A)	0
1-OHP-4030-132-217A	DG 1-CD Load Sequencing and ESF Testing	12-1
1-OHP-4030-132-217B	DG 1-AB Load Sequencing and ESF Testing	12

PROCEDURES

Number	Description or Title	Date or Revision
2-EHP-4030-001-001	U2 Pri Cont Leak Rate Running Total, Rev 1	April 25, 2003
2-EHP-4030-001-001	U2 Pri Cont Leak Rate Running Total, Rev 2 (Superceded by 2-EHP-4030-234-001)	February 9, 2005
2-EHP-4030-234-001	U2 Pri Cont Leak Rate Running Total, Rev 2	March 6, 2007
2-EHP-6040.256.116	AFW Flow Retention	1
2-OHP-4021-016-003	CCW Sys Operation	19
2-OHP-4021-032-001AB	DG 2-AB Operation	16
2-OHP-4021-032-001CD	DG 2-CD Operation	14
2-OHP-4021-032-008AB	Operating DG 2-AB Subsystems	8
2-OHP-4021-032-008CD	Operating DG 2-CD Subsystems	8
2-OHP-4030-232-027AB	AB Diesel Generator Operability Test (Train B)	0
2-OHP-4030-232-027CD	CD Diesel Generator Operability Test (Train A)	0
2-OHP-4030-232-217A	DG 2-CD Load Sequencing and ESF Testing	15
2-OHP-4030-232-217B	DG 2-AB Load Sequencing and ESF Testing	18
12-IHP-5021-EMP-009	Batt Cell Charging	5
12-IHP-5021-EMP-012	ITE 4kV Circuit Breaker Maint	11
12-IHP-5021-EMP-027	CCW Pump Changeover	2
12-IHP-5021-EMP-080	Eaton/Cutler-Hammer 4kV Circuit Breaker Maint	5
12-OHP-2110-CCA-001	Compensatory/Contingency Actions	3
12-OHP-4030-033-001	Supplemental Diesel Generator Testing	5
LOP-7030-MOP-001	Corrective Action Program Management Oversight Processes	7
OHI-4032	Leakage Monitoring Program	4
PMP-3100-IOA-001	Inter-Organizational Agreement Between AEP Utility Operations and AEP Nuclear Generating Grp	2
PMP-7030-OPR-001	Operability Determination	0

REFERENCES

Number	Description or Title	Date or Revision
--------------------	Sonar Inspection of Forebay Area, DC Cook	January 18, 2007
51-9016434-000	AREVA NP Electrical Products HK Switchgear Main Bus Certification Report	0
AEP LTR C 1099-08	Containment Recirculation Sump Water Inventory	October 1, 1999
AEP LTR C 1099-25	Containment Recirculation Sump Water Inventory	November 19, 1999
AEP-NRC-5055-14	AEP LTR - Pump and Valve IST Program	December 28, 2005
AEP-NRC-9945	IEIN 91-56 Eval	June 15, 1992
ATR-U1	Units 1 and 2 Administrative Technical Requirements Manual, Revision 20	May 23, 2001
C 1000-09	Valve Position for Auto Valves in AFW Sys	October 18, 2000
C 1100-08	Valve Position for Auto Valves in AFW Sys	November 10, 2000
CNN3734-E051117101	ABB Switchgear Bus Certification Report	2

REFERENCES

Number	Description or Title	Date or Revision
Commitment 7798	C0600-13: License Amendment Request for CR Habitability and GL99-02 Requirements	June 12, 2000
DB-12-AFWS	Design Basis Document for the AFW Sys	2
DB-12-CCW	Design Basis Document for the CCW Sys	3
DB-12-ECCS	Design Basis Document for the ECC Sys	1
DCCEE-111-QCN	4160V and 600V Switchgear Specifications	0
DIT-B-10061-09	EOP Operator Action Times from Accident Analyses	July 30, 2003
DIT-B-10061-10	EOP Operator Action Times from Accident Analyses	February 11, 2005
GE-28803G	GE Type PJC Relay Instructions	0
GEK-34053G	GE Type IAC Relay Instructions	0
Memo	Comp Measures to Assure Offsite and CR Dose Limits	February 28, 2007
MPR-2136	RWST Model Vortex Testing for D.C.Cook	0
NED-2000-557-REP	Cont Sump Inventory Program Summary Report	0
SLC-05-0018	OE Review Summary for 4.16kV Roll-In Replacement Circuit Breakers for DC Cook	June 14, 2005
VTD-WORT-0001	Worthington Corp Installation and Operating Inst for 4 Cycle-Diesel and Dual Fuel Engines, Type SWB-VEE [Pub No. 4314-EI-OE]	6

SURVEILLANCES

Number	Description or Title	Date or Revision
--------------------	IST Data for 2PP-7W; 1PP-7W	May 16, 2006 through December 16, 2006
--------------------	IST Data for 2PP-7W; 1PP-7W	May 25, 2006 through November 28, 2006
1-EHP-4030-118-001	RWST Isolation Valve Leak Test	5
1-OHP-4030-133-038	Leak Rate Test of Liquid Sys	5
1-OHP-4030-STP-017R	AFW Pump Response Time	9a
1-PP-3W	MDAFW Pump Test Result Summary	November 8, 2003 through November 27, 2006
2-EHP-4030-218-001	RWST Isolation Valve Leak Test	8
2 EHP SP.114	CCW Pump Performance Test	0
2-OHP-4030-233-038	Leak Rate Test of Liquid Sys	3
2-PP-4	TDAFW Pump Test Result Summary	November 25, 2003 through November 1, 2006
2-PP-10E	CCW Pump Test Result Summary	January 15, 2004 through November 30, 2006

SURVEILLANCES

Number	Description or Title	Date or Revision
12-EHP-5030-001-008	Recirculation Loop Total Leak Rate	7
12-IHP-4030-082-003	AB, CD, and N-Train Batt Discharge Test and 24-Month Surveillance Requirements	15
12-IHP-4030-082-006	AB, CD, and N-Train Batt Yearly Surveillance and Maintenance	1

WORK DOCUMENTS

Number	Description	Revision/Date
R0204955	Clean XFMR 1-TR11A per RT00001355-01	April 2, 2005
R0206591	Perform 1-TR1AB Internal Inspection per RT00004459-01	October 18, 2006
R0233595	Perform 12-IHP-6030-IMP-069, Attach 5 for 1-TR101AB, B Train RAT Sudden Press Relay 18-Month Functional Check by I&C Personnel	December 7, 2004
R0253617	Doble Test 1-TR1AB	April 5, 2005
5523075101	Inspect/Test/Clean 1-TR1AB Grd Resistance per RT00016087-01	October 18, 2006
5528787201	Perform 1-Batt-AB 7-Day Surv per 12-IHP-4030-082-001	December 29, 2006
5528852601	Perform 1-Batt-AB 7-Day Surv per 12-IHP-4030-082-001	January 16, 2007

LIST OF ACRONYMS USED

ac or AC	Alternating Current
ADAMS	Agency-Wide Document Access and Management System
AR	Action Request
CAP	Corrective Action Program
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CR	Control Room
CST	Condensate Storage Tank
CVCS	Chemical and Volume Control System
dc	Direct Current
EDG	Emergency Diesel Generator
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
ECCS	Emergency Core Cooling System
ESW	Essential Service Water
Hz	Hertz
IMC	Inspection Manual Chapter
IST	Inservice Testing
LOV	Loss of Voltage
MCC	Motor Control Center
MDAFW	Motor Driven Auxiliary Feedwater
MOV	Motor-operated Valve
NCV	Non-Cited Violation
NPSH	Net Positive Suction Head
NRC	U. S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
ODE	Operability Determination Evaluation
OE	Operating Experience
PARS	Publicly Available Records System
RHR	Residual Heat Removal
RWST	Refueling Water Storage Tank
SDP	Significance Determination Process
SI	Safety Injection
TDAFW	Turbine Driven Auxiliary Feedwater
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
V	Volt