September 21, 2000

Mr. R. P. Powers
Senior Vice President
Nuclear Generation Group
American Electric Power Company
500 Circle Drive
Buchanan, MI 49107-1395

SUBJECT: D. C. COOK NUCLEAR POWER PLANT- NRC INSPECTION REPORT 50-315-00-19(DRP); 50-316-00-19(DRP)

Dear Mr. Powers:

On August 26, 2000, the NRC completed a baseline inspection at your D. C. Cook Units 1 and 2 reactor facility. The inspection results were discussed on August 25, 2000 with you and other members of your staff.

This inspection was an examination of activities conducted under your license as they relate to safety and compliance with the Commission's rules, regulations, and the conditions of your license. Within these areas, the inspection consisted of reviews of selected procedures and representative records, observations of activities, and interviews with personnel. Specifically, this inspection focused on resident inspection activities.

This inspection is the first of our planned inspections at your facility to implement the Revised Reactor Oversight Process and the Risk-Informed Baseline Inspection Program. Our schedule and process for implementing the Revised Reactor Oversight Process and the Risk-Informed Baseline Inspection Program was also discussed with you in our March 10, 2000, public meeting, and documented in letters dated August 15, 2000, and April 20, 2000.

Based on the results of this inspection, the NRC has determined that five violations of NRC requirements occurred, but because of their very low risk significance, the violations are Non-Cited. These violations involved a failure to maintain the loading on a Unit 1 600 VAC within procedural limits, inoperability of an auxiliary feedwater pump due to incorrect flow retention valve settings, failure to monitor control rod position with the rod position deviation monitor inoperable, power range trip setpoint exceeding the limiting safety system setting, and the failure to verify the position of certain essential service water valves. These NCVs are described in the subject inspection report. If you contest the violations or severity level of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region III; and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available **electronically** for public inspection in the NRC Public Document Room **or** from the *Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from* the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

Original signed by
John A. Grobe

John A. Grobe, Director
Division of Reactor Safety

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

Enclosure: Inspection Report 50-315-00-19(DRP);
50-316-00-19(DRP)

cc w/encl: A. C. Bakken III, Site Vice President
J. Pollock, Plant Manager
M. Rencheck, Vice President, Nuclear Engineering
R. Whale, Michigan Public Service Commission
Michigan Department of Environmental Quality
Emergency Management Division
MI Department of State Police
D. Lochbaum, Union of Concerned Scientists

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/s/John A. Grobe

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See Previous Concurrences

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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:	50-315-00-19(DRP); 50-316-00-19(DRP)
Licensee:	American Electric Power Company 1 Cook Place Bridgman, MI 49106
Facility:	D. C. Cook Nuclear Generating Plant
Location:	1 Cook Place Bridgman, MI 49106
Dates:	July 16, 2000 through August 26, 2000
Inspectors:	B. Bartlett, Senior Resident Inspector K. Coyne, Resident Inspector J. Maynen, Resident Inspector D. Passehl, Project Engineer
Approved by:	A. Vegel, Chief Reactor Projects Branch 6 Division of Reactor Projects

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

SUMMARY OF FINDINGS

NRC Inspection Report 50-315-00-19, 50-316-00-19, American Electric Power Company, Unit 1 and 2, conducted between July 16, and August 26, 2000, Event Followup.

The inspection was conducted by resident inspectors. This inspection identified four GREEN issues, all of which involved non-cited violations. The significance of issues is indicated by their color (GREEN, WHITE, YELLOW, RED) and was determined by the Significance Determination Process.

Mitigating Systems

- NO COLOR. During the review of Condition Report 99-20449, the inspectors identified a potential mis-classification of a maintenance preventable functional failure. This condition report described a loss of 250 VDC control power to the Unit 2 CD emergency diesel generator output breaker that occurred on August 7, 1999. The condition evaluation for the condition report concluded that the loss of breaker control power was caused by loose 250 VDC fuse block clips. The licensee determined that the probable cause for the loose fuse clips was a high number of maintenance activities which resulted in repeated mechanical cycling of the fuse clip assembly. The Maintenance Rule evaluation for the condition report concluded that the fuse clip failure was a Maintenance Rule functional failure, but was not maintenance preventable. The inspectors questioned the basis for this conclusion, since the condition evaluation for the condition report attributed the fuse clip failure to activities associated with maintenance.

  After the inspectors' questioning, the licensee determined that the failure was maintenance preventable, and therefore the condition represented a maintenance preventable functional failure. Condition Report 00238033 was initiated to document this issue. Because the performance criteria for the associated 250 VDC system function allowed no maintenance preventable functional failures per 24months, the failure to correctly classify this issue as maintenance preventable may have resulted in the failure to place the system under appropriate monitoring per the requirements of 10 CFR 50.65(a)(1). The inspectors determined this issue to be an Unresolved Item pending additional investigation. (Section 1R12)

Other Activities

- NO COLOR. The inspectors identified a human performance issue in that the Unit 1 Control Room operators did not notice that the 600 volt buses were drawing excessive current while cross-tied. This failure to follow procedural requirements for electrical bus current loading was considered to be a non-cited violation.

  This issue was screened as NO COLOR after a Phase 1 Safety Significance Determination review. (Section 4OA5)

Event Follow-up

- GREEN. On June 12, 2000, the licensee identified that the Unit 2 turbine driven auxiliary feedwater pump was inoperable for 88 hours after entry into Mode 3 (Hot Standby). The Technical Specification 3.7.1.2 action statement allowed the pump to be inoperable for 72 hours. The failure to comply with the action statement of Technical Specification 3.7.1.2 was considered to be a non-cited violation.

  This issue was screened as GREEN (very low safety significance) after a Phase 1 Significance Determination Process review. (Section 4OA3)

- GREEN. On June 28, 2000, the licensee identified that the rod position deviation monitor was inoperable. Due to a software error, the rod position deviation monitor failed to annunciate when the Technical Specification allowable deviation limit was reached. Technical Specification 4.1.3.2, required that with the rod position deviation monitor inoperable, the demand position indication system and the rod position indicator channel should be compared once per 4 hours. The failure to compare the demand position indication and the rod position indicator channel once per 4 hours was considered to be a non-cited violation of Technical Specification 4.1.3.2.

  This issue was screened as GREEN (very low safety significance) after a Phase 1 Significance Determination Process review. (Section 4OA3)

- GREEN. On July 13, 2000, the licensee identified that the neutron high flux trip setpoints for the power range nuclear instrumentation were above the Technical Specification 2.2.1 allowable value of 110 percent of rated thermal power. The licensee determined that the power range nuclear instrument setpoints were set based on an erroneous plant computer calorimetric. The failure to set the power range nuclear instrumentation neutron high flux trip setpoints within the limits of Technical Specification 2.2.1 was considered to be a non-cited violation.

  This issue was screened as GREEN (very low safety significance) after a Phase 1 Significance Determination Process review. (Section 4OA3)

- GREEN. During the recently completed outage the licensee added room coolers to the auxiliary feedwater pump rooms. Due to an error, four valves added to the essential service water system to cool the room coolers were not added to the surveillance which would periodically verify the position of the valves as required by Technical Specification 3.7.4.1. The failure to verify these essential service water valve positions once per 31 days was considered to be a non-cited violation of Technical Specification 3.7.4.1.

  This issue was considered a cross-cutting issue and screened as GREEN (very low risk significance) after a Phase 1 Significance Determination Process review. (Section 4OA3)

Report Details

Summary of Plant Status:

Unit 1 remained defueled throughout the inspection period. Licensee restart efforts have been re-focused from Unit 2 to Unit 1. During this inspection period, the first maintenance outages on both Train “A” and Train “B” were completed. In addition, ice condenser chill down was begun.

Unit 2 operated at or near full power during the inspection period with short duration power level changes for surveillance testing or other operational needs.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather

a. Inspection Scope

The inspectors reviewed the licensee’s procedures and preparations for high temperature, high wind, and flooding conditions. The inspectors reviewed severe weather procedures, emergency plan implementing procedures related to severe weather, annunciator response procedures, and performed general area walkdowns. Additionally, the inspectors reviewed condition reports and the identification and resolution of equipment deficiencies associated with adverse weather mitigation.

The inspectors reviewed annunciator response procedures for high temperature conditions or loss of ventilation to several safety-significant areas, including the switchgear rooms, the essential service water (ESW) pump rooms, and the emergency diesel generator (D/G) rooms. The inspectors walked down risk significant areas that could be affected by severe weather, including the 765 kV and 345 kV switchyards, areas outside the auxiliary and turbine buildings, and the fire pump house. During the walkdowns, the inspectors observed housekeeping conditions and verified that material capable of becoming an airborne missile hazard during high wind conditions or severe weather was appropriately restrained.

The inspectors reviewed applicable documentation relating to severe weather mitigation contained in the Updated Final Safety Analysis Report (UFSAR) and risk analysis notebooks from the licensee’s 1995 individual plant examination. This review was conducted to verify consistency between the licensee’s risk and safety analyses and the as-built plant configuration and operating procedures. Additionally, the inspectors reviewed the following documents:

- Plant Managers Procedure (PMP) 2080.EPP.101, “Emergency Classification,” Revision 3
- PMP 2080 EPP.111, “Natural Emergency Guidelines,” Revision 1

- PMP 2080 SWM.001, “Severe Weather Guidelines,” Revision 0
- 12-OHP [Operations Head Procedure] 4022.001.009, “Seiche,” Revision 0
- 12-OHP 4022.001.010, “Severe Weather,” Revision 0
- 02-OHP 4024.219(220), Drop 47 annunciator response, “DG2AB(CD) Room Vent Abnormal”
- 02-OHP-4024.203, Drop 42 annunciator response, “Electrical Switchgear Room Temperature High”
- 02-OHP-4024.204, Drop 54(64) annunciator response, “East (West) ESW Pump Room Temp Hi or Fan Fail”
- Condition Report (CR) 00-11073, 12-OHP 4022.001.010, Severe Weather, contains a severe weather condition that could be limiting for entering the procedure
- CR 00-10696, Guidance in 12-PMP 4030.001.001 has not been satisfactorily implemented in annunciator response procedures
- CR 00-7623, Installation of diesel generator tornado shield
- CR 00-3365, Unverified Design Information Transmittal for turbine building tornado analysis
- CR 99-22113, Tornado missile impact on diesel intake, exhaust, and room ventilation
- CR 99-4437, Inadequate procedural guidance to respond to a tornado
- CR 98-3194, Site personnel were not made aware of local tornado warning

b. Issues and Findings

There were no findings identified and documented during this inspection.

1R04 Equipment Alignment

.1 Complete Walkdown of the Unit 2 On-Site Emergency Power Sources

a. Inspection Scope

The inspectors performed a complete walkdown of the Unit 2 on-site emergency power sources, the Unit 2 AB D/G and the Unit 2 CD D/G. As part of this inspection, the inspectors reviewed ongoing system maintenance, open job orders (JOs), and design issues for potential effects on the ability of the emergency power sources to perform their design functions. The inspectors ensured that the configuration of the D/Gs was in accordance with applicable operating checklists. The inspectors also performed a complete system status check, which verified acceptable material condition of system components, availability of electrical power to system components, essential support systems availability, and that ancillary equipment or debris did not interfere with system performance. The Unit 2 on-site emergency power sources were selected for this inspection based on their importance as mitigating systems used to prevent core damage. As part of this inspection, the inspectors reviewed the licensee’s historical computerized issue tracking and JO data base, as well as the following documents:

- 02-OHP 4021.032.001AB, “DG2AB Operation,” Revision 6
- 02-OHP 4021.032.001CD, “DG2CD Operation,” Revision 5
- 02-OHP 4021.032.008AB, “Operating DG2AB Subsystems,” Revision 1

- 02-OHP 4021.032.008CD, “Operating DG2CD Subsystems,” Revision 1
- OP-2-5151A, “Flow Diagram Emergency Diesel Generator ‘AB’ Unit No. 2”
- OP-2-5151B, “Flow Diagram Emergency Diesel Generator ‘AB’ Unit No. 2”
- OP-2-5151C, “Flow Diagram Emergency Diesel Generator ‘CD’ Unit No. 2”
- OP-2-5151D, “Flow Diagram Emergency Diesel Generator ‘CD’ Unit No. 2”
- CR 00-9353, ABD/G bypass lubricating oil leaking at gasket seal on filter cover
- CR 00-11180, Oil leaking on 2-LTC-290 (ABD/G bypass lubricating oil filter heater temperature controller)

b. Issues and Findings

There were no findings identified and documented during this inspection.

.2 Partial Walkdown of Unit 2 North Control Room Chiller

a. Inspection Scope

The inspectors performed a partial walkdown of the Unit 2 north control room chiller. At the time of the walkdown, the opposite control room chiller train, 2-HV-ACRA-2, was out of service for routine preventative maintenance performed per Job Order R010115417, “Perform PM on 2-HV-ACRA-2.” The inspectors reviewed the operating status of the north control room ventilation chiller in order to verify compliance with Technical Specification (TS) requirements during maintenance. As discussed in NRC Inspection Report 50-315-00-19; 50-316-00-19, Section O2.2.b.7, the licensee determined that availability of the non-safety related mechanical chiller unit is required to ensure operability of the control room ventilation chillers during periods of elevated lake temperature. The inspectors compared the operating status and configuration of the north control room ventilation chiller to the applicable operating procedures, the system valve lineup, and applicable flow diagrams. As part of this inspection, the inspectors reviewed the following documents:

- 02-OHP 4021.028.014, “Operation of the Control Room Air Conditioning and Pressurization/Cleanup Filter System,” Revision 10
- 02-OHP 5030.001.001, “Operations Plant Tours,” Revision 15
- 12-THP [Technical Head Procedure] 6020 CHM.315, “Miscellaneous Cooling System,” Revision 1
- OP-2-5149, “Flow Diagram Control Room Ventilation Unit No. 2"
- CR 00-10306, 02-OHP 4021.028.014 Valve Lineup Sheet Discrepancies
- CR 00-10386, Operations and Chemistry Procedures do not consistently position CRAC chilled water chemical addition pot vent valves.
- CR 00-10390, Potential loss of configuration control on control room air handler filter differential pressure gage isolation valves.

b. Issues and Findings

There were no findings identified and documented during this inspection.

.3 Partial Walkdown of Unit 2 High Head Safety Injection

a. Inspection Scope

The inspectors performed a partial walkdown of the high head safety injection portion of the Unit 2 centrifugal charging system. The inspectors ensured that the configuration of the high head safety injection system was in accordance with applicable operating checklists and that the system could perform its required design basis functions. The following documents were reviewed during this inspection:

- 02-OHP [Operations Head Procedure] 4021.008.002, Revision 12, Placing Emergency Core Cooling System in Standby Readiness
- 02-OHP 4021.003.001, Revision 17, Letdown, Charging, and Seal Water Operation
- OP-2-5142-42, Flow Diagram Emergency Core Cooling
- OP-2-5129-38, Flow Diagram CVCS [Chemical and Volume Control System] - Reactor Letdown and Charging
- OP-2-5129A-31, Flow Diagram CVCS - Reactor Letdown and Charging

b. Issues and Findings

There were no findings identified and documented during this inspection.

.4 Partial Walkdown of Both Units' Electrical Lineup In Support of Spent Fuel Pool Cooling

a. Inspection Scope

The inspectors performed a partial walkdown of the units' electrical lineup in support of spent fuel pool cooling. The inspectors ensured that the configuration of the electrical lineup was in accordance with applicable operating checklists and that the electrical system supporting spent fuel pool cooling could perform its required design basis function. The following documents were reviewed during this inspection:

- 01-OHP 4030.STP.021, "Event Initiated Surveillances," Revision 9
- 02-OHP 4030.STP.021, "Event Initiated Surveillances," Revision 11
- 01-OHP 4030.STP.030, "Daily and Shift Surveillance Checks," Revision 29
- 02-OHP 4030.STP.030, "Daily and Shift Surveillance Checks," Revision 31
- 01-OHP 4030.STP.031, "Operation Weekly Surveillance Checks," Revision 8
- 02-OHP 4030.STP.031, "Operation Weekly Surveillance Checks," Revision 11

b. Issues and Findings

There were no findings identified and documented during this inspection.

1R05 Fire Protection

a. Inspection Scope

The inspectors performed fire impairment walkdowns of four plant areas related to the Mitigating Systems Cornerstone: Fire Zone 6M, Auxiliary Building 587 foot Elevation; Fire Zone 62, Unit 1 Charging Pump Room; Fire Zone 63, Unit 2 Charging Pump Room; and Fire Zone 65, Unit 2 Safety Injection Pump Room. The following documents were reviewed during this inspection:

- PMP-2270.CCM.001, “Control of Combustible Materials,” Revision 0
- PMP-2270.FIRE.002, “Responsibilities for Cook Plant Fire Protection Program Document Updates,” Revision 0
- PMP-2270.WBG.001, “Welding, Burning and Grinding Activities,” Revision 0
- Plant Mangers Instruction (PMI) 2270, “Fire Protection,” Revision 26
- UFSAR Section 9.8.1, “Fire Protection System”
- CR 00-10357, Self-contained breathing apparatus (SCBA) locker had a tag which instructed workers to contact Radiation Protection prior to opening due to contamination on SCBA harness

b. Issues and Findings

There were no findings identified and documented during this inspection.

1R11 Licensed Operator Requalification

a. Inspection Scope

On August 4, 2000, the inspectors observed Operations Shift “E” during simulator training. The shift performed scenarios designed to exercise the use of Emergency Operating Procedure 02-OHP 4023.ECA-0.0, “Loss of All AC Power.” The shift also performed exercises involving instrument failures. The inspectors assessed communications and implementation of emergency operating procedures. In addition, the inspectors attended the licensee’s critique following performance of the simulator scenarios.

b. Issues and Findings

There were no findings identified and documented during this inspection.

1R12 Maintenance Rule Implementation

The inspectors evaluated the licensee’s implementation of the Maintenance Rule, 10 CFR Part 50.65, for the following Unit 2 safety significant systems: 2 AB D/G, Pressurizer Power Operated Relief Valves (PORVs), Auxiliary Feedwater (AFW), and 250 VDC Batteries. The inspectors reviewed recent Maintenance Rule evaluations for the systems listed above to verify: (1) Scoping in accordance with 10 CFR 50.65; (2) Characterizing failed structures, systems, and components (SSCs) (e.g., a functional failure, a maintenance preventable functional failure, or a repetitive maintenance preventable functional failure); (3) Safety significance classification; (4) 10 CFR 50.65 (a)(1) or (a)(2) classification for the SSCs; and (5) Appropriateness of performance criteria for SSCs classified as (a)(2) or the appropriateness of goals and corrective actions for SSCs classified as (a)(1). The inspectors also interviewed the licensee's Maintenance Rule coordinator and evaluated the licensee's monitoring and trending of performance data with the responsible system engineer.

.1 Unit 2 AB Emergency Diesel Generator

a. Inspection Scope

The inspectors reviewed the following plant procedures, Maintenance Rule documentation, and CRs related to this inspection:

- Plant Managers Instruction (PMI) 6080, "Emergency Diesel Generator Reliability Program," Revision 2
- Regulatory Guide 1.9, "Selection, Design, Qualification, Testing, and Reliability of Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants"
- Regulatory Guide 1.155, "Station Blackout"
- Technical Specifications 4.8.1.1, 4.8.1.2
- CR 99-20724, Relays 2-62-2-DGS-2 and 2-62-DOAB "as found" data out of tolerance

b. Issues and Findings

There were no findings identified and documented during this inspection.

.2 Unit 2 Pressurizer Power Operated Relief Valves

The inspectors reviewed the following plant procedures, Maintenance Rule documentation, and CRs related to this inspection:

- PMP 5035.MRP.001, "Maintenance Rule Program Administration," Revision 0
- NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"
- UFSAR Section 4.0, Reactor Coolant System
- Donald C. Cook Nuclear Plant Probabilistic Risk Assessment, Reactor Coolant System
- Scoping Matrix for Reactor Coolant System
- Scoping Matrix for Compressed Air System
- PORV (a)(1) Action Plan - Draft
- CR 00-7170, Unit Pressurizer PORV backup air regulators found out of spec
- CR 00-6048, During surveillance test air was discovered to be leaking by valve 2-CA-711

b. Issues and Findings

There were no findings identified and documented during this inspection.

.3 Auxiliary Feedwater System

a. Inspection Scope

The inspectors reviewed the following plant procedures, Maintenance Rule documentation, and CRs related to this inspection:

- PMP 5035.MRP.001, “Maintenance Rule Program Administration,” Revision 0
- NUMARC 93-01, “Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants”
- UFSAR Section 10.5.2, “Auxiliary Feedwater System”
- Donald C. Cook Nuclear Plant Probabilistic Risk Assessment, Auxiliary Feedwater System Notebook
- CR 98-6368, Oil samples following maintenance reflect excessive contamination
- CR 00-8589, 2-CF-129 (Chemical Feed to #1 Steam Generator) failed to seat during the performance of 02-OHP 4030.STP.017CS, Attachment #5 - Chemical Feed Valve Test
- CR 00-4755, The N Train Battery west exhaust fan tripped on thermal overload
- CR 00-4843, Unit 2 West motor driven auxiliary feedwater pump (MDAFP) shutdown due to low oil level
- CR 00-9691, Turbine driven auxiliary feedwater pump (TDAFP) room coolers do not start in time
- CR 00-9696, Unit 2 TDAFP is inoperable due to room temperature greater than 85°F
- CR 00-9586, TDAFP declared inoperable due to room cooler not functioning as expected
- CR 00-11191, Station blackout function of AFW system is not scoped into Maintenance Rule

b. Issues and Findings

There were no findings identified and documented during this inspection.

.4 Maintenance Rule Implementation for 250 VDC Battery, A & B Trains

a. Inspection Scope

The inspectors reviewed the following plant procedures, Maintenance Rule documentation, and CRs related to this inspection:

- TS 3.8.2.3, “D.C. Distribution - Operating”
- 12-OHP 4021.082.012, “Locating 250 VDC Grounds,” Revision 0
- 02-OHP 4024.220, “Annunciator #220 Response: Station Auxiliary CD,” Revision 5
- UFSAR, Section 8.0, “Electrical Systems”
- PMP 6065.FUS.001, “Plant Fuse Control Program,” Revision 1
- CR 99-20499, Fuse block clips for 2 CD D/G output breaker were loose

- CR 00-3370, The (a)(3) Periodic Assessment identified 16 SSCs statused as (a)(2) which should have been status (a)(1) for goal setting and monitoring as required under 10 CFR 50.65.
- CR 00-9006, Specific gravity for cell #69 in Unit 1 N Train Battery low out of specification
- CR 00238033, Condition Report 99-20499 had a Maintenance Rule functional failure identified. Further research indicates the condition should also be a maintenance preventable functional failure.
- CR 00238022, The performance criteria for the 250 VDC battery needs to be reviewed for consistency between battery performance and surveillance test failure criteria
- CR 00237057, Thirty-five CRs were evaluated as functional failures but did not receive apparent cause determination as required by corrective action program procedures

b. Issues and Findings

During the review of CR 99-20449, the inspectors identified a potential mis-classification of a maintenance preventable functional failure (MPFF). This CR described a loss of 250 VDC control power to breaker 2-T21D8, the 2 CD D/G output breaker to the T21D 4kV AC Bus, that occurred on August 7, 1999. The loss of control power was identified after the 2 CD D/G was paralleled to the T21D bus and the loss of breaker position indication was observed. The evaluation for the CR concluded that the loss of control power was caused by loose 250 VDC fuse block clips which resulted in a loss of continuity in the control power circuit. The loss of continuity resulted in the loss of remote breaker control and inability to automatically open the D/G output breaker to isolate a circuit fault.

The licensee's investigation determined that the probable cause for the loose fuse clips was repeated mechanical cycling. The CR stated that a high number of maintenance activities were performed that required repeated tagging out of the 2 CD D/G. A D/G tagout normally required racking out the output breakers and removing control power fuses, which caused mechanical cycling of the fuse block clips and impacted their strength.

The inspectors noted that PMP 6065.FUS.001 provided guidance for fuse removal and reinsertion during clearance activities. Step 3.7.2 of this procedure stated, "prior to reinsertion of fuses, they shall be inspected for general physical conditions such as loose ferrules, cracks, overheating, etc. Any questionable fuses shall be replaced." Condition Report 99-20449 included one corrective action to determine if additional training for non-licensed operators in the identification of loose control power fuse blocks was needed.

The 250 VDC battery trains included a safety function of providing electrical protection/isolation. The performance criteria established for this function was zero MPFFs to clear and isolate a circuit fault per train per 24 months. The Maintenance Rule evaluation for CR 99-20499, which was completed on April 26, 2000, concluded that the fuse clip failure was a Maintenance Rule functional failure, but was not maintenance preventable. The licensee determined that the fuse clip failure caused a loss of control power to the breaker and therefore resulted in a failure to provide electrical isolation protection. However, the Maintenance Rule evaluation stated "it is not possible to determine how the vibration and mechanical strain to the fuse clip could predict the point where electrical contact could not be maintained." The inspectors questioned the basis for this conclusion, since the evaluation for the CR attributed the fuse clip failure to clearance activities associated with maintenance. After the inspectors' questioning, the licensee determined that the failure was maintenance preventable, and therefore the condition represented an MPFF. Because the performance criteria for the associated 250 VDC system function allowed no maintenance preventable functional failures per 24 months, the failure to correctly classify this issue as maintenance preventable may have resulted in the failure to place the system under appropriate monitoring per the requirements of 10 CFR 50.65(a)(1). The licensee initiated CR 00238033 to document this issue. Pending additional investigation by the licensee and NRC review of this matter, this issue is identified as **Unresolved Item (URI) 50-316-00-19-01**.

1R13 Maintenance Risk Assessments and Emergent Work Control

The inspectors reviewed the licensee's evaluation of plant risk, scheduling, configuration control, and performance of maintenance associated with planned and emergent work activities.

.1 Removal of the North Spent Fuel Pool Cooling Loop From Service

a. Inspection Scope

The licensee's Outage Risk Assessment Management computer program determined that the scheduled removal of the north spent fuel pool cooling loop from service would result in unacceptable risk. After further review of the risk, the licensee determined that, based on the low decay heat load in the spent fuel pool, the spent fuel pool temperature would not exceed 177°F in the event of a total loss of spent fuel pool cooling. The inspectors reviewed the following documents and determined that the licensee had appropriately re-evaluated the risk to allow the work to proceed:

- Design Information Transmittal B-01443-00, "Removal of Service of (Unit 1) North SFP [spent fuel pool] Cooling Train to support Unit 1 Outage activities."
- PMP 4100.SDR.001, "Plant Shutdown Safety and Management," Revision 4
- CR 00-9099, Potential red risk path entry associated with Unit 1 work schedule

b. Issues and Findings

There were no findings identified and documented during this inspection.

.2 Replacement of the Unit 2 East ESW Pump

a. Inspection Scope

On July 28, 2000, during routine inservice testing, the licensee identified that performance of the Unit 2 West ESW pump entered the alert range due to low developed head. Additional testing revealed a declining performance trend on the Unit 2 East ESW pump as well, and the licensee decided to replace both Unit 2 ESW pumps. Workers replaced the Unit 2 West ESW pump on August 6, 2000. The licensee planned to replace the Unit 2 East ESW pump later the same month. The inspectors reviewed the licensee's on-line risk assessment for the Unit 2 East ESW pump replacement. The following documents were reviewed as part of the inspection:

- PMP 2291.OLR.001, "On-Line Risk Management," Revision 0
- Job Order C206691, "Replacement of 2-PP-7E [Unit 2 East ESW pump]"

b. Issues and Findings

There were no findings identified and documented during this inspection.

.3 Removal From Service of Unit 1 Critical Control Room Power Inverter

a. Inspection Scope

The inspectors reviewed the risk-planning associated with planned maintenance on the Unit 1 Critical Control Room Power Inverter. The following documents were reviewed as part of the assessment:

- PMP 2291.OLR.001, "On-Line Risk Management," Revision 0
- 01-OHP 4021.082.019, "Removing the CCRP [Critical Control Room Power] Inverter from Service," Revision 11

b. Issues and Findings

There were no findings identified and documented during this inspection.

1R15 Operability Evaluations

.1 Operability Determination Review for Component Cooling Water Ventilation

a. Inspection Scope

The inspectors reviewed the operability determination associated with CR 00-6947, "The new auxiliary building temperature calculations assume that there are two component cooling water (CCW) supply fans in-service." This CR was initiated on May 13, 2000, to document a discrepancy between calculation MD-12-HV-002-N, "Heat Gain Calculation, AES [Engineered Safeguards Ventilation] System," Revision 2, and the operating configuration of the CCW ventilation supply fans. Specifically, the calculation assumed that two of the three component cooling water pump ventilation fans were in service during a design basis loss of coolant accident (LOCA); however, only one of the fans automatically starts upon an engineered safeguards feature (ESF) actuation in Unit 2. Two of the CCW ventilation fans are powered from Unit 1 electrical buses and are not configured to automatically start upon an ESF actuation in Unit 2. A minimum of two CCW ventilation fans are required to operate during a design basis LOCA to ensure that sufficient cooling is provided to maintain adequate cooling to the CCW pumps.

The inspectors reviewed the condition report evaluation and operability determination, including compensatory actions for the operable but degraded equipment status. The inspectors reviewed applicable normal operating procedures and emergency operating procedures to verify consistency with the operability determination. Additionally, the inspectors reviewed the engineering calculations supporting the operability determination conclusions. The inspectors reviewed the following documents:

- 02-OHP 4023.E-0, "Reactor Trip or Safety Injection," Revision 13
- 01-OHP 4030.066.4025, "Unit 1 Appendix R and Ventilation Requirements for Unit 2," Revision 1, change 1
- UFSAR, Section 9.9.4, "Auxiliary Building Ventilation System, Design Evaluation"
- Calculation MD-12-HV-002-N, "Heat Gain Calculation, AES System," Revision 2
- Calculation EVAL-MD-12-ACCP-001-N, "Auxiliary Building Temperature Evaluation With One CCW Pump Area Supply Fan"
- CR 00-6947, The new auxiliary building temperature calculations assume that there are two CCW supply fans in-service. However, there is only one of the fans that starts on a Unit 2 loss of offsite power with a LOCA.
- CR 99-5688, Inconsistency between UFSAR, design basis documents, and design calculations for auxiliary building ventilation
- CR 00-11182, EVAL-MD-12-ACCP-001-N, "Auxiliary Building Temperature Evaluation with One CCW Pump Area Supply Fan," does not document all supporting justifications relative to the conclusion
- CR 00-11265, Potential for non-conservative TS 3.7.6.1, "ESF Ventilation System," if analysis requires two AES units to be running

b. <u>Issues and Findings</u>

During the review of information supporting the operability determination, the inspectors noted that Calculation MD-12-HV-002-N credited airflow from the non-accident unit during a design basis accident. Cooling air for each of the emergency core cooling system (ECCS) pumps was supplied from the auxiliary building general area and exhausted by either of the associated Unit's two ESF ventilation trains. Although the non-accident unit's ESF ventilation system does not provide direct ventilation cooling to the accident unit's ECCS pumps, the additional airflow through the auxiliary building caused by the second unit lowers auxiliary building general area temperatures. Therefore, operation of the non-accident unit ESF ventilation unit directly impacted the temperature of the air supplied to the accident unit ECCS pumps. The loss of the non-accident unit ESF ventilation could potentially result in higher maximum ECCS pump room temperatures than assumed in the safety analysis and challenge the continued operability of ECCS equipment during an accident.

The basis for TS 3.7.6.1, "Engineered Safeguards Feature (ESF) Ventilation System," stated that the operability of the ESF ventilation system ensured adequate cooling for ECCS equipment. Technical Specification 3.7.6.1 required operability of the ESF ventilation unit only when the associated unit is operating in Modes 1 (Power Operation) through 4 (Hot Shutdown). With a Unit defueled, or in Modes 5 or 6, the TS did not require the operability of either of that unit's two ESF ventilation trains. Title 10 CFR 50.36, "TSs," Section (c)(2), defined limiting conditions for operation as the lowest functional capability of equipment required for safe operation of the facility. The inspectors questioned if TS 3.7.6.1 limiting condition for operation requirements represented the lowest functional capability of the ESF ventilation system and adequately reflected safety analysis assumptions. The licensee initiated CR 00-11265 to investigate this issue. The licensee stated that their investigation would determine if NRC Administrative Letter 98-10, “Dispositions of TSs that are Insufficient to Assure Plant Safety,” applied to the resolution of this matter. Based on a review of normal operating procedures, the inspectors concluded that operation of the Unit 1 AES system was adequately controlled and that no immediate operability concerns existed. Pending resolution of the licensee’s investigation, this issue is identified as **URI 50-315-00-19-02; 50-316-00-19-02**.

.2 Operability Assessment Review for Degraded Essential Service Water Pumps

a. Inspection Scope

On July 28, 2000, the licensee identified that performance of the Unit 2 East ESW pump had degraded to the alert range for developed pump head. Further testing revealed that both of the Unit 2 ESW pumps and the Unit 1 West ESW pump had a declining performance trend. The inspectors reviewed the operability assessments for CRs written concerning the Unit 1 and Unit 2 West ESW pumps. The following documents were reviewed:

- PMP 7030.OLR.001, “Operability Determination,” Revision 4
- CR 00-04687, Unit 2 West ESW pump does not meet minimum operability requirements
- CR 00-09899, Unit 2 West ESW pump has a degrading performance trend
- CR 00-10471, Unit 1 West ESW pump differential pressure below alert range
- CR 00-10571, Unit 2 West ESW pump below alert limit of 60.6 paid; it is above the low action limit of 60.3 paid
- CR 00-10846, No effort was made to ensure that the discharge pressure gauge was filled and vented
- CR 00-10881, Unit 2 West ESW pump periodically degrading below minimum operability limit
- CR 00-11007, Category X tracking of ESW pump performance

b. Issues and Findings

There were no findings identified and documented during this inspection.

1R16 Operator Workarounds

.1 Review of Selected Operator Workarounds

a. Inspection Scope

The inspectors evaluated the following new operator workarounds (OWAs) to determine if the applicable system function was impacted or if the OWA affected the operator’s ability to implement abnormal or emergency operating procedures:

- OWA 00-05 Ice formation on intermediate deck doors required the non-licensed operators to perform shiftly tours.
- OWA 00-04 Debris build up in ESW lines to the Auxiliary Feedwater Pump room coolers required the non-licensed operators to frequently monitor flow, and flush as necessary.

b. Issues and Findings

There were no findings identified and documented during this inspection.

.2 Review of the Cumulative Effect of Operator Workarounds

a. Inspection Scope

The inspectors reviewed the cumulative effect of OWAs on equipment availability, initiating event frequency, and the ability of the operators to implement abnormal or emergency operating procedures. As part of this inspection, the inspectors reviewed the following licensee documents:

- PMP 4010.OWA.001, Revision 1, "Oversight and Control of Operator Workarounds"
- Workaround Review Board meeting summaries for June, July, and August, 2000
- Workaround list for Unit 1, 2, and common
- Operator Workaround Aggregate Effect Report for Unit 2, dated July 18, 2000
- Performance Assurance Field Observation FO-00-H-074, Observation of August 15, 2000, Workarounds Review Board meeting

The inspectors also interviewed the OWA Coordinator regarding the oversight and control of OWAs.

b. Issues and Findings

There were no findings identified and documented during this inspection.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed and observed post-maintenance testing following routine and emergent maintenance activities. During post-maintenance testing observations, the inspectors verified that the testing was adequate for the scope of the maintenance work which had been performed and that the acceptance criterion were clear and demonstrated operational readiness consistent with the design and licensing basis documents. The inspectors also verified that the impact of the testing had been properly characterized during the pre-job briefing; the test was performed as-written and all testing prerequisites were satisfied; and that the test data was complete, appropriately verified, and met the requirements of the testing procedure. Following the completion of the test, the inspectors verified that the test equipment was removed and that the equipment was returned to a condition in which it could perform its safety function.

The inspectors reviewed the following post-maintenance testing activities and verified that the post-maintenance testing met the requirements of PMP 2291.PMT.001, "Work Management Post Maintenance Testing Matrices," Revision 2 and PMI 2294, "Post Maintenance Testing Program," Revision 0:

- On July 21, 2000, the inspectors observed the post-maintenance testing following a limit switch adjustment on valve 2-FMO-240. The valve was tested using Unit 2 Operations Head Procedure (02-OHP) 4030.STP.018, "Steam Generator Stop Valve Dump Valve Surveillance Test," Revision 9, following completion of Job Order C205568-01, "Troubleshoot/Repair 2-MMO-240."

- On August 5, 2000, the licensee replaced the Unit 2 West ESW pump under Job Order C206304. The replacement pump was tested using 02-OHP 4030.STP.022W, "West ESW System Test," Revision 13.

- On August 23, 2000, the inspectors observed the post-maintenance testing for Job Order R0058298, "Unit 1 Control Air Wet Receiver Outlet Pressure Control Switch."

b. <u>Issues and Findings</u>

There were no findings identified and documented during this inspection.

1R22 <u>Surveillance Testing</u>

.1 <u>Surveillance Testing of the Unit 2 Essential Service Water Pumps</u>

a. <u>Inspection Scope</u>

On July 31, 2000, the inspectors observed performance of the quarterly inservice testing of the Unit 2 East ESW pump. This testing was conducted in accordance with 02-OHP 4030.STP.022E, "East Essential Service Water System Test," Revision 12, to satisfy the requirements of TS 4.0.5. The inspectors observed various testing activities; including briefings, communications, and test procedure performance. The inspectors reviewed gauge calibration requirements, verified test instruments were within proper calibrations, and checked for compliance with ASME code requirements.

The inspectors also reviewed the results of recent surveillance testing for the Unit 2 West ESW pump. Additionally, the inspectors evaluated the corrective actions for the declining ESW pump performance (see Sections 1R13.2 and 1R15.2).

The inspectors reviewed the following documents during this review:

- NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants"
- OMa-1988, Part 6, "Inservice Testing of Pumps in Light Water Reactor Power Plants"
- 02-OHP 4030.STP.022E(W), "East (West) Essential Service Water System Test," Revision 12 (12)
- Engineering Action Plan 00-475a

- CR 00-10571, Unit 2 West ESW pump differential pressure below alert limit
- CR 00-9899, Unit 2 West ESW pump has degrading performance trend
- CR 00-4687, Unit 2 West ESW pump does not meet minimum performance requirements
- CR 00-10471, Unit 1 West ESW pump differential below alert requirement
- CR 00-10846, Test instrument used for 02-OHP 4030.STP.022E was not filled and vented
- CR 00-10721, Operator did not attend to discharge pressure gauge while test gauge was valved in

b. Issues and Findings

There were no findings identified and documented during this inspection.

.2 Pressurizer Power Operated Relief Valve Functional Test

On August 8, 2000, the inspectors observed performance of 02 IHP 4030.STP.152, "Pressurizer Power Operated Relief Valve [PORV] Functional Test," Revision 1, Change 2. Technical Specification 4.4.11.1.a. required monthly channel functional testing of each Pressurizer PORV. The inspectors verified procedural and TS adherence and observed the conduct and knowledge of maintenance and operations personnel. The inspectors observed communications practices and surveillance test control by the operations crew. The inspectors also verified that equipment calibration was within the required periodicity and that the activity was appropriately scheduled. The inspectors reviewed the following documents:

- 02-IHP [Instrument Head Procedure] 4030.STP.152, "Pressurizer Power Operated Relief Valve Functional Test," Revision 1
- Technical Specification 3.4.11, "Reactor Coolant System, Relief Valves - Operating"
- OP-2-99023-2, "Pressurizer Pressure Control Functional Diagram"
- CR 00-9457, Unable to perform STP-152 because procedure would result in inoperability of all PORVs if performed as-written
- CR 00-9465, Pressurizer PORV testing was conducted with PORVs in manual without declaring PORV inoperable

b. Issues and Findings

There were no findings identified and documented during this inspection.

.3 Unit 2 Train "B" Reactor Protection System Surveillance Testing

a. Inspection Scope

The inspectors observed and reviewed surveillance tests for the Unit 2 Train "B" Reactor Protection System (RPS). The inspection included reviews of the applicable TS requirements and Final Safety Analysis Report sections, in addition to the design basis documents and vendor manuals. The following surveillance test procedure was reviewed during this inspection:

- 02-IHP 4030.STP.511, “Train ‘B’ RPS [reactor protection system] and ESF Reactor Trip Breaker and SSPS [solid state protection system] Automatic Trip/Actuation Logic Functional Test,” Revision 3.

b. Issues and Findings

There were no findings identified and documented during this inspection.

1R23 Temporary Modifications

a. Inspection Scope

The inspectors screened active temporary modifications on risk-significant systems in order to assess their effect on the safety functions of these systems. As part of this effort, the inspectors reviewed the following documentation:

- Open Temporary Modification Log - Unit 2
- Open Temporary Modification Log - Unit 1 and Common

b. Issues and Findings

There were no findings identified and documented during this inspection.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

On August 1, 2000, the licensee performed an emergency planning (EP) drill. Prior to the drill, the licensee determined that the results of the drill would be included in the development of performance indicators for drill/exercise participation and emergency response organization participation. The drill scenario involved fuel damage which led to high radiation levels in the auxiliary building and an off-site release.

The inspectors reviewed the drill scenario, observed the drill performance from the simulator and the Technical Support Center, and attended the post-drill critique meetings. In addition, the inspectors reviewed the following documents:

- CR 00-10524, Emergency plan drill assessment
- CR 00-11371, Failure of the operating crew to classify an Unusual Event during August 1, 2000, emergency plan drill
- CR 00-10127, Pagers in Buchanan area are a problem
- CR 00-10843, Weaknesses in shutdown margin computer calculation
- CR 00-10743, Plant process computer failed

b. Issues and Findings

There were no findings identified and documented during this inspection.

4. OTHER ACTIVITIES (OA)

4OA3 Event Follow-up

.1 Licensee Event Reports

a. Inspection Scope

The inspectors reviewed the corrective actions associated with the following licensee event reports.

b. Issues and Findings

(Closed) Licensee Event Report 50-316-00-5-00: Auxiliary feedwater pump inoperable due to incorrect flow retention valve settings. On June 12, 2000, the licensee placed Unit 2 in Mode 3 (Hot Shutdown). At the time of the Mode 3 entry, the Unit 2 turbine-driven auxiliary feedwater pump (TDAFP) was inoperable awaiting its surveillance test. Technical Specification 4.7.1.2.b authorized entry into Mode 3 to allow the TDAFP to be tested. After the licensee reviewed the surveillance test and determined that the TDAFP passed all of its acceptance criteria, the TDAFP was declared operable. Later, the licensee identified that the flow retention limit switches were incorrectly set. The licensee determined that the TDAFP had been inoperable since entry into Mode 3. The licensee reset the flow retention limit switches 88 hours after entry into Mode 3.

Technical Specification 3.7.1.2 required at least three independent steam generator auxiliary feedwater pumps and associated flow paths to be operable whenever the unit was in Modes 1, 2, or 3. The action statement for TS 3.7.1.2a stated, in part, “With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to operable status within 72 hours or be in at least hot standby within the next 6 hours and hot shutdown within the following 6 hours.” Contrary to the above, the TDAFP was inoperable for 88 hours after entry into Mode 3 due to the pump discharge valve flow retention limit switches being incorrectly set. The failure to comply with the action statement of TS 3.7.1.2a for the inoperable TDAFP was considered a violation of TS 3.7.1.2. This violation is being treated as a **Non-Cited Violation (50-316-00-19-03)** consistent with Section VI.A. of the NRC Enforcement Policy. This violation was entered into the licensee’s corrective action program as CR 00-8741 and CR 00-8830. This issue was screened as GREEN (very low risk significance) after a Phase 1 Significance Determination Process review. This LER and NCV are closed.

(Closed) Licensee Event Report 50-316-00-7-00: Technical Specification 3.0.3 shutdown initiated due to inoperable rod position indications. On June 28, 2000, the licensee was increasing Unit 2 reactor power following an outage and determined that two individual rod position indicators (IRPI) deviated from the group step counter by greater than that allowed by TS. The TSs allowed only one IRPI to be inoperable. Having 2 IRPIs inoperable resulted in a TS 3.0.3 entry and initiation of a plant shutdown.

Additionally, the rod position deviation monitor failed to annunciate when 2 IRPIs deviated from their group step counter demand by greater than 18 steps. The error had existed since installation of the software in the early 1990's. With the rod position deviation monitor inoperable, TSs required rod position readings be taken once per 4 hour. The unrecognized software error contributed to the failure to comply with the TS requirements.

The licensee determined the cause of the position deviation to be temperature-induced drift in the IRPI circuitry. The licensee adjusted the IRPI circuitry and the rod position indication was restored within the TS allowable value. The licensee planned to correct the position deviation monitor software error prior to Unit 1 restart.

Technical Specification Surveillance 4.1.3.2, required, in part, that with the rod position deviation monitor inoperable, the demand position indication system and the rod position indicator channel should be compared once per 4 hours. The failure to perform the rod position comparisons once per 4 hours was a violation of TS 4.1.3.2. This violation is being treated as a **Non-Cited Violation (50-316-00-19-04)** consistent with the Section VI.A of the NRC Enforcement Policy. This violation is in the licensee's corrective action system as Condition Report 00-9358. This issue was screened as GREEN (very low risk significance) after a Phase 1 Significance Determination Process review. This LER and NCV are closed.

(Closed) Licensee Event Report 50-316-00-8-00: Erroneous plant process computer input data resulted in power range trip setpoints above TS limit. On July 13, 2000, the licensee identified that the neutron high flux trip setpoints for the power range nuclear instrumentation (NIs) were above the TS 2.2.1 allowable value of 110 percent of rated thermal power. The licensee determined that on June 25, 2000, a calorimetric was performed that was in error due to feedwater temperature instruments reading high. The high feedwater temperature resulted in the plant computer calculating a slightly low reactor power. When the NIs were adjusted based upon this calorimetric, all four NI setpoints were accidentally set between 110.24 percent and 111.04 percent reactor power. These setpoints were higher than the TS 2.2.1 allowable value of 110 percent. The licensee immediately reset the NIs to their required value of less than or equal to 109 percent.

The licensee determined that the procedure for adjusting NIs based upon secondary calorimetric calculations did not contain restrictions on the allowable amount of adjustment. The feedwater temperature input to the plant computer was corrected by tightening a loose computer card. The licensee has scheduled the surveillance procedure to be revised prior to its next usage. The failure to set the power range NIs to less than or equal to the values required in TS 2.2.1 was a violation. This violation is being treated as a **Non-Cited Violation (50-316-00-19-05)** consistent with Section VI.A of the NRC Enforcement Policy. This violation is in the licensee's corrective action system as Condition Report 00-9437. This issue was screened as GREEN (very low risk significance) after a Phase 1 Significance Determination Process review. This LER and NCV are closed.

(Closed) Licensee Event Report 50-316-00-9-00: Common cause ventilation failure results in inoperable auxiliary feedwater pumps. During routine operator rounds on July 5, 2000, the licensee determined that room coolers to the Unit 2 turbine driven auxiliary feedwater pump (TDAFP) were not operating properly. Subsequently, the licensee determined that the coolers were blocked with sand and silt in the ESW cooling lines. The lines were cleared and the room coolers restored to operable. The licensee had previously determined that the room coolers for the AFW pumps were required to support operability.

On July 7, 2000, reduced cooling water flow to the Unit 2 East Motor Driven Auxiliary Feedwater Pump (MDAFP) room cooler were observed and the pump was declared inoperable. The lines were cleared and the room cooler was restored to operable. During the time that each auxiliary feedwater pump was inoperable, TSs limiting condition for operations were met.

The licensee determined that a common cause design failure resulted in the room cooler ESW return valves being throttled such that sand and silt was not allowed to pass. Licensee analysis determined that the TDAFP would continue to be operable for 4 hours following the failure of the room cooler and that there was no required mission for the TDAFP that was longer than four hours. The licensee also determined that the Unit 2 East MDAFP would continue to operate for 2.5 hours following room cooler failure and that was sufficient time to allow manual compensatory measures.

This item is in the licensee's corrective action system as Condition Report 00-9586, 00-9639, 00-9640, and 00-9674. This issue was screened as GREEN (very low risk significance) after a Phase 1 Significance Determination Process review. This LER is closed.

(Closed) Licensee Event Report 50-316-00-10-00: Failure to verify position of ESW valves as required by TSs. During the recently completed outage the licensee added room coolers to the auxiliary feedwater pump rooms. Due to an error, four valves added to the ESW system to cool the room coolers were not added to the surveillance which would periodically verify the position of the valves as required by TS Surveillance 3.7.4.1. The failure to verify valve position once per 31 days was a violation of TS 3.7.4.1. The valves were found to be in their correct position. This violation is being treated as a **Non-Cited Violation (50-316-00-19-06)** consistent with the Section VI.A of the NRC Enforcement Manual. This violation is in the licensee's corrective action system as Condition Report 00-9693. This issue was screened as GREEN (very low risk significance) after a Phase 1 Significance Determination Process review. This LER and NCV are closed.

.2 Previous NRC Inspection Report Findings

a. Inspection Scope

The inspectors reviewed the licensee's corrective actions for the following previously identified NRC inspection report findings:

b. Issues and Findings

Closure of Severity Level IV Violations Under Revised Enforcement Policy

On May 1, 2000, the NRC revised NUREG-1600, "General Statement of Policy and Procedures for NRC Enforcement Actions," (Enforcement Policy). Section VI.A, "Non-Cited Violation (NCV)," of the Enforcement Policy discusses the NRC's enforcement approach for Severity Level IV violations. The Policy allows dispositioning of a Severity Level IV violation as a non-cited violation provided certain requirements are met. These requirements include entry of the violation into the licensee's corrective action program and restoration of compliance with NRC requirements, as well as other considerations described in the Enforcement Policy. In accordance with the Enforcement Policy, dispositioning of a severity Level IV violation as an NCV allows closure of the violation without a written response from the licensee. The NRC has conducted a review of the following Severity Level IV violations, and considers it appropriate to close these violations consistent with Section VI.A of the Enforcement Policy:

	Violation Number	Corrective Action Program File Number
•	50-315/95009-01 50-316/95009-01	CR 95-1224
•	50-315/97002-03	CR 99-18269
•	50-315/97002-06	CR 99-18271
•	50-315/97002-07	CR 99-18274
•	50-315/97018-01 50-316/97018-01	CR 99-18129, CR 99-18130, CR 99-18131, CR 99-18355
•	50-315/98008-01	CR 99-18118, 98-1909
•	50-315/98008-02	CR 99-18115, 98-1906
•	50-316/98012-01	CR 99-18106
•	50-315/98012-02	CR 98-3255

Closure of New Fuel Receipt Inspection Procedural Violations

- 50-315/97002-02: Failure to follow procedures step by step.
- 50-315/97002-04: Failure to follow procedure (supervisor monitoring of Foreign Material Exclusion Zone (FMEZ)).
- 50-315/97002-05: Failure to follow procedure (FMEZ work practices).

These three Severity Level IV violations involved procedural adherence and foreign material exclusion program control violations identified during new fuel receipt inspections conducted during January and February of 1997. The licensee canceled Maintenance Head Procedures (MHP) 12-MHP 4050.FD.002, "Unloading of New Fuel

Assemblies From Shipping Containers," and 12-MHP 4050.FD.005, "Handling of New Fuel Assemblies for Inspections and Associated Work," and incorporated these procedures into Revision 5 of 12-MHP 4050.FD.001, "Receipt and Storage of New Fuel Assemblies," in July 1997. Subsequent new fuel receipt inspection observations, documented in NRC Inspection Report No. 50-315/97015(DRP); 50-316/97015(DRP), Section M4.1, "New Fuel Receipt Inspection (Unit 2)," concluded that the licensee had initiated extensive new fuel inspection program upgrades and no additional concerns were identified during observations of new fuel receipt inspections. As documented in NRC Inspection Report Numbers 50-315/99017; 50-316/99017 and 50-315-00-13; 50-315-00-13, inspector observations of core off-load and core reload concluded that fuel movement operations were adequately controlled with appropriate foreign material exclusion controls. Additionally, the licensee has identified the failure to follow procedures during new fuel inspections within the corrective action program in CR 99-18181, 99-18182, and 99-18267. Based on the above, these violations are closed.

4OA5 Other

.1 Inspector Identified Human Performance Issue

During routine control room observations on August 22, 2000, the inspectors identified that the phase 2 amperage meter for the 600 VAC Bus 11A exceeded the maximum allowable current. Procedure 01-OHP 4021.082.003, "Feeding 600 Volt Buses Through Bus Tie Breakers," Revision 7, Step 3.1 specified that the maximum loading on Bus 11A was 225 amps as indicated on the bus amperage indicator. Contrary to this procedural requirement, the inspectors noted that the phase 2 amperage for Bus 11A indicated approximately 230 amps. After the inspectors notified the control room staff of the high loading condition, the operators reduced electrical bus loading on Bus 11A below the 225 amp limitation. The inspectors were informed that the previous operations shift had identified a high, but acceptable, loading condition on Bus 11A, and turned over that bus loading should be monitored. The inspectors concluded that the operators failed to appropriately monitor and control the loading on Bus 11A. The risk significance of this occurrence was minimal as the 600 Volt buses were not required to be operable per TSs. However, the inspectors determined that this event was an example of poor human performance by the Unit 1 Control Room operators. As discussed in Section 4OA5.2 below, the licensee has identified an adverse trend in human performance. This issue was screened as NO COLOR after a Phase 1 Significance Determination Process review.

The inspectors determined that the failure to follow plant procedural requirements was a Violation of TS 6.8.1. This violation is being treated as a **Non-Cited Violation (50-315-00-19-07)**, consistent with Section VI.A of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as CR 00-11658.

.2 Licensee Identified Human Performance Issues

On August 24, 2000, the licensee identified an adverse trend in Operations human performance. Operations personnel wrote CR 237022 to document the issue and identify several specific examples of human performance errors. Operations personnel identified that the cause of the following Licensee Event Reports was due in part to human performance issues:

- Licensee Event Report (LER) 50-315-/00-04-00: Partial loss of off-site power results in start of emergency diesel generators. This issue was discussed in NRC Inspection Report 50-315-00-17; 50-316-00-17.

- LER 50-316-00-07-00: Technical Specification 3.0.3 shutdown initiated due to inoperable rod position indications. This LER is discussed in Section 4OA3 of this report.

- LER 50-316-00-08-00: Erroneous plant process computer input data resulted in power range trip setpoints above TS limit. This LER is discussed in Section 4OA3 of this report.

- LER 50-316-00-10-00: Failure to verify position of essential service water valves as required by TSs. This LER is discussed in Section 4OA3 of this report.

In addition, the licensee identified a number of additional condition reports which could be attributed to human performance issues. Based on the above findings, operations management directed that each control room operating shift shall conduct a 1-hour training session on the above events to fucus attention on reducing the occurrence of human performance errors. Additionally, operations management established a human performance root cause team to investigate these events and make recommendations for correcting the human performance problems.

4OA6 Management Meetings

The inspectors presented the inspection results to Mr. Powers and other members of licensee management listed below at the conclusion of the inspection on August 25, 2000. The licensee acknowledged the findings presented. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

A. Bakken, Site Vice President
M. Barfelz, Regulatory Affairs
T. Foster, Performance Assurance
R. Gaston, Regulatory Affairs
J. Gebbie, Plant Engineering
W. Harland, Work Control
M. Hoskins, System Engineering
W. Kropp, Regulatory Affairs
R. Meister, Regulatory Affairs
J. Molden, Director, Maintenance
D. Naughton, System Engineering
S. Partin, Assistant Operations Manager
R. Powers, Senior Vice President
B. Wallace, Manager, Training
L. Weber, Manager, Operations
D. Wood, Manager, Radiological Protection, Chemistry, and Environmental

LIST OF INSPECTIONS PERFORMED

The following inspectable-area procedures were used to perform inspections during the report period. Documented findings are contained in the body of the report.

Inspection Procedure		Report Section
Number	Title	
71111-01	Adverse Weather	1R01
71111-04	Equipment Alignments	1R04
71111-05	Fire Protection	1R05
71111-11	Licensed Operator Requalification	1R11
71111-12	Maintenance Rule Implementation	1R12
71111-13	Maintenance Risk Assessment and Emergent Work Evaluation	1R13
71111-15	Operability Evaluations	1R15
71111-16	Operator Workarounds	1R16
71111-19	Post-maintenance Testing	1R19
71111-22	Surveillance Testing	1R22
71111-23	Temporary Modifications	1R23
71114-06	Drill Evaluation	1EP6
71153	Event Followup	4OA3

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-316-00-19-01	URI	Potential mis-classification of a Maintenance Rule maintenance preventable functional failure
50-315-00-19-02 50-316-00-19-02	URI	Technical Specification 3.7.6.1, “ESF Ventilation System,” potentially non-conservative
50-316-00-19-03	NCV	Auxiliary feedwater pump inoperable due to incorrect flow retention valve settings
50-316-00-19-04	NCV	Failure to perform 4 hour rod position surveillance with inoperable rod deviation monitor
50-316-00-19-05	NCV	Erroneous plant process computer input data resulted in power range trip setpoints above TS limit
50-316-00-19-06	NCV	Failure to verify position of essential service water valves as required by TSs
50-315-00-19-07	NCV	Failure to follow procedural controls on amperage limit with cross-tied 600VAC busses

Closed

50-315/95009-01 50-316/95009-01	VIO	Failure to follow procedures - 3 examples
50-315/97002-02	VIO	Failure to follow procedures step by step
50-315/97002-03	VIO	Failure to follow procedures (undersized sling)
50-315/97002-04	VIO	Failure to follow procedure (supervisor monitoring of Foreign Material Exclusion Zone (FMEZ))
50-315/97002-05	VIO	Failure to follow procedure (FMEZ work practices)
50-315/97002-06	VIO	Inadequate procedures (step sequence, FMEZ practices, rigging requirements)
50-315/97002-07 50-316/97002-07	VIO	Failure to follow procedures (FME briefings)
50-315/97018-01 50-316/97018-01	VIO	Failure to have instructions of a type appropriate to the circumstances

50-315/98008-01	VIO	Failure to install jam nuts on Unit 1 CD D/G exhaust manifold bracket as required by modification drawing
50-315/98008-02	VIO	Failure to determine adequate root cause for missing jam nuts on Unit 1 CD D/G
50-316/98012-01	VIO	Inadequate corrective actions related to monitoring reactor vessel level indication for venting the reactor head
50-315/98012-02	VIO	Procedure inappropriate to the circumstance for moving ice
50-316/00005-00	LER	Auxiliary feedwater pump inoperable due to incorrect flow retention valve settings
50-316/00007-00	LER	Technical Specification 3.0.3 shutdown initiated due to inoperable rod position indications
50-316/00008-00	LER	Erroneous plant process computer input data resulted in power range trip setpoints above TS limit
50-316/00009-00	LER	Common cause ventilation failure results in inoperable auxiliary feedwater pumps
50-316/00010-00	LER	Failure to verify position of essential service water valves as required by TSs
50-316-00-19-03	NCV	Auxiliary feedwater pump inoperable due to incorrect flow retention valve settings
50-316-00-19-04	NCV	Failure to perform 4 hour rod position surveillance with inoperable rod deviation monitor
50-316-00-19-05	NCV	Erroneous plant process computer input data resulted in power range trip setpoints above TS limit
50-316-00-19-06	NCV	Failure to verify position of essential service water valves as required by TSs
50-315-00-19-07	NCV	Failure to follow procedural controls on amperage limit with cross-tied 600VAC busses

LIST OF ABBREVIATIONS

AC	Alternating Current
AES	Engineered Safety Features Ventilation
AFW	Auxiliary Feedwater System
ASME	American Society of Mechanical Engineers
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CR	Condition Report
CRAC	Control Room Air Conditioning Unit
CVCS	Chemical and Volume Control System
D/G	Diesel Generator
DRP	Division of Reactor Projects
ECCS	Emergency Core Cooling System
EP	Emergency Planning
ESF	Engineered Safety Features
ESW	Essential Service Water
FMEZ	Foreign Material Exclusion Zone
IHP	Instrument Head Procedure
IRPI	Individual Rod Position Indicator
JO	Job Order
kV	Kilovolt
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
MC	Manual Chapter
MDAFP	Motor Driven Auxiliary Feedwater Pump
MHP	Maintenance Head Procedure
MOV	Motor Operated Valve
MPFF	Maintenance Preventable Functional Failure
NCV	Non-Cited Violation
NI	Nuclear Instrumentation
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
OHI	Operations Head Instruction
OHP	Operations Head Procedure
OSO	Operations Standing Order
OWA	Operator Workaround
PDR	Public Document Room
PMI	Plant Manager’s Instruction
PMP	Plant Manager’s Procedure
PMT	Post-maintenance Testing
PORV	Power Operated Relief Valve
PPC	Plant Process Computer
RCS	Reactor Coolant System
RPS	Reactor Protection System
RWST	Refueling Water Storage Tank
SCBA	Self-Contained Breathing Apparatus
SFP	Spent Fuel Pool

SRO	Senior Reactor Operator
SSC	Structures, Systems, and Components
SSPS	Solid State Protection System
STP	Surveillance Test Procedure
TDAFP	Turbine Driven Auxiliary Feedwater Pump
TDB	Technical Data Book
THP	Technical Head Procedure
TS	Technical Specification
URI	Unresolved Item
UFSAR	Updated Final Safety Analysis
VDC	Volts Direct Current
VIO	Violation