June 13, 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 INSPECTION REPORT 50-315/2000013(DRP); 50-316/2000013(DRP)

Dear Mr. Powers:

This refers to the inspection conducted on April 2, 2000, through May 27, 2000 at the D. C. Cook Units 1 and 2 reactor facilities. The inspection was an examination of activities conducted under your license as they relate to compliance with the Commission rules and regulations and with the conditions of your license. Areas reviewed included Operations, Maintenance, Engineering, and Plant Support. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observations of activities in progress. The inspectors also reviewed observations and findings as they related to the NRC Manual Chapter 0350 Case Specific Checklist for D. C. Cook. The enclosed report presents the results of that inspection.

During this inspection period, we noted that progress has been made in preparing the plant for restart. The loading of fuel into the Unit 2 reactor vessel was well controlled with appropriate management and Performance Assurance oversight. The inspectors conducted an assessment of operability determinations and determined that the evaluations were thorough and that corrective actions were commensurate with plant safety. Based upon the determination that the plant design for the emergency diesel generator load shed/conservation circuitry met licensee requirements, Case Specific Checklist Item 7, “Resolution of Non-Safety Related Cables Going to Shunt Trip Coils,” was closed. In addition, numerous Restart Action Matrix items and Guidelines for Restart items were assessed and closed.

However, we did note multiple examples of operator configuration control issues and poor communication by engineering of operational limits. In response to these issues, we observed that your staff initiated corrective actions to address the performance issues in accordance with your corrective action program.

Based on the results of this inspection, the NRC has determined that five violations of NRC requirements occurred involving the failure to restore the residual heat removal system to the proper lineup after completion of surveillance testing, the failure to comply with Technical Specifications when two essential service water loops were inoperable in March 1997, failure to submit a licensee event report within 30 days of the discovery of an entry into Technical Specification 3.0.3 in March 2000, failure to construct the emergency diesel generator exhaust and air intake structures in accordance with the design criteria specified in the Updated Final Safety Analysis Report, and failure to perform inspections of boron injection tank manway bolts in accordance with Technical Specification 4.0.5a. These violations are being treated as Non-Cited Violations (NCV), consistent with Section VI.A.1 of the Enforcement Policy. 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, DC 20555-0001.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC Public Electronic Reading Room (PERR) link at the NRC homepage, http://www.nrc.gov/NRC/ADAMS/index.html.

Sincerely,

/RA/

John A. Grobe, Director
Division of Reactor Safety

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

Enclosure: Inspection Report 50-315/2000013 (DRP); 50-316/2000013(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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DATE	06/13/00		06/13/00		06/ /00		06/13/00		06/13/00		06/13/00	

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and air intake structures in accordance with the design criteria specified in the Updated Final Safety Analysis Report, and failure to perform inspections of boron injection tank manway bolts in accordance with Technical Specification 4.0.5a. These violations are being treated as Non-Cited Violations (NCV), consistent with Section VI.A.1 of the Enforcement Policy. 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, DC 20555-0001.

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John A. Grobe, Director
Division of Reactor Safety

Docket Nos. 50-315; 50-316
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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:	50-315/2000013(DRP); 50-316/2000013(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:	April 2, 2000 through May 27, 2000
Inspectors:	B. L. Bartlett, Senior Resident Inspector K. A. Coyne, Resident Inspector J. D. Maynen, Resident Inspector N. Shah, Region III Inspector M. Holmberg, Region III Inspector K. Green-Bates, RIII Inspector A. Dunlop, RIII Inspector
Approved by:	A. Vegel, Chief Reactor Projects Branch 6 Division of Reactor Projects

EXECUTIVE SUMMARY

D. C. Cook Units 1 and 2
NRC Inspection Report 50-315/2000013(DRP); 50-316/2000013(DRP)

This inspection included aspects of licensee operations, maintenance, engineering, and plant support. The report covers a 6-week period of resident inspection activities and includes follow-up to issues identified during previous inspection reports.

Operations

- Refueling activities for entry into Mode 6 were well controlled. Licensee supervisors exercised effective command and control and refueling activities were accomplished in a smooth and deliberate manner. Communications, foreign material exclusion, shift staffing, and Technical Specification equipment operability requirements were met. Oversight from licensee management and Performance Assurance department personnel was evident. (Section O1.2)

- The inspectors performed walkdowns of several safety-related systems that were required to be operable prior to Mode 6 (core reload). These included the residual heat removal system, essential service water system, electrical distribution, and the chemical and volume control system. The inspectors identified no conditions that would have precluded these systems from performing their safety functions. (Section O2.1)

- The licensee identified a number of degraded and non-conforming conditions which will exist at the time of restart of Unit 2. The inspectors reviewed a sample of the licensee's operability determination evaluations and compensatory actions associated with these degraded and non-conforming conditions. The inspectors determined that the licensee's evaluations and actions reviewed were adequate to support restart of Unit 2. (Section O2.2)

- The licensee and the NRC identified mis-positioned valves and other configuration control errors. The licensee promptly initiated interim corrective actions and began a formal root cause analysis to determine long term corrective actions. The corrective actions were effective as evidenced by the decrease in the number of configuration control errors following the implementation of interim corrective actions. (Section O2.3)

- A non-cited violation was identified for the failure to restore the residual heat removal system to the proper lineup after completion of surveillance testing. The licensee failed to close two valves that had been opened approximately two months earlier during containment local leak rate testing. While filling the system, approximately 200 gallons of water spilled into the east residual heat removal pump room sump through the two previously opened valves. The licensee initiated an investigation and implemented satisfactory corrective actions. The inspectors also identified that the control room operators failed to document this spill in the control room logs, contrary to management expectations. (Section O2.4)

Maintenance

- The inspectors concluded that the observed work was performed in accordance with procedures, the current revision of the appropriate procedures were in use at the work sites, and proper work safety and radiological protection practices were noted. Work items were appropriately scheduled in the plan of the day. (Section M1.1)

Engineering

- The licensee had not fully evaluated limiting conditions for the essential service water system and could not demonstrate that analysis assumptions bounded worst case accident conditions. The licensee had not fully considered the impact of expected circulating water system perturbations on the essential service water pump minimum operability limit. Additionally, following the licensee's identification of the failure to perform an adequate 10 CFR 50.59 safety evaluation to support a revision to the Technical Data Book, the licensee failed to withdraw the associated Technical Data Book revision. (The Technical Data Book was used to provide the acceptance criteria for safety-related pump quarterly inservice testing.) Because essential service water system operability was not required at the time of the inspection, the safety impact of these deficiencies was minimal. (Section E2.1)

- The inspectors identified that the licensee misapplied instrument uncertainty to the inservice testing program. The recommended acceptance criteria for several pumps, including the residual heat removal, essential service water, and spent fuel pool cooling pumps was set non-conservatively but did not result in component inoperability. (Section E2.2)

- Two non-cited violations were identified. The first involved the failure to comply with Technical Specifications when two essential service water loops were inoperable in March 1997. The second involved the failure to submit a licensee event report within thirty days of the discovery of an entry into Technical Specification 3.0.3. The licensee issued plant administrative requirements to ensure the operability of the essential service water system with one or more essential service water pumps inoperable in the opposite unit. The violations are in the licensee's corrective action program as CR 00-07856. (Section E8.1)

- Case Specific Checklist Item 7, "Resolution of Non-Safety Related Cables Going to Shunt Trip Coils," is closed based on the plant design for the emergency diesel generator load shed/conservation circuitry meeting industry standards which were existing at the time of plant licensing. Additional studies and balance-of-plant cable testing performed by the licensee provided added assurance that the design of the load shed/conservation circuitry is acceptable. (Section E8.2)

- A non-cited violation was identified for the failure to construct the emergency diesel generator exhaust and air intake structures in accordance with the design criteria specified in the Updated Final Safety Analysis Report Section 1.4.1. The licensee determined that a single tornado-generated missile could potentially damage both of the same units' emergency diesel generator air intake structures, which would result in a common mode failure of both of the emergency diesel generators. The licensee also determined that the plant was outside its design basis, and issued Licensee Event Report (LER) 50-315/99020-00. The licensee issued a design change package to place a barrier between each air intake and exhaust structure to correct the condition. The inspectors reviewed the licensee's corrective actions and concluded that the actions were adequate to support Unit 2 restart. (Section E8.3)

- A non-cited violation was identified for the failure to perform inspections of boron injection tank manway bolts in accordance with Technical Specification 4.0.5a. The bolts were removed from the licensee's inservice inspection program during the second 10-year update; however, no justification for the removal was found. The missed inspections were performed during the current outages for both units and no concerns were identified with the bolts. The licensee revised the inservice inspection program to include these missed components and revised the appropriate procedure. (Section E8.3)

- The NRC Inspection Manual Chapter 0350 panel reviewed licensee actions and inspector assessments related to selected Restart Action Matrix issues. The issues reviewed were acceptable for Unit 2 restart and were closed. (Section E8.3)

- The NRC Inspection Manual Chapter 0350 panel reviewed licensee actions and inspector assessments related to the following issues: C.1.1 Root Cause Determination, C.1.2 Corrective Action Development, C.1.3 Corrective Action Plan Implementation and Effectiveness, C.2.1 Self Assessment Capability, C.3.1 Assessment of Staff, C.3.2 Assessment of Corporate Support, C.4 Assessment of Physical Readiness of the Plant, and C.5 Assessment of Compliance with Regulatory Requirements. (Section E8.4)

Report Details

Summary of Plant Status

Unit 1 remained defueled throughout the inspection period. The licensee continued work on replacing the concrete that was removed to facilitate steam generator removal. Welding activities were also in progress.

Unit 2 entered Mode 6, Refueling, on April 10, 2000, when the first fuel assembly for the upcoming fuel cycle was placed in the reactor vessel. On April 22, 2000, Unit 2 entered Mode 5, Cold Shutdown. At the end of this report period, the licensee was making preparations to enter Mode 4, Hot Shutdown.

I. Operations

O1 Conduct of Operations

O1.1 General Comments

The inspectors conducted frequent observations of control room activities and equipment operation during the extended outage of both reactor units. Overall, plant operations were performed using approved operating procedures and reflected good operating practices. Noteworthy observations and findings are detailed in the report sections which follow.

O1.2 Refueling Observations (Unit 2)

a. Inspection Scope (71707)

The inspectors observed the licensee's preparations for entering Mode 6 (Refueling). The inspectors observed refueling activities at the spent fuel pool, inside containment and in the control room.

b. Observations and Findings

Based upon observed activities, the inspectors determined that fuel movement was well controlled. The senior reactor operator-core alterations (SRO-CA) exercised effective command and control and refueling activities were accomplished in a smooth and deliberate manner. The inspectors determined that Technical Specification (TS) requirements supporting Mode 6 were met, including residual heat removal and ventilation system operability and shift staffing requirements. The licensee maintained adequate communications during fuel movement. Foreign material exclusion was maintained in the vicinity of the spent fuel pool and reactor cavity. Management and Performance Assurance oversight were evident during refueling activities. The inspectors reviewed refueling procedures and determined that procedures adequately controlled refueling operations.

c. Conclusions

Refueling activities for entry into Mode 6 were well controlled. Licensee supervisors exercised effective command and control and refueling activities were accomplished in a smooth and deliberate manner. Communications, foreign material exclusion, shift staffing, and TS equipment operability requirements were met. Oversight from licensee management and Performance Assurance department personnel was evident.

O2 Operational Status of Facilities and Equipment

O2.1 Assessment of Readiness for Core Reload (Unit 2)

a. Inspection Scope (71707)

On April 4 through 7, 2000, the inspectors conducted walk downs of equipment and systems required to support Unit 2 core reload. The walk downs included the residual heat removal system (RHR), essential service water system (ESW), electrical distribution, and the chemical and volume control system (CVCS). Additionally, the inspectors evaluated general area housekeeping and control of transient combustible materials.

b. Observations and Findings

The inspectors identified no conditions that would have precluded these required Mode 6 systems from accomplishing their safety functions. General area housekeeping and control of transient combustible materials was adequate to support refueling activities. The inspectors identified the following discrepancies during the walk downs:

- Inappropriate storage of compressed gas cylinders: The inspectors found two small, unrestrained oxygen bottles in the auxiliary building elevator mechanical equipment room on the 667 foot elevation of the auxiliary building. The bottles were used to support Job Order (JO) C53652, “Fabricate and Install Block Wall Modifications.” The inspectors informed the shift manager, the bottles were appropriately restrained, and the licensee initiated CR 00-5265. During a followup walkdown on April 20, 2000, the inspectors noted that compressed gas bottles no longer required to support work activities were located in the vicinity of safety-related equipment. This was not in accordance with management expectations concerning the storage of compressed gas cylinders. The licensee initiated CR 00-5983 and removed gas cylinders that no longer supported ongoing work activities from the auxiliary building.

  On May 1, the licensee wrote CR 00-6307 referencing eight condition reports over the last 5 months documenting deficiencies associated with compressed gas bottle storage. The licensee instituted corrective actions including reminding personnel of the gas bottle restraint limitations, performing additional management tours, and reminding operators on rounds to report gas bottle storage issues. During subsequent routine plant walk downs, the inspectors did not identify any additional gas bottle storage deficiencies.

- Valves in Boration Flowpath Not Sealed in Position: On April 19, 2000, the inspectors performed a walkdown of the boration flowpath required by TS 3.1.2.1, “Boration Systems.” Plant Managers Procedure (PMP) 4043.SLV.001,” Sealed/Locked Valves,” Revision 0, provided requirements for the locking or sealing of manual valves in order to prevent inadvertent operation. Figure 1 to PMP 4043.SLV.00, “Sealed\Locked Valve List,” required that the West Centrifugal Charging Pump Discharge Header Shutoff Valve 2-CS-301W, and Flow Control Valve 2-QRV-251 Outlet Shutoff Valve, 2-CS-303, be sealed in the open position. The inspectors identified that, although these valves were in the open position, they were not sealed. The inspectors immediately informed the Unit Supervisor of this condition and the valve seal locking devices were installed. The licensee initiated CR 00-5810. The safety impact of the failure to seal these valves was minimal because the valves were in the required position to maintain a TS 3.1.2.1 boration flowpath. The licensee determined that the valves had been operated during system testing associated with 2-EHP.SP.DCP-269, but the equipment operator failed to install the seal locking device after the valves were opened. Contrary to the requirements of PMP 4043.SLV.001, the licensee failed to install locking devices on 2-CS-301W and 2-CS-303. This failure constitutes a violation of minor significance and is not subject to formal enforcement action.

c. Conclusions

The inspectors performed walkdowns of several safety-related systems that were required to be operable prior to Mode 6 (core reload). These included the residual heat removal system, essential service water system, electrical distribution, and the chemical and volume control system. The inspectors identified no conditions that would have precluded these systems from performing their safety functions.

O2.2 Operability Evaluations Associated With Degraded and Non-Conforming Conditions

a. Inspection Scope (37551, 62707, 71707, C.1.2.j)

The licensee identified a number of degraded and non-conforming conditions which will exist at the time of restart of Unit 2. The inspectors reviewed a sample of the licensee’s operability determination evaluations (ODEs) associated with these degraded and non-conforming conditions.

The inspectors also reviewed these items as they related to Manual Chapter 0350 Staff Guidelines for Restart Approval Item C.1.2.j, “Interim corrective actions have been developed and documented when permanent corrective action will take an excessive amount of time to implement or cannot be completed before the licensee plans to restart the facility.”

b. Observations and Findings

The inspectors reviewed the following condition reports (CRs) associated with degraded or non-conforming conditions identified by the licensee:

- CR 98-06364 Engineered Safety Features (ESF) Ventilation System Heat Gain Calculation
- CR 99-02455 Residual Heat Removal System Vibration
- CR 99-17235 Emergency Diesel Generator (D/G) Fuel Oil Filter Housing
- CR 99-17063 Spent Fuel Pit Exhaust Ventilation System Deficiencies
- CR 99-18596 D/G Fuel Pump Modification
- CR 99-27264 Unit 2 AB D/G Starting Air Tubing Failure
- CR 00-00670 Operability of Control Room Ventilation System with Essential Service Water Temperature above 65°F

b.1 Condition Report 98-06364

The Engineered Safety Feature (ESF) exhaust ventilation (AES) system was designed to provide cooling to safety related equipment located within the auxiliary building general areas and the ESF equipment rooms. In 1997, an NRC architect engineering inspection questioned the accuracy of the AES system heat gain calculation. The architect engineering team inspection finding was documented in NRC Inspection Reports 50-315/316-97201 and 50-315/316-98004. In response to the finding, the licensee revised the calculation to correct several erroneous inputs.

In 1998, the licensee determined that the revised calculation showed that several areas cooled by the AES system could exceed the design criteria of 125°F during a design basis accident. The licensee initiated Condition Report 98-6364 to document that the AES system may not be capable of meeting its design basis. This issue was documented in NRC Inspection Report 50-315/316-98021.

Subsequently, the licensee determined that the higher temperatures did not affect the environmental qualification of this equipment; therefore, the higher temperatures did not make the equipment cooled by the AES system inoperable. The licensee's conclusion was documented in NRC Inspection Report 50-315/316-98027. The inspectors reviewed the condition report and the Inspection Reports associated with this issue and determined that the licensee's actions were adequate to support Unit 2 restart.

b.2 Condition Report 99-02455

Condition Reports 99-00996, 99-02455, and 99-02466 documented excessive flow-induced vibration and cavitation of the RHR piping when the RHR system was aligned to the normal cooldown line. Although the RHR system vibration had existed for several years, the RHR piping and components had not been analyzed for potential fatigue cracking.

In March 1999, the licensee completed an operability evaluation, 91-18-ODE-355, for the RHR system which concluded that the RHR system was operable in Mode 5 (Cold

Shutdown) and Mode 6. However, in NRC Inspection Report 50-315/316-99009, the inspectors questioned the adequacy of this evaluation. Subsequently, the licensee performed dye penetrant examinations on the branch line welds which were susceptible to vibration-induced failures, conducted additional system walk downs, and reviewed the results of the most recent ten-year inservice inspection of the RHR piping. The licensee identified no existing fatigue cracks and concluded that the RHR system was operable but degraded.

The licensee reduced the system vibration by aligning the RHR system through the safety injection line as the preferred flow path for shutdown cooling. The licensee planned to develop more extensive corrective actions to address the vibration and cavitation concerns. The inspectors reviewed the actions taken to address the RHR vibration concerns and determined that the actions were adequate to support Unit 2 restart.

b.3 Condition Report 99-17063

Two Condition Reports, CR 99-16465 and CR 99-17063, documented several deficiencies that indicated the spent fuel pit exhaust ventilation system (AFX) was incapable of meeting its design function. Two concerns were of particular significance: (1) the engineering calculation for the exhaust fan pressure drop (DCCHV12FH01S) indicated that the fan motors were overloaded during normal operation, and (2) following a fuel handling accident, the AFX system response time was inadequate to prevent an unfiltered release.

The AFX system was common to both units and consisted of two 30,000 cubic feet per minute (cfm) exhaust fans which directed air through a filter housing and discharged through the Unit 1 vent stack. Normally, one of the fans was in standby, and a bypass damper routed air around the charcoal filters. During a fuel handling accident, the bypass damper was designed to reposition and direct the ventilation exhaust air through the charcoal filters before it reached the vent stack.

Licensee staff verified that the fan motors were operating within their rated power and that the calculation was in error. The fan curve used in the calculation assumed that the flow damper upstream of the fans was in the full open position; however, the actual damper position restricted air flow. Subsequently, the licensee began revising the calculation and verified that station procedures contained instructions to determine the actual motor power whenever the upstream flow damper position was changed.

To address the response time concern, the licensee initiated a compensatory action to place the charcoal filters in service whenever fuel or heavy loads were moved near the fuel pool. Additionally, a revised analysis was submitted to the NRC showing that, in the absence of this system, control room and offsite doses would remain within prescribed limits. The inspectors reviewed the licensee’s evaluation and compensatory actions and determined that the actions were adequate to support Unit 2 restart.

b.4 Condition Report 99-17235

Condition Report (CR) 99-17235 documented a repeat failure of the Unit 2 CD D/G duplex fuel oil filter housing. The original failure was documented in CR 99-10446 and was discussed in NRC Inspection Report 50-315/316-99015 (DRP):

> "On May 4, 1999, a minor maintenance activity was performed to replace incorrectly sized bolts on the Unit 2 CD emergency D/G duplex fuel oil filter housing. As the maintenance worker was tightening the bolted connection, the filter housing fractured at one of the bolt holes. Condition Report 99-10446 was written to document the event, and JO C49235 was written to replace the filter housing. The inspectors did not find any documented plan for troubleshooting the filter housing fracture; instead, the JO directed the workers to replace the filter housing. A new filter housing was installed on June 29, 1999. Several days after the new filter housing was installed on the Unit 2 CD D/G, cracks were identified around the same bolt hole where the original housing had broken. The licensee determined that the bolt hole had broken because the filter housing was not aligned properly with the support stanchion."

On October 5, 1999, the Unit 2 CD D/G filter housing was replaced. On October 7, 1999, two DCPs, 1-DCP-4339 and 2-DCP-730, were approved to realign the support stanchion and add valves for isolation of the fuel oil filter. The licensee evaluated the condition and determined that the fuel oil housing installation constituted an operable but degraded condition. The inspectors reviewed the licensee's operability determination evaluation and walked down the emergency diesel generators. The inspectors observed that the licensee had added shims to all of the filter housings to align each filter housing with its support stanchion. The inspectors concluded that the licensee's evaluation and corrective actions for this condition were adequate to support restart of Unit 2.

b.5 Condition Report 99-18596

On July 16, 1999, the licensee identified that the inspection port plugs and breather caps on four of the Unit 2 CD D/G fuel injector pumps were different than the plugs and breather caps installed on the other eight cylinders. The licensee investigated the condition and found that new inspection port plugs and breather caps were installed on the Unit 2 CD D/G as part of JO C35644 in April 1996, because like-for-like spares were not available. However, the licensee could not find any approved design change documentation of safety evaluations which supported the configuration change.

The licensee found that a similar configuration change was made to the Unit 1 CD D/G under JO C36970. An engineering evaluation for this change stated that neither the inspection port plugs nor the breather caps had any safety functions. The evaluation also stated that D/G operability was not impacted by the condition and that the D/G would remain operable even if the plugs and caps were removed. The licensee concluded that the evaluation for the Unit 1 CD D/G was also applicable to the Unit 2 CD D/G; therefore, the licensee concluded that the Unit 2 CD D/G was operable with a non-conforming condition. Deficiency tag 38142 was placed on the Unit 2 CD D/G to request that the inspection port plugs and breather caps be restored to the original design configuration. The licensee planned to perform this work after Unit 2 restart.

The inspectors reviewed the evaluation for the Unit 1 CD D/G and concluded that the evaluation was also applicable to the configuration change made on the Unit 2 CD D/G. The evaluation and corrective actions for this condition were adequate to support restart of Unit 2.

b.6 Condition Report 99-27624

During a maintenance run of the Unit 2 AB D/G on November 19, 1999, the starting air tubing to the #4 front bank failed. The licensee repaired the tubing and completed additional inspections. The licensee identified starting air lines on the other D/Gs that were bent or misaligned. The licensee initiated CR 99-27624 and Action Requests to repair the damaged starting air lines. The licensee determined that the bent or misaligned starting air tubing would not render the D/Gs inoperable. The inspectors reviewed the licensee's evaluation and proposed corrective actions and determined that these actions were adequate to support the restart of Unit 2.

b.7 Condition Report 00-0670:

Technical Specification 4.7.5.1.a, "Control Room Emergency Ventilation System," required that the control room air temperature be maintained less than 95°F. The control room air handling unit (AHU) cooling coils can be supplied cooling by either of two cooling supplies; (1) ESW directly, or (2) chilled water from the non safety-related mechanical refrigeration chiller. The heat sink for the mechanical chiller units was the ESW system. The licensee determined that when the ESW supply temperature exceeded 65°F, the control room ventilation system was incapable of maintaining control room temperature less than 95°F without reliance upon the nonsafety-related mechanical chiller units.

The licensee determined that the original licensing and design basis for the control room ventilation credited the capability for the non safety-related chillers for all accident scenarios except a seismic event. The chillers were originally designed to Seismic Class 3 standards and therefore could not be credited following a design basis earthquake. DCP-275, "Control Room Air Conditioning Upgrade," improved the seismic capability of the chiller units in accordance with the Seismic Qualification Utility Group (SQUG) Generic Implementation Procedure (GIP). The GIP provided a technical approach, generic procedures, and documentation requirements which could be used by utilities to verify the seismic adequacy of the equipment needed for plant safe shutdown following a safe shutdown earthquake. In a February 3, 2000 safety evaluation report, the NRC staff concluded that the licensee had met the purpose and intent of the criteria in the GIP. The licensee concluded that the application of the SQUG methodology to the control room ventilation mechanical chillers allowed the demonstration of the seismic capability of the chiller units.

The licensee stated that the control room mechanical chillers will be placed in the 10 CFR 50.65 maintenance rule program and the Generic Letter 89-13, "Service Water

System Problems Affecting Safety Related Systems," program. The inspectors concluded that the operability determination adequately supported the restart of Unit 2.

c. Conclusions

The licensee identified a number of degraded and non-conforming conditions which will exist at the time of restart of Unit 2. The inspectors reviewed a sample of the licensee's operability determination evaluations and compensatory actions associated with these degraded and non-conforming conditions. The inspectors determined that the licensee's evaluations and actions reviewed were adequate to support restart of Unit 2.

The inspectors also reviewed these items as they related to Manual Chapter 0350 Staff Guidelines for Restart Approval Item C.1.2.j, "Interim corrective actions have been developed and documented when permanent corrective action will take an excessive amount of time to implement or cannot be completed before the licensee plans to restart the facility." The inspectors concluded that interim corrective actions were effectively evaluated and documented in the condition reports and operability determinations reviewed.

O2.3 Performance Issues Related to Configuration Control

a. Inspection Scope

During this inspection report period, the licensee identified several configuration control issues. The inspectors had identified configuration control issues previously in NRC Inspection Report 50-315/316-99022. Due to the anticipated changes in system status as additional systems are restored to service, the inspectors followed up to ensure adequate licensee assessment and corrective actions.

b. Observations and Findings

Following the occurrence of a number of configuration control issues, the licensee began a root cause assessment to determine necessary corrective actions. Each of the identified issues was documented in a condition report. The issues were:

- Contaminated water spilled on the floor of the RHR pump room (CR 00-5356 issued April 9, 2000). During the fill and vent of the recirculation sump suction line, approximately 200 gallons of slightly contaminated water was spilled to the floor of the RHR pump room. This issue is discussed in additional detail in Section O2.4, below.

- Three CVCS valves found not sealed (CR 00-5810 issued April 19, 2000). Three CVCS valves were identified as being in their correct positions but were not sealed even though called for by procedure. Two of the valves were identified by NRC inspectors. This issue was discussed previously in Section O2.1 of this inspection report.

- Reactor coolant pump seal return flow path not properly aligned (CR 00-6165 issued April 27, 2000). Two seal return containment isolation valves were in a closed position instead of open. This resulted in the lifting of the seal return safety valve when the charging system was used to commence reactor coolant pressurization. The relief valve discharged to the pressurizer relief tank and the water was contained.

- Letdown valves found not open (CR 00-6164 issued April 27, 2000). During the performance of an RCS fill and vent procedure, the operating crew determined that the three CVCS letdown isolation valves were in the closed position instead of the expected open position.

Following the identification of the mis-positioned seal return valves and the mis-positioned letdown isolation valves, the licensee formed a rapid event response team and wrote a level 2 CR (required a full root cause analysis). In addition, the licensee re-verified valve and system lineups prior to continuing with plant evolutions. Additional prompt corrective actions included the briefing of all operating crews on the configuration control issues and the stressing of the importance of procedural adherence.

Interim corrective action included the issuance of Operations Standing Order SO-2000-0004, Revision 0, regarding configuration control on May 5, 2000. The standing order included corrective measures such as:

- Requiring the mark up of flow prints prior to performing evolutions,

- Requiring additional reviews of abnormal position control logs,

- Performing walk down of systems immediately after re-energizing or filling the system with fluid,

- Placement of additional controls on procedures in use, to ensure that the status is appropriately maintained at all times, and

- Maintaining an up-to-date listing of systems that had been turned over to operations department personnel.

The standing order was to remain in place while the formal root cause analysis was completed and long term corrective actions were implemented.

At the end of the report period, the root cause analysis had not been completed. However, the inspectors had noted a decrease in the occurrence of configuration control issues following the implementation of the interim corrective actions. The interim corrective actions appeared effective in reducing configuration control errors.

c. <u>Conclusions</u>

The licensee and the NRC identified mis-positioned valves and other configuration control errors. The licensee promptly initiated interim corrective actions and began a formal root cause analysis to determine long term corrective actions. The corrective actions were effective as evidenced by the decrease in the number of configuration control errors following the implementation of interim corrective actions.

O2.4 Inadequate Surveillance Test Restoration Results in Leak in Residual Heat Removal Pump Room (Unit 2)

a. Inspection Scope (61726, 71707)

On April 8, 2000, the licensee performed 02-Operations Head Procedure (OHP) 4021.008.001, Attachment 6, “Filling the Recirc Sump Lines,” Revision 7, in order to refill the east containment recirculation sump suction line. This line had been previously drained to support maintenance activities and testing of ICM-305, “Recirculation Sump to East RHR/CTS [containment spray] Pumps Suction pursuant to 10 CFR 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.” During the refill procedure, approximately 200 gallons of Refueling Water Storage Tank (RWST) water leaked through two open vent and drain connections into the east RHR pump room sump.

b. Observations and Findings

Appendix J testing of 2-ICM-305 was performed on February 18, 2000 in accordance with procedure 02-EHP [engineering head procedure] 4030.STP [surveillance test procedure].203, “Type B and C Leak Rate Testing.” Procedure 02-EHP 4030.STP.203 required 2-RH-102E, “2-ICM-305 Downstream Test Connection Shutoff Valve,” and 2-RH-103E, “Recirculation Sump to East RHR Pump PP-35E Suction Header Drain Valve,” to be opened during leak rate testing. These valves should have been closed following the testing in accordance with Step 5.2.14 of 02-EHP 4030.STP.203, which required performance of a post-test valve lineup. Due to a potential conflict with other work activities, the licensee did not immediately perform the post-test valve lineup following completion of the local leak rate testing. On April 8, 2000, the licensee failed to identify the need to perform the post-test valve lineup prior to initiating refill of the containment sump suction line.

The procedure for suction line refill, 02-OHP 4021.008.001, Attachment 6, did not specifically address the required positions of 2-RH-102E and 2-RH-103E. The inspectors noted that Procedure 02-OHP 4021.008.002, “Placing the Emergency Core Cooling System in Standby Readiness,” required operators to verify that 2-RH-102E and 2-RH-103E were closed prior to entry into Mode 4. However, the required lineups of Procedure 02-OHP 4021.008.002 had not been completed prior to the refill of the 2-ICM-305 suction line.

During the refill evolution, after the RWST was aligned to refill the recirculation sump suction lines, water spilled from valves 2-RH-102E and 2-RH-103E. The water spill resulted an increase in the east RHR pump room sump and actuation of a control room annunciator due to high sump level in the east RHR pump room. The licensee stopped the refill procedure and determined that approximately 200 gallons of RWST water had leaked into the east RHR pump room. The cause of the water leakage was determined to be the failure to close valves 2-RH-102E and 2-RH-103E prior to initiating recirculation sump suction line refill.

Technical Specification 6.8.1 required, in part, that written procedures shall be established, implemented and maintained covering the applicable procedures recommended in Appendix A of Regulatory Guide (RG) 1.33, Revision 2, February 1978. Regulatory Guide 1.33, “Quality Assurance Program Requirements (Operation),” Revision 2, February 1978, Appendix A, recommended, in part, that procedures be written for containment local leak rate detection testing. Procedure 02-EHP 4030.STP.203 provided instructions on the conduct of containment local leak rate detection testing and PMP 4030.EXE.001, “Conduct of Surveillance Testing,” provided instructions on the conduct of surveillance testing. Section 3.4.4 of PMP 4030.EXE.001 stated that the Unit Supervisor or Shift Manager shall ensure that system or component conditions that were altered for the surveillance test were restored to normal conditions. Contrary to the above, the Unit Supervisor failed to ensure that the RHR system valve lineup was restored to normal conditions in accordance with Step 5.2.14 of 02-EHP 4030.STP.203 following surveillance testing conducted on February 18, 2000. The inspectors determined that the failure to follow plant procedural requirements was a Violation of TS 6.8.1. This Severity Level IV violation is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee’s corrective action program as CR 00-5356 (NCV 50-315/316-2000013-01).

The inspectors reviewed the control room logs and determined that a log entry had not been made to document the spill in the RHR pump room. The inspectors concluded that the failure to log this unexpected system transient did not meet licensee expectations for control room log keeping.

c. Conclusions

A non-cited violation was identified for the failure to restore the residual heat removal system to the proper lineup after completion of surveillance testing. The licensee failed to close two valves that had been opened approximately two months earlier during containment local leak rate testing. While filling the system, approximately 200 gallons of water spilled into the east residual heat removal pump room sump through the two previously opened valves. The licensee initiated an investigation and identified satisfactory corrective actions. This violation is in the licensee’s corrective action program as CR 00-5356. The inspectors also identified that the control room operators failed to document this spill in the control room logs, contrary to management expectations.

O8 Miscellaneous Operations Issues

O8.1 (Closed) Inspection Followup Item 50-315/99001-01: “Residual heat removal system vibration.”

Residual heat removal system vibration had been previously discussed in NRC Inspection Report 50-315/316-99022, Section E2.2.b.1. As discussed in

Section O2.2.b.2 above, the inspectors have concluded that the licensee's operability determination supporting RHR system operability is adequate to support Unit 2 restart. Additionally, the inspectors have performed several walk downs of RHR system components with the system in various operating configurations and concluded that the licensee's corrective actions have reduced RHR system vibration. Based upon these system observations and review of the licensee's corrective actions and operability determination for RHR system vibration, IFI 50-315/99001-01 is considered closed.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

a. Inspection Scope (62707)

The inspectors observed all or portions of the following maintenance activities and reviewed associated documentation:

- C0054758, Modify Containment Heating Ventilation and Air Conditioning Structural Supports
- C0056914, Repair Fitting Leak on Speed Increaser to the Unit 2 East Charging Pump
- C0044722, Modify Unit 2 Auxiliary Building Large Bore Pipe Supports
- C0053652, Fabricate and Install Block Wall Modifications
- 02 OHP-4030.STP.022W, West Essential Service Water System Test
- 12 MHP 5040.010.003, Ice Condenser Support Activities (Temporary Conditions)

b. Observations and Findings

The inspectors concluded that the observed work was performed in accordance with procedures, the current revision of the appropriate procedures were in use at the work sites, and proper work safety and radiological protection practices were noted. Work items were appropriately scheduled in the plan of the day.

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 Review of Essential Service Water Pump Minimum Operability Limits

a. Inspection Scope (37751,61726)

In order to resolve concerns associated with operability of opposite unit ESW pumps when the ESW system was operated in a cross tied configuration, the licensee issued an administrative technical requirement (ATR). The ATR provided guidance to ensure ESW system operability with one or more ESW pumps inoperable and is discussed in more detail in Section E8.1 below. Under conditions allowed by the ATR, mitigation of a design basis loss of coolant accident (LOCA) could rely on a single ESW pump. The licensee revised the ESW system analysis to demonstrate the capability of a single ESW pump to mitigate a design basis LOCA. The inspectors reviewed the revised analysis and the implementation of operational restrictions.

b. Observations and Findings

The inspectors reviewed calculation MD-02-ESW-77-N, "U2 ESW System Analysis For As Left 3/29/00 Flow Balance Conditions to Determine The Allowable Minimum Operability Requirements." The purpose of calculation MD-02-ESW-77-N was to determine appropriate test acceptance criteria for quarterly inservice testing based on the as-left condition of the ESW system following flow balance testing. Section E2.2 of NRC Inspection Report 50-305/316-2000001 described several concerns raised by the inspectors associated with the acceptance criteria established for ESW system flow balance testing.

b.1 ESW System Analysis Calculation Did Not Fully Consider Expected System Operational Perturbations

The assumptions used in the revised calculation differed from the previous ESW system calculation. Examples of changes from previous analyses included:

- ESW pump speed was reduced to accommodate the effects of degraded D/G voltage and frequency.

- ESW pump suction pressure was reduced to a pressure commensurate with a one foot screenhouse water level drawdown. This amount of drawdown was expected with three running circulating water pumps (assumed to be in the non-accident unit) and all three lake water intake tunnels available.

- The ESW supply to the auxiliary feedwater (AFW) pumps was reduced by approximately 200 gpm. Updated Final Safety Analysis Report Table 9.8-5, "Essential Service Water System Minimum Flow Requirements Per Train," specified that 450 gpm of ESW flow would be supplied to the AFW pumps during a LOCA. The licensee used a value of approximately 250 gpm based on the capability of the condensate storage tank to provide a AFW suction source during the initial phase of an accident.

Calculation MD-02-ESW-77-N indicated that the ESW minimum operability limit needed to be adjusted upward from 61.8 psid to 62.3 psid. The inspectors questioned whether the assumptions used in the calculation bounded expected operational conditions. The licensee's analysis assumed a loss of offsite power (LOOP) was the most limiting condition. The inspectors questioned if continuity of offsite power could represent a more severe challenge to ESW pump operability than LOOP conditions. The availability of offsite power following an accident would allow continued operation of the accident unit's circulating water pumps. The additional flow from the circulating water system would lower screenhouse level (resulting in a lower suction pressure for the ESW pumps) and raise the discharge vault level (causing a higher backpressure on the ESW system).

The inspectors reviewed permissible circulating water system configurations to verify consistency with the ESW system analysis assumptions. Normal operating procedures for operation of the circulating water system allow isolation of one of the three available intakes for deicing and zebra mussel cleanup activities. Isolation of one of the intake tunnels could result in over a doubling of the expected screenhouse level drawdown and an associated reduction in ESW pump suction pressure. Therefore, the assumption of a one foot drawdown in screenhouse level may not have bounded all expected operational configurations. The inspectors concluded that the licensee had not fully considered the impact of expected system perturbations associated with these activities on the ESW pump minimum operability limit. The licensee subsequently re-performed the ESW system analysis to determine appropriate limiting conditions for system operation.

b.2 <u>Failure to Withdraw Revision to Technical Data Book After Identification of Documentation Error</u>

Technical Data Book (TDB) Figure 2-15.1, "Safety-Related Pump In-service Test Hydraulic Reference," provided acceptance criteria for quarterly in-service testing of safety-related pumps. Revision 43 to the TDB was issued on May 16, 2000 to reflect the revised ESW pump minimum operability limit. Although calculation MD-02-ESW-77-N provided the basis for the TDB revision, the accompanying 10 CFR 50.59 safety evaluation failed to consider the reduction in ESW supply to the AFW pumps. After the TDB revision was issued, the licensee identified the failure to adequately perform the 10 CFR 50.59 evaluation. The licensee immediately withdrew the safety screening for TDB, Revision 43, but failed to withdrawal the actual revision to the TDB. In-service testing of both the Unit 2 east and west ESW pumps was conducted using the acceptance criteria specified in Revision 43 to the TDB. The licensee wrote CR 00-7148 and CR 00-7098 to document the failure to perform an adequate 10 CFR 50.59 evaluation and the failure to withdraw Revision 43 from the TDB. ESW system operability was not required at the time of the inspection; therefore, the safety impact of this failure to maintain adequate control over Revision 43 to the TDB was minimal.

c. Conclusions

The licensee had not fully evaluated limiting conditions for the essential service water system and could not demonstrate that analysis assumptions bounded worst case accident conditions. The licensee had not fully considered the impact of expected circulating water system perturbations on the essential service water pump minimum operability limit. Additionally, following the licensee's identification of the failure to perform an adequate 10 CFR 50.59 safety evaluation to support a revision to the Technical Data Book, the licensee failed to withdraw the associated Technical Data Book revision. (The Technical Data Book was used to provide the acceptance criteria for safety-related pump quarterly inservice testing.) Because essential service water system operability was not required at the time of the inspection, the safety impact of these deficiencies was minimal.

E2.2 Incorrect Instrument Uncertainty Correction Applied to Inservice Testing (IST) Acceptance Criteria

a. Inspection Scope

The licensee determined that twenty safety-related centrifugal pumps were design limited. The licensee considered a pump to be design limited if the pump was unable to meet safety analysis minimum flow requirements with the maximum degradation allowed by the American Society of Mechanical Engineers (ASME) Code. The safety-related pumps limited by design requirements included the component cooling water (CCW), ESW, spent fuel pool cooling, safety injection, and RHR pumps. The acceptance criteria applied to design limited pumps during quarterly inservice pump testing must reflect safety analysis requirements. Additionally, the acceptance criteria must be set higher than the minimum safety analysis requirement to ensure that test instrument inaccuracies do not result in a pump being inappropriately declared operable. As discussed in NRC Inspection Report 50-315/316-99021, Section E2.1.b.2, the licensee had not consistently applied an instrument uncertainty correction to minimum operability limits for safety-related pumps in the past.

On April 1, 2000, the licensee issued Revision 4 to Procedure 12-EHP 5070.ISI.017R, "Section XI Centrifugal Pump Performance Verification," to include the effect of instrument uncertainty on IST acceptance criteria for design limited pumps. The revised test acceptance criteria specified in 12-EHP 5070.ISI.017R, raised the low action for design limited pumps to account for the effects of potential non-conservative instrument inaccuracies. The inspectors reviewed the licensee's methodology for determining the impact of instrument uncertainty upon test acceptance criteria.

b. Observations and Findings

The licensee performed in-service testing in accordance with TS 4.0.5 to verify pump operability and monitor pump degradation. In general, IST on safety-related centrifugal pumps was conducted by establishing a reference flowrate and measuring pump differential pressure. The measured differential pressure was compared to the IST low action limits to determine if the pump was capable of meeting safety-analysis assumptions. The licensee determined pump differential pressure by subtracting the pump suction pressure from the pump discharge pressure. Although the ASME code allows an instrument inaccuracy of up to 2 percent of gauge range, the licensee calibrated gauges used for IST testing to an accuracy of 1 percent or less.

The licensee's instrument uncertainty methodology combined the suction and discharge pressure gauge uncertainty using the square root sum of squares (SRSS) method. Rather than converting the gauge percent uncertainty to a pressure error, the licensee directly combined the suction and discharge gauge error using the SRSS method. The licensee calculated this correction by considering one percent error on the suction and discharge gauges. The licensee concluded that the appropriate instrument error to consider was 1.4 percent of the minimum operability limit obtained from the safety analysis. The recommended IST acceptance criteria for design limited pumps was then specified as the minimum pump operability limit determined by the system safety analysis adjusted upward by 1.4 percent to account for gauge error.

The inspectors determined that the licensee's methodology did not correctly consider the impact of gauge error on measured test results. Specifically, the inspectors identified two weaknesses in the licensee's methodology:

- The method weighted the impact of suction and discharge pressure gauge error equally, without considering the actual pressure measurement error associated with each gauge. Because the range of the suction pressure gauge was generally less than the range of discharge pressure gauge, one percent error represented a larger value for the discharge pressure measurement than the suction pressure measurement. Furthermore, the measured suction parameter for the boric acid transfer pumps and the essential service water pumps was liquid level rather than pressure. Suction pressure was calculated from liquid level for these pumps. Consequently, for these pumps, the suction and discharge instruments used for the IST testing were not measuring the same property, and therefore directly combining the associated gauge error using the SRSS method was inappropriate.

- The methodology did not consider the full gauge error. Because the gauges used for IST testing were generally calibrated on a percent of gauge range basis, application of the of the instrument uncertainty correction to the minimum operability limit underestimated the measurement error.

The inspectors reviewed the recommended acceptance criteria for several safety-related pumps and determined that procedure 12-EHP 5070.ISI.017R recommended non-conservative acceptance criteria. For example, the acceptance criteria for Procedure 02 OHP 4030.STP.54E(W), "East (West) Residual Heat Removal Train Operability Test - Shutdown," was specified as 142.7 psid, based upon adjusting the RHR pump minimum operability limit upward by 1.4 percent. The inspectors determined that a proper application of instrument error would have resulted in a test acceptance criteria of 149.2 psid. Although the inspectors determined that several pumps had non-conservative acceptance criteria, including the RHR, spent fuel pool cooling, and ESW pumps; the licensee's methodology could also result in overly conservative acceptance criteria. For example, the error correction for the centrifugal charging was specified as 33.3 psid; however, a proper application of the licensee's methodology would have resulted in a total error of 30 psid. The inspectors concluded that the licensee's methodology did not ensure that minimum operability limits were appropriately reflected in the recommended test acceptance criteria.

The licensee wrote CR 00-6123 on April 27, 2000 to document the misapplication of instrument uncertainty. The inspectors reviewed recent surveillance testing data for the residual heat removal pumps and determined that the pumps remained operable, even when the full effect of instrument uncertainty was considered. During the evaluation of CR 00-6123, the licensee evaluated the other design limited pumps required in Mode 5 and determined that the instrument uncertainty methodology error did not result in any pump inappropriately being declared operable.

c. Conclusions

The inspectors identified that the licensee misapplied instrument uncertainty to the inservice testing program. The recommended acceptance criteria for several pumps, including the residual heat removal, essential service water, and spent fuel pool cooling pumps was set non-conservatively, but did not result in component inoperability.

E8 Miscellaneous Engineering Issues

E8.1 Failure to Recognize Conditions Resulting in Inoperability of Essential Service System

(Closed) URI 50-315/316-99021-02: Review of TS requirements for ESW operability during design basis accident.

a. Inspection Scope (37751, 71707)

The licensee evaluated the impact of inoperable ESW pumps on overall ESW system operability and issued additional administrative controls to ensure that ESW system TS operability was maintained when ESW pumps were inoperable. The inspectors reviewed the implementation of these administrative controls and the operability impact of the past practice of maintaining the ESW system cross-tied when ESW pumps become inoperable.

b. Observations and Findings

The ESW system consisted of two independent headers, with each header served by two pumps. Each independent header served both units, and was supplied by a Unit 1 pump and a Unit 2 pump. The Unit 1 and Unit 2 portions of the ESW headers could be split by shutting either of motor operated two cross-tie valves. Technical Specification 3.7.4.1.a, "Essential Service Water," required at least two independent ESW loops to be operable when a unit was in Modes 1 (Power Operation) through 4 (Hot Shutdown), but was not specific about what constituted an operable ESW loop. In the past, the licensee had interpreted that TS 3.7.4.1.a required one operable ESW pump in the associated unit for each service water loop considered operable, regardless of the status of the opposite unit ESW pumps. As discussed in NRC Inspection Report 50-315/316-990021, Section E.2.1.b.6, the inspectors identified that the licensee's implementation of TS 3.7.4.1 operability requirements may not have provided adequate assurance that design basis and safety analysis assumptions were met.

Essential service water pump minimum operability limits had previously been based on calculation, NEMP940921AF, Revision 1, "CCW Hx [heat exchanger] Flow Multiplier." This calculation modeled the ESW system with two ESW pumps supplying cooling water to a single cross connected system header. Therefore, based on this modeled configuration, an ESW pump located in the opposite unit would be required to meet the design basis safety analysis assumptions. This configuration was consistent with Section 9.8.3.2 of the UFSAR, which stated that two ESW pumps were sufficient to supply all service water requirements for unit operation, shutdown, refueling, or post accident operation.

b.1 Evaluation and Corrective Actions to Ensure ESW System Operability With ESW Pumps Out-of-Service

In July 1999, the licensee identified an industry issue related to the impact of portions of the service water system located in a shutdown unit on the operability of service water to an operating unit. The licensee wrote CR 99-17580 to document this operating experience issue, which was originally identified in December 1990, at another licensed power reactor. During the evaluation for CR 99-17580, the licensee determined that during a design basis accident, the failure of an ESW pump in a non-accident unit could result in diversion of ESW cooling flow to the non-accident unit. This condition could result in the inability of the ESW system to satisfy its required safety function in the accident unit. The licensee concluded that the failure to close the ESW header cross-tie valves to isolate a failed or inoperable ESW pump would result in the associated ESW loop becoming inoperable for both units.

The licensee has concluded TS 3.7.4.1 requirements did not ensure ESW system operability with one or more ESW pumps inoperable. In accordance with NRC Administrative Letter 98-10, "Dispositioning of TSs That Are Insufficient to Assure Plant Safety," the licensee issued Administrative Technical Requirement 1(2)-ESW-1, "Essential Service Water," to provide interim guidance ensure ESW system operability until a TS amendment was issued to clarify TS 3.7.4.1.a operability requirements. Administrative Technical Requirement 1(2)-ESW-1 provided direction to shut ESW header cross-tie valves in the event that an ESW pump becomes inoperable. The NRC held a public meeting on May 16, 2000, with the licensee to discuss this issue and assess the adequacy of the licensee's resolution of this issue. The NRC staff concluded that the ATR provided adequate controls to ensure ESW system operability for Unit 2 restart.

b.2 Past ESW System Operating Practices

The inspectors reviewed completed job orders and equipment clearances to determine if the licensee had previously placed the plant in a configuration with the ESW system cross connected with both ESW pumps in a unit unavailable. A preventative maintenance recurring task for inspection and cleaning of the ESW pump screenhouse bays required both ESW pumps in a unit to be removed from service during plant refueling outages. The last time this recurring task was performed when TS 3.7.4.1.a

ESW system operability was required was March 1997. This maintenance was performed in accordance with under JO R53231-13 and clearance number 1970280. Although clearance 1970280 included contingency plans to restore the Unit 1 ESW system to service within one hour, the clearance also specified that the Unit 1 portion of the ESW system be cross-tied with Unit 2. Unit 2 was in Mode 3 (Hot Standby) during this time, and consequently two operable ESW loops were required by TS 3.7.4.1.a. Because the licensee failed to shut the associated cross tie valves with the Unit 1 pumps removed from service, both ESW loops for Unit 2 were inoperable while clearance 1970280 was in effect. The licensee did not recognize the operability impact of Unit 1 ESW pumps on Unit 2 portion of the ESW system, and therefore failed to enter the applicable action statements for TS 3.7.4.1.a and TS 3.0.3. Technical Specification 3.0.3 required that Unit 2 be placed in hot shutdown (Mode 4) within 6 hours. Contrary to the above, Unit 2 remained in Mode 3 with both ESW loops inoperable. The failure to comply with TS 3.7.4.1 requirements to maintain Unit 2 ESW loop operability represented continued plant operation under a condition prohibited by TSs. The inspectors determined that since the capability of the Unit 2 ESW pumps was maintained to provide accident mitigation, and since Unit 2 was shutdown, the safety significance of the failure to comply with TS 3.0.3 was minimal. Consequently, this Severity Level IV violation is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as CR 00-7560 (NCV 50-315/316-2000013-02).

b.3 <u>Failure to Report Condition Prohibited by TSs</u>

The licensee completed the reportability evaluation for CR 99-17580 on March 21, 2000, and concluded that, although the ESW system had been operated with ESW pumps out- of- service with the cross-tie valves open, this condition was not reportable under 10 CFR 50.73, "Licensee Event Report System." The licensee did not consider past operation with pumps out of service and cross-tie valves open to be reportable as either an unanalyzed condition that significantly compromised plant safety or a condition outside the design basis.

The inspectors determined that the licensee's reportability evaluation was narrowly focused and did not evaluate the reportability under 10 CFR 50.73(a)(2)(i)(B) as a condition prohibited by the plant's TSs. The inspectors concluded that past operation of the ESW system with both ESW pumps in a unit out of service with the cross ties opened would have resulted in both ESW loops being inoperable and therefore would have required entry into TS 3.0.3. NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73," Rev. 1 Section 3.2.2 "TS Prohibited Operation or Condition," that entry into TS 3.0.3 for any reason or justification is reportable under 10 CFR 50.73.

10 CFR 50.73 required, in part, that the licensee shall submit a licensee event report within 30 days after discovery of any condition prohibited by the plant's TSs. Technical Specification 3.7.4.1.a required that two independent service water loops be operable when the unit is in Mode 3. During the period of March 14 through 16, 1997, Unit 2 remained in Mode 3 with both service water loops inoperable. Contrary to the 10 CFR 50.73 requirements, the licensee failed to submit an LER for a condition prohibited by the TSs 3.7.4.1.a and 3.0.3. This Severity Level IV violation is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A.1 of the NRC

Enforcement Policy. This violation is in the licensee's corrective action program as CR 00-07856 (NCV 50-315/316-2000013-03). URI 50-315/316-99021-02 is closed.

The licensee's reportability conclusion was based, in part, on the erroneous assumption that credit for manual operator action could be applied for ESW system operability. Although manual action credit for determining plant safety impact may have been warranted, crediting manual actions to maintain TS operability was inappropriate. Because the licensee's past operability and reportability determination was based on the capability of manual operator actions to restore the ESW system configuration, the inspectors reviewed the licensee's bases for application of manual actions and ESW system procedural guidance. The inspectors concluded that for ESW system operation guidance lacked specific direction to shut the ESW cross-tie valves to prevent flow diversion in the event of an ESW pump failure. For example, Annunciator Response Procedure 01(02)-OHP 4024..104(204), "Essential Service Water and Component Cooling," drops 75 and 76, "ESW Pump Start From Unit 2(1) SI" did not include specific direction to close the ESW cross-tie valves upon failure of an ESW pump. Emergency Operating Procedure 02-OHP 4023.E-0, "Reactor Trip or Safety Injection," Revision 12, Step 11 required operators to verify that the ESW cross tie valves were open. Procedure 02-OHP- 4022.019.001, "ESW System Rupture," Revision 0 included instruction to shut cross-tie valves but would only be entered if operators suspected an ESW piping failure.

c. <u>Conclusions</u>

Two non-cited violations were identified. The first involved the failure to comply with TSs when two essential service water loops were inoperable in March 1997. The second involved the failure to submit a licensee event report within thirty days of the discovery of an entry into TS 3.0.3. The licensee issued plant administrative requirements to ensure the operability of the essential service water system with one or more essential service water pumps inoperable in the opposite unit.

E8.2 <u>Case Specific Checklist Item 7, "Resolution of Non-Safety Related Cables Going to Shunt Trip Coils," Restart Action Matrix Item R.2.7.5, and LER 50-315/98016-02, "Non-Safety Related Cables Routed to Safety Related Equipment,"</u>

a. <u>Inspection Scope (37551)</u>

By NRC letter dated September 17, 1999, the NRC transmitted the updated Case Specific Checklist (CSC) for the D. C. Cook Nuclear Plant, which identified specific issues requiring resolution prior to restart. The inspectors focused on the licensee's corrective actions for resolution of Case Specific Checklist Item 7, "Resolution of Non-Safety Related Cables Going to Shunt Trip Coils."

The standard applied to evaluate the acceptability for resolution of these items was that described in paragraphs C.1.1 "Root Cause Determination," C.1.2 "Corrective Action Development," and C.1.3 "Corrective Action Plan Implementation and Effectiveness," of Enclosure (2) of the NRC letter transmitting the CSC.

The inspectors discussed this issue in NRC Inspection Report 50-315/316-99032 and determined that licensing commitments related to non-safety related cables in the load shed circuitry appeared inconsistent and that electrical separation documents were not clear or concise and contained what appeared to be conflicting or insufficient information. Region III requested and received NRR's assistance in evaluating this concern to determine if the existing cable configuration is acceptable. The result of the evaluation is described below.

b. Observations and Findings

Background

On March 23, 1998, with Unit 1 and Unit 2 in cold shutdown, a licensee system engineer determined that the control cables that are used to shed non-safety loads from both safety-related buses had been classified during initial plant design as non-safety related. Therefore, these control cables used for emergency diesel generator load shedding are installed in balance-of-plant trays without physical separation. These control cables perform the safety function of load shedding non-safety loads from safety-buses on loss of offsite power and on loss of offsite power with a safeguards initiation signal. Load shedding of non-safety loads is required because the emergency diesel generators can not start or carry all of the loads that are normally on the safety buses. The safety buses carry both safety and non-safety loads. The concern was that since the control cables that shed non-safety loads for each safety train's emergency diesel generator are run next to each other without physical separation, a fault in one control cable might propagate into the control cables of the opposite train. If enough non-safety loads are not shed, both trains of emergency diesel generator power could be degraded.

The inspectors and NRR personnel evaluated the following issues to resolve this concern.

The D. C. Cook licensing basis with respect to designating load-shed cables (used to perform safety-related functions) as non-safety related and routing them in common balance-of-plant trays without physical separation.

During the design and licensing of the D. C. Cook plant, the licensee classified the load-shed circuitry (cables) associated with non-safety loads as non-1E since these cables did not serve safety-related loads. At the time of the D. C. Cook licensing, industry standards such as IEEE 279 and 308, pertaining to the design of the protection systems and the emergency power systems were just being developed. As documented in the Safety Evaluation of the D. C. Cook, Units 1 and 2, dated September 10, 1973, IEEE 308 served as the primary bases for judging the acceptability of the emergency power systems and IEEE-279 was used for evaluating the reactor protection and control system. The design of the cables under discussion was consistent with the requirements of IEEE-308 and was made before and without the benefit of subsequent standards such as IEEE-384 and 379, which further developed separation and single-failure criteria for Class 1E systems and associated circuits. The D.C. Cook plant design allows control cables for the non-1E loads to be installed in common trays with other balance-of-plant cables.

The licensee met the requirements of the IEEE 308 standard at the time in establishing the control cables for the load-shed circuitry for the balance-of-plant loads powered from the safety buses to be non-safety related. Therefore, D. C. Cook licensing basis with respect to designating load-shed cables as non-safety related and routing them in common balance-of-plant trays without physical separation was an acceptable design approach and was consistent with the implementation of existing standards.

<u>The acceptability of the cables in meeting the current configuration to meet the licensing basis.</u>

During the design and licensing of the D. C. Cook plant, the licensee classified the load-shed circuitry (cables) associated with non-safety loads as non-1E since these cables did not serve safety-related loads. According to the licensee, the control circuits for the associated non-safety related loads are run in raceways designed to meet Class 1E standards. The licensee determined that the non-1E load-shed control cables carry very low current relative to their thermal rating and; therefore, do not increase cable temperature appreciably over ambient temperature; the cable jackets are made of fire-retardant materials such that in the unlikely event of a short, the cable jackets will not readily provide a combustible material; and non-1E power and control cables are routed in separate trays with the same physical separation as exists between Class 1E power and control cable trays. Further, the licensee took into consideration that the non-safety related control cables under discussion were protected from shorts by redundant safety grade isolation devices, thus, isolating the fault before it could damage other cables in the same tray even with a single failure.

The design of load-shed cables rely on electrical protective devices and similar cable construction as Class 1E to support circuit independence. In addition, the licensee has performed the cable separation study, the diesel generator reliability study, and the cable testing program to support the conclusion that the absence of physical separation for the load shed/conservation control cables for the balance-of-plant loads will not significantly degrade the performance of the emergency power systems. Therefore, the balance-of-plant load shed/conservation cables in their current configuration meet the D. C. Cook plant licensing basis and are acceptable.

<u>Whether fault tests performed were adequate to support safe plant restart with the present cable routing configuration.</u>

Although the cable routing configuration meets the licensing basis, the licensee performed a cable testing program to explore the effects of sustained high currents in the control cabling to address the concern that a faulted balance-of-plant control cable associated with load-shed circuitry will not propagate to adjacent control cables in the same tray and degrade their function. A total of 11 tests were conducted on cables obtained from stores and installed spares that were representative of actual plant configurations used in the load shed/conservation circuit cabling. Results of this testing program demonstrated that for sustained currents in excess of the control cable conductor ratings, the adjacent cables were not significantly affected. On the basis of these tests, the licensee concluded that the control cable, when subjected to extreme overload (fault condition), produces excessive smoke, but does not damage proximate cables within the same raceway. The NRC staff determined that the testing program undertaken by the licensee provided added assurance that the balance-of-plant load shed/conservation cables are adequate for safe plant restart and operation.

NRC staff reviewed the licensee's conformance to Regulatory Guide 1.75 and IEEE 279, respectively, as applied to load-shed cables associated with non-safety loads supplied from safety buses. NRC staff determined that the D. C. Cook licensing basis with respect to designating load-shed cables as non-safety was acceptable, and conforms to Regulatory Guide 1.75 to the extent described in D. C. Cook's response to Regulatory Guide 1.75. In addition, the applicable section of the 1968 edition of IEEE 279 was not applied to the electrical system load-shed circuitry for the balance-of-plant loads powered from the safety buses.

Since the configuration of the load shed cables did not meet current industry guidance, an evaluation of risk implications of the as-built configuration was initiated. As a result of this evaluation, the NRC staff determined that the impact of the current configuration on the reliability of the EDG was not significant based on probabilistic risk insights. Consequently, consideration of a plant-specific backfit was not warranted.

c. Conclusions

Case Specific Checklist Item 7, "Resolution of Non-Safety Related Cables Going to Shunt Trip Coils," is closed. The plant design for the emergency diesel generator load shed/conservation circuitry met industry standards which were existing at the time of plant licensing. Additional studies and balance-of-plant cable testing performed by the licensee provided added assurance that the design of the load shed/conservation circuitry is acceptable. Restart Action Matrix Item R.2.7.5, LER 50-315/98016-02, "Non-Safety Related Cables Routed to Safety Related Equipment," is also closed.

E8.3 Inspectors Review of Restart Action Matrix Items

a. Inspection Scope (92700)

In a letter dated July 30, 1998, the NRC informed the licensee that an oversight panel had been established in accordance with NRC Manual Chapter (MC) 0350, and a checklist was enclosed which specified activities which the NRC considered necessary to be addressed prior to restart. In accordance with MC 0350, an inspection plan was developed to evaluate the effectiveness of the licensee's actions to correct the items listed on the Case Specific Checklist.

In addition to the Case Specific Checklist, the NRC MC 0350 oversight panel developed a Restart Action Matrix (RAM) to track the completion of NRC and licensee activities which were determined necessary for plant restart. The NRC MC 0350 oversight panel assessed the RAM items on the basis of importance, from "risk significant" to "little or no risk significance" and established criteria for inspection of the RAM items based on the relative risk. For low-risk significant items, the panel criteria required that: (1) the licensee had written a condition report to track the issue addressed by the RAM item, and (2) the licensee appropriately tracked the item as required for restart.

b. Observations and Findings

The inspectors reviewed the following low-risk items and concluded that the licensee's actions met the requirements of the MC 0350 oversight panel restart criteria; therefore, the following items are discussed.

- (Closed) RAM Item R.1.31, LER 50-316/99003-00: Fuses not installed for cable passing through containment penetration. On September 26, 1999, during an inspection of a Unit 2 electrical cabinet, maintenance personnel were unable to find fuses for the lighting power transformer cable that passes through containment electrical penetration (CEP) 2-CEP-3P3. Upon further investigation, the licensee determined that the cable had been installed without the required fuse protection during initial plant construction. Without the fuse protection, penetration 2-CEP-3P3 was vulnerable to damage by fault currents if a single breaker failed during certain electrical faults. The licensee wrote CR 99-23923 to document the missing fuses. Licensee Event Report 50-316/99003-00 was written to report that the missing fuses were a condition outside the design basis of the plant.

  The licensee investigated the missing fuses and found that the fuses were supposed to have been installed as part of a plant modification in 1979. The licensee issued limited design change package (LDCP) 2-LDCP-4369 to direct installation of the backup containment penetration fuses in 600V switchgear compartment 2-21D3. The field work for this modification was completed and the LDCP was returned to operations on March 16, 2000. The inspectors concluded that the installation of the fuses restored the plant to its design basis for containment penetration electrical circuit protection.

  The licensee was evaluating the safety significance of the missing fuses and planned to include the results of the evaluation in a supplement to the LER. Licensee Event Report 50-316/99003-00 will remain open pending the inspectors' review of the LER supplement. Restart Action Matrix Item R.1.31 is closed.

- (Closed) RAM Item R.1.37, LER 50-315/20002-00: Large bore piping not meeting the code of record results in safety systems being seriously degraded. In 1989, the licensee established a large bore piping reconstitution project (LBPRP) after a number of as-built discrepancies were identified during piping walk downs. The licensee wrote CR 00-3602 to document these discrepancies.

  To correct the discrepancies, the licensee issued design change package 2-DCP-647. The licensee also added administrative controls to prevent changing Unit 2 operational Modes until the required modifications for each Mode were completed. The inspectors reviewed the licensee's evaluation and proposed corrective actions and concluded that the licensee's actions adequately address the identified deficiencies. Licensee Event Report 50-315/20002-00 and RAM Item R.1.37 are closed.

- <u>(Closed) RAM Item R.2.1.19, LER 50-315/98053-00</u>: Use of inoperable substitute sub-cooling margin monitor. On November 9, 1998, the licensee found that the plant process computer (PPC) core exit thermocouple indications were reading 30 to 35°F too high. The licensee's investigation determined that the PPC inputs were calibrated as linear signals rather than as a thermocouple curve. The licensee also determined that the calibration error was identified after changing the calibration procedure to require a cross-check of the PPC sub-cooling margin readout with the reactor coolant system (RCS) sub-cooling margin monitor. The licensee wrote CR 98-6831 to document the finding, and both units' thermocouple inputs to their respective PPCs were re-calibrated using a corrected calibration procedure to account for the non-linearity.

  Technical Specification 3.3.3.8 required that the post-accident monitoring instrumentation channels shown in Table 3.3-11 shall be operable in Modes 1, 2, and 3. Table 3.3-11 listed the RCS sub-cooling margin monitor as one of the required post accident instruments; however, a note to Table 3.3-11 stated that the PPC sub-cooling margin readout can be used as a substitute for the sub-cooling monitor instrument. The licensee's engineering department determined that the PPC sub-cooling margin readout had been used to substitute for the TS required sub-cooling monitor for greater than 30 days. However, the PPC sub-cooling margin readout had not been calibrated within the surveillance requirement of TS 4.3.3.8, Table 4.3-7. On November 30, 1998, the licensee concluded that the failure to comply with the TS 4.3.3.8 surveillance requirement for the PPC sub-cooling margin readout when it was substituted for the TS required sub-cooling monitor constituted a missed TS surveillance. The licensee submitted LER 50-315/98053-00 to document the missed TS surveillance.

  The inspectors reviewed the licensee's investigation into this event and determined that the PPC indication error was conservative based on the shape of the non-linearity. Therefore, the failure to properly account for the non-linearity of the thermocouples had low safety significance.

  The licensee planned to submit a supplement to LER 50-315/98053-00 to include the results of an evaluation of the significance and root cause of the event. Licensee Event Report 50-316/98053-00 will remain open pending the inspectors' review of the LER supplement. Restart Action Matrix Item R.2.1.19 is closed.

The CRs written to document both of the above RAM items referenced the licensee's root cause investigation performed as part of CR 99-0594. Condition Report 99-0594 was written to document the licensee's historical failure to maintain the design basis of the plant. The NRC has reviewed the licensee's corrective actions for maintenance of the design basis (NRC Inspection Report 50-315/316-2000007) and determined that acceptable quality design control products had been produced and that adequate design controls existed.

- <u>(Closed) RAM Item R.2.1.24; LER 50-315/99014-00,</u> “Requirements of TS 4.0.5 Not Met for Boron Injection Tank Bolting.”

  The licensee determined that the boron injection tank manway bolts were not included in the inservice inspection program as required. These components were removed from the program during the second 10-year update, however, no justification for the removal was found. The missed inspections were performed during the current outages for both units and no concerns were identified with the bolts. The In-Service Inspection Program has been revised to include these missed components. In an effort to prevent recurrence, PMI-5070, “In-service Inspection,” was revised to include a note that revisions to the inservice inspection program should be made in accordance with the requirements of 10 CFR 50.59, such that future changes would be adequately documented.

  Technical Specification 4.0.5a. stated in-service inspections of ASME Code Class 1, 2, and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. In this case, the boron injection tank manway bolts were not included in the inservice inspection program such that the required inspections were not performed. Failure to perform the required inspections is a violation of TS 4.0.5a. However, this Severity Level IV violation is being treated as a non-cited violation (50-315/316-2000013-05), consistent with Section VI.A.1 of the NRC Enforcement Policy. The licensee documented this issue in Condition Report 99-12988. Based on the licensee’s actions, RAM Item R.2.1.24 and LER 50-315/99014 is considered closed.

- <u>(Closed) RAM Item R.2.1.25; LER 50-315/99015,</u> “Radiation Monitoring System Not Tested in Accordance with TS Surveillance Requirements.”

  On April 8, 1999, the licensee identified that the channel test methodology for the Eberline Radiation Monitors did not meet the intent of the surveillance requirements. These requirements were contained in the TS (4.3.3.1, 4.3.2.1.1 and 4.4.6.1a) and in the Offsite Dose Calculation Manual (ODCM). Since 1982, the licensee has been testing the monitor’s spare channel, which bypasses the radiation detector, instead of the active channel. The specific equipment affected included the lower and upper containment, steam generator power operated relief valve, steam jet air ejector vent effluent, and the gland seal leak-off radiation monitors. The monitors were declared inoperable on April 9, 1999. The licensee attributed the event to personnel error.

  Subsequently, the monitors were verified (via channel functional testing) to be functional and the affected procedures were revised. Additionally, the licensee identified that the use of the spare channel does activate the actual process automatic actuation functions, such as process and containment isolation. Therefore, the licensee believes that the associated safety functions would have been met had a release occurred.

The potential for an unmonitored release was minimal, as all effluents were either released through the unit stacks, a monitored release point, or were subject to periodic grab sampling.

This issue was another example of the surveillance program problems that were being tracked and addressed by Manual Chapter 0350 Checklist item 1, "Programmatic Breakdown in Surveillance Testing." The licensee's remediation efforts for this program were reviewed in Inspection Report 50-315/316-99033(DRS). RAM Item R.2.1.25 and LER 50-315/99015 are considered closed.

- (Closed) RAM Item R.2.1.26; LER 50-315/99016, "TS Requirements for Source Range Neutron Flux Monitors Not Met."

  On June 15, 1999, the licensee determined that the Units 1 and 2 source range nuclear instrumentation channels were inoperable. Specifically, the TS (3.3.1.1 and 3.9.2) surveillance tests failed to verify the operability of the source range monitor "High Flux at Shutdown" alarm. This alarm is used to warn operators of a reactivity excursion (specifically an unanticipated boron dilution) during reactor Modes 3, 4, and 5. Licensee staff initiated compensatory measures, including hourly checks of the source range monitors, and revised the surveillance procedures. On June 17, 1999, the monitors were verified operable for each Unit.

  When actuated, the "High Flux at Shutdown" alarm also activates the containment evacuation horns, as they share the same circuitry. During periodic surveillance testing of these horns, licensee staff observed, but did not document, receipt of the flux alarm signal. The licensee believed that the flux alarm was functional. Additionally, the licensee had installed redundant source range monitors with a corresponding high flux alarm on both Units. These alarms were operational and would have indicated a potential reactivity excursion.

  The cause of the event was a past, misinterpretation of an industry standard (IEEE 279, 1968) regarding the definition of "channel." Specifically, the licensee believed that since the alarm was shared by both source range monitors, it did not meet the definition of "channel" and was not required to be verified operable.

  This issue was another example of the surveillance program problems that were being tracked and addressed by Manual Chapter 0350 Checklist item 1, "Programmatic Breakdown in Surveillance Testing." The licensee's remediation efforts for this program were reviewed in Inspection Report 50-315/316-99033(DRS). RAM Item R.2.1.26 and LER 50-315/99016 are closed.

- (Closed) RAM Item R.2.1.27; LER 50-315/99021, "Generic Letter 96-01 Test Requirements Not Met in Surveillance Tests."

  The licensee determined that the surveillance tests for the solid state protection system, engineered safeguards system, and the emergency diesel generator load shed and sequencing circuits did not meet the Generic Letter 96-01 test requirements in all cases. Test procedures were either revised or established to incorporate the missed test requirements for Unit 2. The inspectors reviewed a sample of these procedures and verified that the surveillances were adequate. Although the Unit 1 procedures have not yet been updated, the required changes were being tracked by Condition Reports 99-02611, 99-07428, 99-08195, and 99-14318. Based on the licensee's completed and planned actions, this RAM item and LER 50-315/99021 are considered closed.

- (Closed) RAM Items R.2.3.2 and R.2.3.24; Refueling Water Storage Tank Level Uncertainty and Vortexing Concerns. The licensee identified in LER 50-315/97011-03 the potential for the RWST level instrumentation uncertainties to lead to premature swap over of emergency core cooling systems to recirculation sump operation. This condition was reported to the NRC in LER 50-315/97011-03. This issue was incorporated into the NRC RAM and classified as a high priority inspection Item R.2.3.2. The NRC identified as EEI 50-315/98009-03; 50-316/98009-03, an apparent failure by the licensee to consider the potential for vortexing and air entrainment in establishing the RWST low-low level setpoint. This issue was incorporated into the NRC RAM and classified as a high priority inspection Item R.2.3.24. These issues had been previously discussed in NRC Inspection Reports 50-315/316-99029(DRS); 50-315/316-99029(DRS) and 50-315/316-2000007(DRS); 50-315/316-2000007(DRS) and could not be closed due to the lack of completed documentation including calculations and testing data. The inspectors reviewed the completed calculations and testing which were performed to resolve these issues.

  The inspectors reviewed instrument uncertainty calculation 1-2-UNC-339 Calc 2 "Setpoint Calculation for RWST Level Alarms, RHR Pump Trip Interlock, and Operation Points," Revision 1. Previously, inspectors had identified that this calculation did not address the issues associated with elevation uncertainties for the RWST level instruments. The licensee measured the location of the Unit 2 RWST level instrument with respect to the centerline of the outlet pipe and confirmed a one inch error from that assumed in supporting inputs to this calculation. The licensee documented this error in CR 00-05634 and concluded that this error would affect indicated level by less than 0.25 percent. The licensee documented that an action should be created to check the Unit 1 RWST level instrument elevations. The licensee specified that prior to Mode 4, calculation 1-2-I9 C6 would be revised to reflect the appropriate values and that reviews would be performed to evaluate elevation changes in references for calculations 1-2-UNC339 Calc 1 and Calc 2. Additionally, inspectors confirmed in draft procedure OHP 4023.ES-1.3 "Transfer to Cold Leg Recirculation" Revision 6, that procedure actions were consistent with the RWST level instrument setpoints established in calculation 1-2-UNC339 Calc 2. The inspectors considered these actions adequate to resolve this issue.

The inspectors reviewed NED-2000-520-REP; MPR 2136 "RWST Vortex Testing for D. C. Cook Nuclear Power Plant." In this report, the licensee documented the results of RWST quarter scale flow testing used to determine the flowrates and tank levels associated with vortex formation or onset of air ingestion. The testing data was used to develop equations that bounded the uncertainties associated with the scale model testing as documented in calculation MD-12-RWST-002-N "RWST Vortex Model Test Result Evaluation." The conclusions from this calculation and report NED-2000-520-REP; MPR 2136 supported the licensee's basis for the caution note in draft Revision 6 to OHP 4023.ES-1.3 "Transfer to Cold Leg Recirculation," which established the lowest level operating point in the RWST at an indicated 9 percent level. The test data demonstrated that neither vortexing nor air entrainment would occur in the RWST tank at the levels and flowrates expected during the swapover from injection to cold leg recirculation operation. The inspectors considered these actions adequate to resolve this issue. RAM Items R.2.3.2 and R.2.3.24 and LER 50-315/97011-03 are closed.

- (Closed) RAM Item R.2.3.61; LER 50-315/99017,"Improperly Installed Fuel Oil Return Relief Valve Renders Emergency Diesel Generator Inoperable Due to Personnel Error."

  On June 25, 1999, the licensee declared the Unit 1 "CD" diesel generator inoperable after discovering that the fuel oil return relief valve was installed backwards. The correct orientation of this valve is important to ensure adequate cooling of the generator fuel oil injector pump. The corresponding valves on the other diesel generators (both Units) were installed correctly. The valve had been in the incorrect position since 1986 and was believed to have resulted from personal error.

  The licensee believed the generator was functional, as the TS (4.8.1.1.2) surveillance tests had been successfully completed. Additionally, licensee trend data of the diesel generator's exhaust temperature and combustion pressure, indicated that the pump was being adequately cooled during operation.

  Subsequently, the licensee reinstalled the fuel oil return valve and declared the generator operable. RAM Item R.2.3.61 and LER 50-315/99017 are considered closed.

- (Closed) RAM Item R.2.3.62, LER 50-315/99020-00: Emergency diesel generators declared inoperable due to inadequate protection of air intake, exhaust and room ventilation structures from tornado missile hazards. On July 27, 1999, the licensee identified that the combustion air intake and exhaust piping, and the room ventilation supply duct for the D/Gs did not meet the UFSAR design criteria for protection from tornado-generated missiles. This finding was documented in CR 99-13242. The licensee evaluated the finding and concluded that the D/Gs were in an operable, but degraded, condition. The licensee also determined that the plant was outside its design basis, and LER 50-315/99020-00 was issued.

The licensee determined that a single tornado-generated missile could potentially damage both of the same units' D/G air intake structures, which would result in a common mode failure of both of the D/Gs. The licensee issued design change package 2-DCP-4324 to place a barrier between each D/Gs air intake and exhaust structures to correct the condition. The inspectors reviewed the licensee's corrective actions and concluded that the actions were adequate to support Unit 2 restart. RAM Item 2.3.62 is closed.

10 CFR 50, Appendix B, Criterion III, "Design Control," required, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. The design criterion specified in the Updated Final Safety Analysis Report, Section 1.4.1, required, in part, that, "structures, systems, and components of reactor facilities which are essential to the prevention, or to the mitigation of the consequences, of nuclear accidents . . . shall be designed to withstand, without undue risk to the health and safety of the public, the forces that might be reasonably imposed by the occurrence of an extraordinary natural phenomenon such as earthquake, tornado, flooding condition, high wind, or heavy ice." Contrary to the above, on July 27, 1999, the licensee identified that the emergency diesel generator exhaust and air intake piping and emergency diesel generator room ventilation supply ducts were not protected from tornado-generated missiles. The inspectors concluded that the failure to construct the D/G exhaust and air intake structures in accordance with the design criteria specified in UFSAR Section 1.4.1 constituted a violation of 10 CFR 50. Appendix B, Criterion III. This Severity Level IV violation is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation was entered into the licensee's corrective action program as CR 99-13242. (NCV 50-315/316-2000013-04).

- <u>(Closed) RAM Item R.2.13.2, LER 50-316/20001-00</u>: Through-liner hole discovered in containment liner. On January 17, 2000, the licensee identified a through-liner hole at the location of a weld repair in the Unit 2 containment liner. The licensee wrote CR 00-0850 to document the hole in the containment liner, and the licensee planned to complete the repair and testing of the Unit 2 containment liner prior to restart of the plant.

  The licensee was evaluating the root cause of the through-liner hole and planned to include the results of the evaluation in a supplement to the LER. Licensee Event Report 50-316/20001-00 will remain open pending the inspectors' review of the LER supplement. Restart Action Matrix Item R.2.13.2 is closed.

- (Closed) RAM Item R.2.13.4; LER 50-315/99019, "Victoreen Containment High Range Radiation Monitors Not Environmentally Qualified to Withstand Post-LOCA Conditions."

  On May 21, 1999, the licensee identified that the containment high range radiation monitors may not be environmentally qualified to withstand the effects of a loss of coolant accident. Specifically, the licensee was unable to establish whether these monitors were evaluated under NRC Information Notice 97-45, "Environmental Qualification Deficiency for Cables and Containment Penetration Pigtails." This Notice indicated that these monitors may become inoperable owing to moisture intrusion during a loss of coolant accident or pipe break event. Subsequently, the licensee declared both Units containment monitors inoperable.

  During the associated investigation, the licensee identified that these monitors were environmentally qualified for the specified events, but that the documentation was not well maintained. This investigation was discussed in CR 99-12927. Subsequently, the licensee issued Letter no. SLC-00-940, "Evaluation of the Installed Configuration of the High Range Radiation Monitors," dated April 28, 2000, referencing the appropriate qualification documents. The inspectors reviewed this letter and verified that each key component of the monitors were evaluated in an environmental qualification analysis and that these analyses were consistent with the expected conditions following a loss of coolant accident or pipe break.

  This issue was another example of the program problems that were already being tracked and addressed by Manual Chapter 0350 Checklist item 3, "Programmatic Breakdown in the Maintenance of the Design Basis." RAM Item R.2.13.4 and LER 50-315/99019 are considered closed.

- (Closed) RAM Item R.2.16.2; LER 50-315/99018, "Refueling Water Storage Tank Suction Motor Operated Valve Inoperable Due to Inadequate Design."

  The licensee determined that based on MOV weak link calculations, the thrust required to operate valves ½-IMO-910 and ½-IMO-911 would exceed the yield rating of the valves' yoke assembly. This concern was similar to the MOV design issues that resulted in the breakdown of the MOV program. A significant effort on reestablishing the design-basis of MOVs for Unit 2 was recently completed and reviewed to be acceptable by the NRC in Inspection Report 50-315/316-2000002. Based on this review, the inspectors verified the Unit 2 valves were capable of performing their design-basis requirements without exceeding the valves weak link. Although only the Unit 2 MOVs have been addressed at this time, the licensee will be conducting a similar effort on reestablishing the design-basis capability of MOVs for Unit 1. A subsequent review by the NRC will be conducted in order to close-out NRC's review of the Generic Letter 89-10 program. As such, based on the licensee's completed and planned actions, this RAM item and LER 50-315/99018 are considered closed.

c. Conclusion

The NRC Manual Chapter 0350 panel reviewed licensee actions and inspector assessments related to selected Restart Action Matrix issues. The issues reviewed were acceptable for Unit 2 restart and were closed.

A non-cited violation was identified for the failure to construct the emergency diesel generator exhaust and air intake structures in accordance with the design criteria specified Updated Final Safety Analysis Report Section 1.4.1. The licensee determined that a single tornado-generated missile could potentially damage both of the same units' emergency diesel generator air intake structures, which would result in a common mode failure of both of the emergency diesel generators. The licensee also determined that the plant was outside its design basis, and issued Licensee Event Report (LER) 50-315/99020-00. The licensee issued a design change package to place a barrier between each air intake and exhaust structure to correct the condition. The inspectors reviewed the licensee's corrective actions and concluded that the actions were adequate to support Unit 2 restart. This violation was entered into the licensee's corrective action program as CR 99-13242.

Another non-cited violation was identified for the failure to perform inspections of boron injection tank manway bolts accordance with TS 4.0.5a. The bolts were removed from the licensee's inservice inspection program during the second 10-year update; however, no justification for the removal was found. The missed inspections were performed during the current outages for both units and no concerns were identified with the bolts. The licensee revised the inservice inspection program to include these missed components and revised the appropriate procedure. In an effort to prevent recurrence, PMI-5070, "In-service Inspection," was revised to include a note that revisions to the inservice inspection program should be made in accordance with the requirements of 10 CFR 50.59, such that future changes would be adequately documented. This violation is in the licensee's corrective action program as CR 99-12988.

E8.4 Inspectors Review of NRC Manual Chapter 0350 "Staff Guidelines for Restart Approval"

a. Inspection Scope

In letters to the licensee dated July 30, 1998, and October 13, 1998, the NRC documented the implementation of the NRC Manual Chapter (MC) 0350 "Staff Guidelines for Restart Approval". Included with these letters were checklists which specified activities which the NRC considered necessary to be addressed prior to restart. Enclosure 2 to these letters listed specific criteria which would be evaluated prior to plant restart. In accordance with MC 0350, an inspection plan was developed to evaluate the effectiveness of the licensee's actions to address the selected items. During this inspection period the MC 0350 panel reviewed licensee actions and inspector assessments related to the issues discussed below:

b.1 Staff Guidelines for Restart Approval Item C.1.1, Root Cause Determinations

- (Closed) Staff Guidelines for Restart Approval Item C.1.1.b: Potential root causes of the conditions requiring shutdown and any associated problems were thoroughly evaluated.

  The licensee evaluated the potential root causes associated with the conditions which contributed to the shutdown through the conduct of independent assessments and root cause evaluations per the corrective action program. The licensee documented the specific root causes that contributed to the shutdown in a letter to the NRC dated March 19, 1999, “Donald C. Cook Nuclear Power Plant, Units 1 and 2 Enforcement Actions 98 -150, 98 -151, 98 -152 and 98 -186 Reply to Notice of Violation dated October 13, 1998.” The MC 0350 panel assessed the adequacy of the root cause determinations through the assessment of inspection findings and licensee performance. The MC 0350 panel determined that the licensee did thoroughly evaluate potential root causes as demonstrated in the identification of programmatic weaknesses in key performance areas including design control and TS required surveillance testing. NRC assessment of licensee efforts to address potential root causes of the conditions requiring shutdown and any associated problems was documented in NRC Inspection Reports 50-315/316-99024 and 99029.

- (Closed) Staff Guidelines for Restart Approval Item C.1.1.c: The scope of the analysis considered the applicability of related issues on similar systems, structures, components, procedures, processes, or activities at their own and other industry facilities in an attempt to identify trends or generic industry concerns.

  The scope of the licensees root cause analysis was comprehensive as reflected in the broad scope of the Expanded System Readiness Reviews, Programmatic Assessments, and Functional Area Assessments. NRC inspectors assessed the adequacy of these licensee efforts as documented in NRC Inspection Reports 50-315/316-99012, 99013, 99018, 99024, and 99029. Based on multiple inspection insights which verified the adequate scope of the licensees discovery effort, the MC 0350 panel determined that this criteria was met.

- (Closed) Staff Guidelines for Restart Approval Item C.1.1.e: Rationale for rejecting potential root causes was clearly defined and documented for all root causes evaluated.

  NRC inspectors determined that the licensees root cause evaluation process was clearly defined, documented and effectively implemented, as documented in NRC Inspection Report 50-315/316-99024. The licensees corrective action program included a defined process for the review of root cause determinations, including a process to ensure that the rationale for rejecting potential root causes was clearly defined and documented.

- (Closed) Staff Guidelines for Restart Approval Item C.1.1.f: The licensee's rationale for terminating the root cause and casual factors analyses was based on a documented process that provides a reasonable basis for all conclusions reached.

  NRC inspectors determined that the licensees root cause evaluation process was clearly defined, documented and effectively implemented, and included documented rationale for terminating a root cause analyses. NRC assessment of the licensees corrective action process has been verified and is documented in NRC Inspection Reports 50-315/316-99024 and 99029.

- (Closed) Staff Guidelines for Restart Approval Item C.1.1.g: The population of potential root causes and their respective evaluations have been independently reviewed by the licensee's oversight committee.

  The licensee's oversight committee did conduct independent review of a population of root causes related to the plant shutdown. NRC assessment of the licensee's root cause evaluation review process has been verified and is documented in NRC Inspection Report 50-315/316-99024.

b.2 Staff Guidelines for Restart Approval Item C.1.2, Corrective Action Development

- (Closed) Staff Guidelines for Restart Approval Item C.1.2.a: The proposed corrective actions are clearly cross-referenced to all of the associated root causes and causal factors they are intended to correct, as appropriate.

  The licensee's restart action plan for corrective action clearly correlated corrective actions to the identified root causes and actions to prevent recurrence. Inspector's assessment of licensee corrective action was documented in NRC Inspection Report 50-315/316-99029.

- (Closed) Staff Guidelines for Restart Approval Item C.1.2.b: Each of the corrective actions is assigned an appropriate priority based on safety significance to ensure the proper resources and attention are devoted.

  Licensee condition reports are screened by the Event Screening Committee. The Committee is comprised of members from various plant departments and among its duties is assigning a significance category for each condition report to ensure the proper resources and attention are devoted. The NRC has verified that each of the corrective actions is assigned an appropriate priority based on safety significance to ensure the proper resources and attention are devoted. The verification is documented in NRC Inspection Reports 50-315/316-99024 and 99029.

- <u>(Closed) Staff Guidelines for Restart Approval Item C.1.2.c</u>: Proposed corrective actions identify the desired conditions to be achieved and are adequate to preclude repetition.

  Overall, the licensee's proposed corrective actions for condition reports were adequately developed to identify the desired conditions to be achieved and preclude repetition. Corrective actions for numerous conditions adverse to quality have been discussed and reviewed with NRC personnel during numerous NRC Inspection Manual Chapter 0350 Restart Panel meetings and during various inspections. NRC Inspections have determined that the licensee identified causes were reasonable with respect to the identified weaknesses, and identified corrective actions were appropriate to address the identified causes. This information has been verified as documented in NRC Inspection Reports 50-315/316-99023, 99024, 99026, 99032, and 99029.

- <u>(Closed) Staff Guidelines for Restart Approval Item C.1.2.d</u>: Corrective actions are sufficiently detailed to ensure that all activities related to completion of the corrective action have been identified (i.e., procedure or drawing changes, TS changes, etc.).

  The licensee's corrective actions are sufficiently detailed to ensure that all activities related to completion of the corrective action have been identified. One significant example includes resolution of ice condenser issues, where the scope of corrective actions was comprehensive. Fifty-eight corrective actions were specified in licensee's Restart Action Plan 6, which included revisions to procedures, enhanced training, component modifications, and improvements to the overall material condition in the ice condenser to address deficiencies in surveillance testing, material condition and the design basis for the ice condenser. This information is documented in NRC Inspection Reports 50-315/316-99024, 99026, and 99029.

- <u>(Closed) Staff Guidelines for Restart Approval Item C.1.2.e</u>: Corrective actions include restoring systems and equipment to service and verifying they can perform their intended safety functions through post- maintenance or post-modification testing.

  The NRC has verified that corrective actions include restoring systems and equipment to service and verifying they can perform their intended safety functions through testing. The NRC performed several inspections to verify this item, including inspections of corrective actions, ice condenser design, maintenance and testing, surveillance testing, in-service testing, engineering follow up, and motor-operated valves. This information is documented in NRC Inspection Reports 50-315/316-99022, 99026, 99029, 2000001, 2000002, and 2000007.

- <u>(Closed) Staff Guidelines for Restart Approval Item C.1.2.f</u>: The licensee performed safety evaluations to ensure that corrective actions (e.g., procedure changes or modifications) did not result in a loss of safety margin.

  The NRC assessed the adequacy of licensee corrective actions to address Case Specific Checklist Items No. 4A, "Failure to Perform Safety Evaluations or Screenings," and No. 4B, "Inadequate Safety Evaluations," as documented in Inspection Report 50-315/99023;50-316/99023. Based on the results of this inspection, major improvements were noted regarding the implementation of the 10 CFR 50.59 safety evaluation program at D. C. Cook. Process improvements, close management oversight, and good quality training contributed to the performance improvements. As a result, the NRC concluded that corrective actions to address Case Specific Checklist Restart Item No. 4A and 4B were adequate and these items were closed. In addition, as documented in that inspection report, and Inspection Report 50-316/200003, the licensee staff has demonstrated the ability to perform effective safety evaluations to ensure that modifications or procedure changes do not result in a loss of safety margin. This item is considered closed.

- <u>(Closed) Staff Guidelines for Restart Approval Item C.1.2.g</u>: The licensee adhered to applicable industry codes and standards during the development and analysis of corrective actions.

  The licensee adhered to applicable industry codes and standards during the development and analysis of corrective actions. Examples include corrective actions for the emergency operating procedures program, hydrogen recombiner operability issues, residual heat removal suction valve interlock removal, and resolution of instrument uncertainties, setpoints and/or instrument bias. This information is documented in NRC Inspection Reports 50-315/316/99029, 99032, and 2000010.

- <u>(Closed) Staff Guidelines for Restart Approval Item C.1.2.h</u>: The licensee expanded the scope of the corrective actions to consider all of the causal factors that contributed to the deficiency or problem, including potential generic concerns.

  The licensee documented the specific root causes that contributed to the shutdown and the expanded scope of the corrective actions to consider all of the causal factors in a letter to the NRC dated March 19, 1999, "Donald C. Cook Nuclear Power Plant, Units 1 and 2 Enforcement Actions 98 -150, 98 -151, 98 -152 and 98 -186 Reply to Notice of Violation dated October 13, 1998." The expansion of the corrective actions was also reflected in the seven revisions to the licensee's restart plan. NRC inspectors assessed the adequacy of these licensee efforts as documented in NRC Inspection Reports 50-315/316-99012, 99013, 99018, 99024, and 99029. Based on multiple inspection insights which verified that the licensee expanded the scope of the corrective actions when appropriate, the MC 0350 panel determined that this criteria was met.

- <u>(Closed) Staff Guidelines for Restart Approval Item C.1.2.i</u>: Development of the corrective actions included insights from the organizations or individuals that may have contributed to the event, those responsible for developing the corrective actions, and those responsible for implementing the corrective actions.

  Through the accomplishment of multiple critical self assessments including Expanded System Readiness Reviews, Programmatic Assessments, and Functional Area Assessments, the licensee included insights from the organizations or individuals that may have contributed to the event. NRC inspectors assessed the adequacy of these licensee efforts as documented in NRC Inspection Reports 50-315/316-99012, 99013, 99018, 99024, and 99029. Based on multiple inspection insights which assessed the adequacy of licensee corrective actions including development and implementation of these actions, the MC 0350 panel determined that this criteria was met.

- <u>(Closed) Staff Guidelines for Restart Approval Item C.1.2.j</u>: Interim corrective actions have been developed and documented when permanent corrective action will take an excessive amount of time to implement or cannot be completed before the licensee plans to restart the facility.

  The NRC has conducted extensive inspections of licensee corrective actions to resolve restart related issues. The NRC Inspection Manual Chapter 0350 Restart Panel has reviewed the corrective actions and has confirmed that corrective actions are assigned a required start and completion date, and that problems are being corrected commensurate with the complexity and safety significance of the action.

  The programmatic issues which lead to the licensee's shutdown have been inspected and closed in previous NRC inspection reports, including:

  - Design Control (Inspection Report 50-315/316-99023)
  - Corrective Action Program (Inspection Report 50-315/316-99024)
  - Engineering Corrective Action (Inspection Report 50-315/316-99029)
  - Surveillance Testing and Emergency Operating Procedures (Inspection Report 50-315/316-99033)
  - Engineering Followup (Inspection Report 50-315/316-2000007)
  - Expanded System Readiness Reviews (Inspection Reports 50-315/316-99002, 003, 006, 007, 009, 012, and 018)

  Additional long-term corrective actions are being implemented by the licensee and are being tracked and controlled by the licensee's department leadership plans and the licensee's business plan.

  Specific issues which were identified during the shutdown but which could not be corrected prior to the restart of the facility have been evaluated in accordance with the licensee's corrective action and operability determination processes. As necessary, interim corrective actions were developed and documented. A sample of operability determinations was reviewed by NRC inspectors. This review was documented in the Restart Readiness Assessment Team Inspection

Report (NRC Inspection Report 50-315/316-2000003) and in Section O2.3, of this inspection report. For the sample reviewed, the inspectors determined that appropriate interim corrective actions were developed and documented when permanent corrective actions could not be implemented prior to restart.

- <u>(Closed) Staff Guidelines for Restart Approval Item C.1.2.k</u>: All corrective actions have been incorporated into a comprehensive corrective action plan, which has been approved by the licensee's independent oversight committee.

  The licensee incorporated corrective actions into the Restart Plan which provided the overall logic, strategy, and content, for actions that needed to be completed to support restart of DC Cook Unit 2. The Restart Plan also established a broad basis for continuing improvement in licensee performance. The Restart Plan has been approved by the licensee's independent oversight committee. The MC 0350 panel assessed the licensees implementation of the Restart Plan through frequent management meetings and continuous evaluation of NRC inspection findings. Based on the licensee's overall effectiveness in implementing the Restart Plan, the MC 0350 panel determined that this criteria was met. This information is also documented in NRC Inspection Reports 50-315/316-99024 and 99029.

b.3 <u>Staff Guidelines for Restart Approval Item C.1.3, Corrective Action Plan Implementation and Effectiveness</u>

- <u>(Closed) Staff Guidelines for Restart Approval Item C.1.3.a</u>: Each of the corrective actions is assigned a required start and completion date commensurate with the complexity and safety significance of the action.

  The NRC has conducted extensive inspections of licensee corrective actions to resolve restart related issues. The NRC Inspection Manual Chapter 0350 Restart Panel has reviewed the corrective actions and has confirmed that corrective actions are assigned a required start and completion date, and that problems are being corrected commensurate with the complexity and safety significance of the action in a manner sufficient to support the plant's return to operation. This was evidenced by closure of the Confirmatory Action Letter in February 2000, and further documented in NRC Inspection Reports 50-315/316-99023, 99024, and 99029.

- <u>(Closed) Staff Guidelines for Restart Approval Item C.1.3.b</u>: An organization and individual have been designated with lead responsibility for each of the corrective actions.

  In response to the extended shutdown of the D.C. Cook units, the licensee developed a Restart Plan and associated Restart Action Plans and procedures to address the full spectrum of factors necessary to safely restart the Units. The Restart Action Plans include a listing of source documents, such as condition reports. Condition reports are screened on a daily basis by the Event Screening Committee. The Committee is comprised of members from various plant departments and is responsible for evaluating and assigning the appropriate significance category for each condition report, identifying other actions (such as operability reviews, root cause investigations, etc.) considered appropriate, and assigning the condition reports to the applicable department for resolution. Each action is then assigned to a lead individual within the department with completion dates commensurate with the safety significance of the action. Various NRC inspections have verified the above statements. The NRC staff's assessment that organizations and individuals have been designated with lead responsibility for each of the corrective actions has been verified and is documented in NRC Inspection Reports 50-315/316-99024 and 99029.

- (Closed) Staff Guidelines for Restart Approval Item C.1.3.c: The responsible individual has sufficient authority, resources, and management support to ensure that the action will be adequately completed.

  Individuals designated with lead responsibility for each of the corrective actions has sufficient authority, resources, and management support to ensure that the action will be adequately completed. Restart Action Plans provide the detailed action steps to resolve restart issues and are controlled by personnel at the department manager/superintendent level. Licensee management has made a clear commitment to expend the necessary resources to resolve the deficiencies at the Cook plant needed for restart. This has been made clear through numerous NRC Inspection Manual Chapter 0350 Restart Panel meetings with the licensee. The licensee's long-range resource commitments were discussed with the NRC staff during the April 18, 2000, public meeting. The NRC staff's assessment that responsible individuals have sufficient authority, resources, and management support to ensure that the corrective actions will be adequately completed has been verified and is documented in NRC Inspection Reports 50-315/316-99021, 99024, and 99029; and in the meeting minutes from the April 18, 2000, public meeting.

- (Closed) Staff Guidelines for Restart Approval Item C.1.3.d: The licensee has defined objectives to be achieved from implementing the corrective action plan, including interim objectives to assess the progress of the plan. The objectives are focused on ensuring a lasting improvement in the operation and maintenance of the plant.

  In addition to Restart Action Plans which manage significant restart items, the licensee developed Leadership Plans to capture those actions necessary for long-term improvement of the departments, including functional, programmatic, and organizational actions. The licensee established priorities for the planned corrective actions as those to be resolved prior to restart, those to be performed in the near term, either before or after restart, and those long-term actions to be performed after restart. The NRC staff's assessment that the licensee has defined objectives to be achieved from implementing the corrective action plan, including interim objectives to assess the progress of the plan; and that the objectives are focused on ensuring a lasting improvement in the operation and maintenance of the plant, has been verified and is documented in NRC Inspection Report 50-315/316-99001 and 99004.

- <u>(Closed) Staff Guidelines for Restart Approval Item C.1.3.e</u>: Whenever possible, the licensee's objectives are based on a measurable set of criteria that the licensee can readily track and trend, as appropriate, to provide continuous monitoring of the implementation and effectiveness of the corrective action plan. These measures should form the acceptance criteria for closure and provide precursor indication of declining performance.

  The licensee’s has developed and implemented a set of performance indicators to monitor the effectiveness of corrective actions including the trending of self-identified condition reports and root cause quality. Leadership Plan managers provide schedule progress to the Restart Team for update of the integrated restart schedule. To evaluate the effectiveness of the corrective actions, the licensee has a Corrective Action Review Board to assess the adequacy of root cause evaluations. To assess the effectiveness of Leadership Plan actions, the licensee uses sampling reviews of individual items, department self-assessments, and Performance Assurance department independent reviews. NRC has verified that the Performance Assurance group has conducted effective periodic assessments of the corrective action program. The NRC staff’s assessment that the licensee's objectives are based on a measurable set of criteria that the licensee can readily track and trend has been verified and is documented in NRC Inspection Reports 50-315/316-99024 and 99029.

- <u>(Closed) Staff Guidelines for Restart Approval Item C.1.3.f</u>: The licensee has anticipated and addressed potential conflicts of implementing the corrective action plan with existing facility operational (maintenance, engineering, etc.) practices, regulatory requirements, or personnel activities.

  The licensee has anticipated and addressed potential conflicts of implementing the corrective action plan with existing facility operational practices, regulatory requirements, or personnel activities. This was evidenced by the defueling of both units’ reactor vessels in mid-1999, which enhanced plant safety. In addition, licensee personnel performed the defueling without significant problems. Performance Assurance Department personnel performed effective audits and surveillances to address potential conflicts of implementing the corrective action plan. One item of note is that the licensee uses outside engineering contractors and consultants to augment their engineering staff to perform safety evaluations and screenings. The NRC staff’s assessment that the licensee has anticipated and addressed potential conflicts of implementing the corrective action plan with existing facility operational (maintenance, engineering, etc.) practices, regulatory requirements, or personnel activities has been verified and is documented in NRC Inspection Reports 50-315/316-99023, 99024, and 99029.

- (Closed) Staff Guidelines for Restart Approval Item C.1.3.g: The corrective action plan contains guidance for the licensee to assess changing information or conditions to determine whether the licensee must modify the corrective action plan.

  The licensee's Restart Plan describes processes to supplement the corrective action process. The Expanded System Readiness Review process, using restart criteria, is used to ensure that identified problems are appropriately prioritized for completion prior to restart. The licensee's Restart Plan and Action Plans have been discussed during meetings with the licensee throughout the NRC Inspection Manual Chapter 0350 Restart Panel process. NRC has verified that the licensee modified corrective actions associated with restart items as necessary to address changing conditions. This was evidenced by detailed NRC review of many high and low priority restart action matrix items and review of the licensee's condition reports. The NRC staff's assessment that the corrective action plan contains guidance for the licensee to assess changing information or conditions to determine whether the licensee must modify the corrective action plan has been verified and is documented in NRC Inspection Reports 50-315/316-99024, and 99029.

- (Closed) Staff Guidelines for Restart Approval Item C.1.3.h: The licensee has developed training on both the lessons learned from the event analysis and root cause determination and the technical and administrative changes made to the facilities or practices that includes a discussion regarding why the changes are necessary.

  The licensee developed training for conduct of Expanded System Readiness Reviews which was comprehensive. The subject matter and level of detail provided in the associated lesson plans was appropriate in that it provided the Expanded System Readiness Review teams with the required information needed to understand and perform assigned tasks during the Expanded System Readiness Reviews. NRC also reviewed operator requalification training and determined that the licensee's training and operations departments have appropriately addressed past program weaknesses and continue to address issues affecting training program quality. The Maintenance Proficiency Evaluation training program was also reviewed and found to be thorough and focused on improving the performance of both the maintenance workers and supervisors. The NRC staff's assessment that the licensee has developed training on both the lessons learned from the event analysis and root cause determination and the technical and administrative changes made to the facilities or practices that includes a discussion regarding why the changes are necessary has been verified and is documented in NRC Inspection Reports 50-315/316-99001, 99002, 99016, and 99029.

- <u>(Closed) Staff Guidelines for Restart Approval Item C.1.3.i</u>: The corrective action plan includes requirements to have self-assessments, and as necessary, independent assessments, of the implementation and effectiveness of the plan.

  The licensee has several oversight organizations with responsibilities unique to restart, to supplement the Performance Assurance department responsibilities under 10 CFR 50, Appendix B. The System Readiness Review Board applies direct oversight to the Expanded System Readiness Review program, functional assessments, programmatic assessments, and Restart Action Plans. Performance Assurance, the Plant Nuclear Safety Review Committee, Senior Management Review Team, the Nuclear Safety Design Review Committee, and the Independent Safety Review Group provide indirect oversight of the readiness reviews and Restart Action Plans. All these groups have performed their charter functions and effectively challenged internal assessments and conclusions. The NRC staff's assessment that the corrective action plan includes requirements to have self-assessments, and as necessary, independent assessments, of the implementation and effectiveness of the plan has been verified and is documented in NRC Inspection Reports 50-315/316-99006, 99013, 99024, and 99029.

- <u>(Closed) Staff Guidelines for Restart Approval Item C.1.3.j</u>: In cases where long-term actions remain to be accomplished, the licensee has clearly documented when the action will be complete, the basis for the delay in completing the action, and how the action will be tracked and trended to ensure completion.

  The licensee has established several processes to track corrective actions that have been deferred to post restart including the System Index Database System (SIDS). The inspectors assessed the process utilized by the licensee to prioritize items and determined that the licensee was appropriately utilizing SIDS to track and disposition potential restart related items, as well as effectively evaluating corrective actions for deferral to post restart. The inspectors assessments regarding the licensee's process for delaying corrective actions was documented in Inspection Report 50-315/99022; 50-316/99022. Licensee long term corrective actions to sustain performance improvement were incorporated into Leadership Plans, which have subsequently been integrated into the licensee's Business Plan. The licensee briefed the MC 0350 panel on the Business Plan at a public meeting on April 18, 2000. The Business Plan documented the licensee's plans, goals, and long term corrective actions for sustaining performance improvement. This item is considered closed.

- <u>(Closed) Staff Guidelines for Restart Approval Item C.1.3.k</u>: The licensee has established a predefined time frame following completion of the corrective actions during which they will continue to monitor the effectiveness of the corrective actions.

  The licensee uses performance indicators and other assessment tools to evaluate the effectiveness of its actions. Use of these indicators will continue to be monitored. Inspections by NRC determined that appropriate mechanisms are in place for trending of condition reports and that corrective action department personnel are identifying potential adverse trends. The NRC staff's assessment that the licensee has established a predefined time frame following completion of the corrective actions during which they will continue to monitor the effectiveness of the corrective actions has been verified and is documented in NRC Inspection Report 50-315/316-99024.

b.4 Staff Guidelines for Restart Approval Item C.2.1, Self Assessment Capability

- (Closed) Staff Guidelines for Restart Approval Item C.2.1.c: Effectiveness of licensee's independent review groups. The inspectors attended selected licensee Plant Operations Review Committee meetings, Independent Safety Review Group meetings, Nuclear Safety Design Review Committee meetings, Design Review Board meetings, and System Readiness Review Board meetings in order to assess the effectiveness of the independent review groups. The inspections were performed as part of the routine resident observations of NRC Inspection Reports 50-315/316-99020, 99021, 99022, 200001 and 200013. The inspectors determined that the independent review groups were effective in the performance of their responsibilities. Committee members were observed to routinely ask challenging, technical questions to ensure that issues were being effectively evaluated.

  Assessments of the licensee's Performance Assurance organization were performed and documented in the closure of Restart Approval Item C.2.1.a, Effectiveness of the Quality Assurance Program. (Reference NRC Inspection Reports 50-315/316-99021,022, 024, 025, 026, 029, 032, 033, and 034.)

b.5 Staff Guidelines for Restart Approval Item C.3.1, Assessment of Staff

- (Closed) Staff Guidelines for Restart Approval Item C.3.1.d: Understanding of plant issues and corrective actions. During inspections of the licensee's Expanded System Readiness Reviews (Inspection Reports 50-315/316-99002, 03, 06, 07, 09, 12, 18), and the NRC Engineering Corrective Action Team Inspection (Inspection Report 50-315/316-99029), the inspectors determined that the licensee's staff understood the plant issues and the need for corrective actions. The licensee has implemented leadership plans to ensure long term improvements in the areas of problem identification and corrective actions. Assessments performed as part of the Engineering Corrective Action Team Inspection and the Restart Readiness Assessment Team Inspection (Inspection Report 50-315/316/2000003) support the conclusion that the licensee's staff understands the plant issues and corrective actions.

b.6 Staff Guidelines for Restart Approval Item C.4, Assessment of Physical Readiness of the Plant

- (Closed) Staff Guidelines for Restart Approval Item C.4.f: Significant Hardware Issues Resolved. NRC inspection activities have focused on the correction of the licensee's hardware issues and the processes used to identify and correct these issues. Items such as recirculation sump volumes, containment, motor operated valves and safety-related breakers were assessed by specific NRC inspections and closed. The inspections determined that the licensee was identifying and correcting significant hardware issues. The inspections also determined that when problems were encountered that stop work orders or other prompt corrective actions were taken. There were no significant material aging issues.

Programmatic and functional area assessments performed by the licensee in an effort to identify and determined the extent of condition of issues were the subject of specific NRC inspections. These inspections determined that the licensee's reviews were sufficient to ensure significant hardware, program and process issues were being identified and resolved on a timeliness appropriate to their safety significance. Staff Guidelines for Restart Approval Item C.3.1.a, Demonstrated Commitment to Achieving Improved Performance Through the Results of the Programmatic Readiness Assessment (Staff) was closed in NRC Inspection Report 315/316-99022, issued March 17, 2000.

Overall, NRC inspection results concluded that the licensee had resolved significant hardware issues or that the issues would be completed prior to the appropriate plant operating conditions.

- <u>(Closed) Staff Guidelines for Restart Approval Item C.4.i</u>: Maintenance backlog managed and impact on operation assessed. The licensee developed and implemented a process to assess all items in the backlog to verify that the respective issues can be deferred until after restart without an adverse effect on the operation of the plant. The licensee's backlog of maintenance and design issues was assessed by Senior Risk Analysts and documented in NRC Inspection Report 50-315/316-200004. The assessment determined that the backlog was appropriately managed and would support licensee restart. The impact of the backlog on operation was determined by the licensee and the inspectors to be acceptable.

b.7 <u>Staff Guidelines for Restart Approval Item C.5, Assessment of Compliance with Regulatory Requirements</u>

- <u>(Closed) Staff Guidelines for Restart Approval Item C.5.f</u>: Significant enforcement issues have been resolved.

The licensee documented planned corrective actions to address the Severity Level II Violation in a letter to the NRC dated March 19, 1999, "Donald C. Cook Nuclear Power Plant , Units 1 and 2 Enforcement Actions 98 -150, 98 -151, 98 -152 and 98 -186 Reply to Notice of Violation dated October 13, 1998." The MC 0350 panel assessed the implementation and effectiveness of licensee corrective actions through the evaluation of inspection findings. The performance problems which significantly contributed to the escalated enforcement: surveillance testing, corrective action program, control of the facility design basis, and conduct of safety evaluations, were incorporated into the MC 0350 Case Specific Checklist. As documented in NRC Inspection Reports 50-315/316-99023, 99024, 99026, 99029, 99032, 99033, 200001, and 200007, the inspectors determined that the licensee had taken sufficient corrective actions to address the MC 0350 Staff Guidelines for Restart Approval Items related to the problems addressed in the enforcement action. This item is considered closed.

c. Conclusions

The NRC Manual Chapter 0350 panel reviewed licensee actions and inspector assessments related to the following issues: C.1.1 Root Cause Determination, C.1.2 Corrective Action Development, C.1.3 Corrective Action Plan Implementation and Effectiveness, C.2.1 Self Assessment Capability, C.3.1 Assessment of Staff, C.3.2 Assessment of Corporate Support, C.4 Assessment of Physical Readiness of the Plant, and C.5 Assessment of Compliance with Regulatory Requirements, and determined that licensee actions were adequate to warrant closure of the issues.

E.8.5 (Closed) Violation 50-315/316-95009-03: Three Examples of Inadequate Post-Maintenance Testing (PMT).

This issue had previously been reviewed in NRC Inspection Report 50-315/316-98030, which concluded the majority of the corrective actions had been implemented. The remaining issues to be resolved were the deletion of the requirement to use the PMT matrices in PMI-2294, “Post Maintenance Testing Program,” when identifying PMT and some of the matrices in PMP-2291.PMT.001, “Work Management Post Maintenance Testing Matrices,” could be enhanced. The requirement to use the PMT matrices, when applicable, was included in current version of PMI-2294. The PMT matrices in PMP-2291.PMT.001 were reviewed and revised as necessary. A recent self-assessment in this area; however, indicated the matrices in the MOV area needed further revision to conform with the PMT criteria of the MOV program. These changes were to be incorporated in the next revision. Based on the licensee’s actions, this violation is considered closed.

E8.6 (Closed) IFI 50-315/99002-02(DRS): Unreviewed change in steam generator fouling factor for Unit 1. Subsequently, the licensee replaced the Unit 1 steam generators thereby resolving past fouling issues. This replacement was discussed in Inspection Report 50-315/99035(DRS). Therefore, this item is considered closed.

E8.7 (Closed) IFI 50-315/316-98030-01(DRS): Failure to correct engineering training deficiencies. This item was previously tracked as IFI 50-315/316-97023-01.

Subsequent to this issue, the NRC completed the review of Manual Chapter 0350 Checklist item 2, “Corrective Action Program Breakdown.” This review was documented in Inspection Reports 50-315/316-99024(DRS) and 99029(DRS). The inspectors also reviewed the results of a licensee training accreditation inspection completed in March 2000, and selected meeting minutes from management committees overseeing training performance. These committees were defined in Attachments 1-3 of station procedure PMP 2070.600, Revision 1, “Training Administration and Qualification.” Overall, the licensee appeared to be making reasonable progress towards improving the engineering training program. Therefore, this item is considered closed.

E8.8 (Closed) IFI 50-315/316-98004-06(DRS): NRC identified several, minor deficiencies in licensee calculation no. HXP911210AF. Subsequently, the licensee identified several additional problems with this calculation. These problems were being tracked under Condition Reports 98-3539; 99-939, 99-4955, 99-5739, 99-6643; and 00-4955. The licensee planned to revise this calculation as part of the overall Calculation Reconstitution Project. This project was one of several intended to address Manual Chapter 0350 Checklist item 3, "Programmatic Breakdown in the Maintenance of the Design Basis." The specific issue with calculation HXP911210AF is considered closed, based on the licensee's corrective actions.

E8.9 (Closed) Inspection Follow-up Item (IFI) 50-315/316-99024-01 (DRS) - Screening of Issues in Plant Databases for inclusion in the electronic corrective action program (ECAP). During a September 1999, corrective action program inspection, inspectors noted that integration of other plant databases into the ECAP process was not yet complete or clearly understood, such that all conditions adverse to quality were recognized, prioritized, and monitored commensurate with their safety significance.

Due to a past practice at D.C. Cook of closing CRs prior to completion of the corrective action activity, it had been identified that legacy conditions considered adverse to quality were contained within databases other than ECAP. Although the job order system generated corrective actions to remedy conditions adverse to quality, licensee staff stated that under the previous corrective action program, all conditions adverse to quality were required to be documented in ECAP as the CR process provided additional screening features for operability, reportability, condition investigation and trending of issues. The licensee generated CR 99-22780 to evaluate and document the potential bypass of the corrective action program created by the work control process.

The inspectors reviewed the licensee's corrective actions for the following four alternative databases which could contain conditions adverse to quality: Updated Final Safety Analysis Report Changes, the Job Order System, the Restart Issues database, and the Document Procedure Change System. The inspectors noted that all items had been reviewed for re-start ramifications, and that legacy database items were being reviewed with vigor. The inspectors reviewed 20 CR's which had been generated, or re-opened, as a result of the review and concluded that the corrective actions being implemented were appropriate. This issue is considered closed.

IV. Plant Support

R1 Radiation Protection and Chemistry Controls (71750)

During normal resident inspection activities, routine observations were conducted in the area of radiation protection and chemistry controls using Inspection Procedure 71750. No uncontrolled releases of radioactive material were identified.

S1 Conduct of Security and Safeguards Activities (71750)

During normal resident inspection activities, routine observations were conducted in the area of security and safeguards activities using Inspection Procedure 71750. No discrepancies were noted.

F1 Control of Fire Protection Activities (71750)

During normal resident inspection activities, routine observations were conducted in the area of fire protection activities using Inspection Procedure 71750. No discrepancies were noted.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of the licensee management at the conclusion of the inspection on May 26, 2000. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

X2 Summary of MC 0350 Restart Action Matrix Items

The inspectors reviewed selected items from the NRC Inspection Manual Chapter 0350 Staff Guidelines for Restart Approval (SGRA) Items and the Restart Action Matrix (RAM). The following list indicates NRC and RAM Items which are discussed in the report:

- RAM Item R.1.31, LER 50-316/99003-00, “Fuses not installed for cable passing through containment penetration,” was closed in Section E8.3.

- RAM Item R.1.37, LER 50-315/00002-00, “Large Bore Piping Not Meeting the Code of Record Results in Safety Systems Being Seriously Degraded,” was closed in Section E8.3.

- RAM Item R.2.1.19, LER 50-315/98053-00, “Use of inoperable substitute sub-cooling margin monitor,” was closed in Section E8.3.

- RAM Item R.2.1.24, LER 50-315/99014-00, “Requirements of TS 4.0.5 Not Met for Boron Injection Tank Bolting,” was closed in Section E8.3.

- RAM Item R.2.1.25, LER 50-315/99015, “Radiation Monitoring System Not Tested in Accordance with TS Surveillance Requirements,” was closed in Section E8.3.

- RAM Item R.2.1.26, LER 50-315/99016, “TS Requirements for Source Range Neutron Flux Monitors Not Met,” was closed in Section E8.3.

- RAM Item R.2.1.27, LER 50-315/99021, “Generic Letter 96-01 Test Requirements Not Met in Surveillance Tests,” was closed in Section E8.3.

- RAM Items R.2.3.2 and R.2.3.24, “Refueling Water Storage Tank Level Uncertainty and Vortexing Concerns,” was closed in Section E8.3.

- RAM Item R.2.3.61; LER 50-315/99017,“Improperly Installed Fuel Oil Return Relief Valve Renders Emergency Diesel Generator Inoperable Due to Personnel Error,” was closed in Section E8.3.

- RAM Item R.2.3.62, “Emergency diesel generators declared inoperable due to inadequate protection of air intake, exhaust and room ventilation structures from tornado missile hazards,” was closed in Section E8.3.

- RAM Item R.2.7.5, LER 50-315/98016-02, “Non-Safety Related Cables Routed to Safety Related Equipment,” was closed in Section E8.2.

- RAM Item R.2.13.2, LER 50-316/20001-00,”Through-liner hole discovered in containment liner,” was closed in Section E8.3.

- RAM Item R.2.13.4, LER 50-315/99019, “Victoreen Containment High Range Radiation Monitors Not Environmentally Qualified to Withstand Post-LOCA Conditions, was closed in Section E8.3.

- RAM Item R.2.16.2, LER 50-315/99018, “Refueling Water Storage Tank Suction Motor Operated Valve Inoperable Due to Inadequate Design,” was closed in Section E8.3.

- SGRA Item C.1.1.b, “Potential root causes of the conditions requiring shutdown and any associated problems were thoroughly evaluated,” was closed in Section E8.4.

- SGRA Item C.1.1.c, “The scope of the analysis considered the applicability of related issues on similar systems, structures, components, procedures, processes, or activities at their own and other industry facilities in an attempt to identify trends or generic industry concerns, “ was closed in Section E8.4.

- SGRA Item C.1.1.e, “Rationale for rejecting potential root causes was clearly defined and documented for all root causes evaluated, “ was closed in Section E8.4.

- SGRA Item C.1.1.f, “The licensee’s rationale for terminating the root cause and casual factors analyses was based on a documented process that provides a reasonable basis for all conclusions reached,” was closed in Section E8.4.

- SGRA Item C.1.1.g, "The population of potential root causes and their respective evaluations have been independently reviewed by the licensee's oversight committee," was closed in Section E8.4.

- SGRA Item C.1.2.a, "The proposed corrective actions are clearly cross-referenced to all of the associated root causes and causal factors they are intended to correct, as appropriate," was closed in Section E8.4.

- SGRA Item C.1.2.b, "Each of the corrective actions is assigned an appropriate priority based on safety significance to ensure the proper resources and attention are devoted,“ was closed in Section E8.4.

- SGRA Item C.1.2.c, "Proposed corrective actions identify the desired conditions to be achieved and are adequate to preclude repetition," was closed in Section E8.4.

- SGRA Item C.1.2.d, "Corrective actions are sufficiently detailed to ensure that all activities related to completion of the corrective action have been identified (i.e., procedure or drawing changes, TS changes, etc.),“ was closed in Section E8.4.

- SGRA Item C.1.2.e, "Corrective actions include restoring systems and equipment to service and verifying they can perform their intended safety functions through post- maintenance or post-modification testing," was closed in Section E8.4.

- SGRA Item C.1.2.f, "The licensee performed safety evaluations to ensure that corrective actions (e.g., procedure changes or modifications) did not result in a loss of safety margin," was closed in Section E8.4.

- SGRA Item C.1.2.g, "The licensee adhered to applicable industry codes and standards during the development and analysis of corrective actions," was closed in Section E8.4.

- SGRA Item C.1.2.h, "The licensee expanded the scope of the corrective actions to consider all of the causal factors that contributed to the deficiency or problem, including potential generic concerns," was closed in Section E8.4.

- SGRA Item C.1.2.i, "Development of the corrective actions included insights from the organizations or individuals that may have contributed to the event, those responsible for developing the corrective actions, and those responsible for implementing the corrective actions," was closed in Section E8.4.

- SGRA Item C.1.2.j, "Interim corrective actions have been developed and documented when permanent corrective action will take an excessive amount of time to implement or cannot be completed before the licensee plans to restart the facility," was closed in Section E8.4.

- SGRA Item C.1.2.k, “All corrective actions have been incorporated into a comprehensive corrective action plan, which has been approved by the licensee’s independent oversight committee,” was closed in Section E8.4.

- SGRA Item C.1.3.a, “Each of the corrective actions is assigned a required start and completion date commensurate with the complexity and safety significance of the action,” was closed in Section E8.4.

- SGRA Item C.1.3.b, ”An organization and individual have been designated with lead responsibility for each of the corrective actions,” was closed in Section E8.4.

- SGRA Item C.1.3.c, ”The responsible individual has sufficient authority, resources, and management support to ensure that the action will be adequately completed,” was closed in Section E8.4.

- SGRA Item C.1.3.d, “The licensee has defined objectives to be achieved from implementing the corrective action plan, including interim objectives to assess the progress of the plan. The objectives are focused on ensuring a lasting improvement in the operation and maintenance of the plant,” was closed in Section E8.4.

- SGRA Item C.1.3.e, “Whenever possible, the licensee's objectives are based on a measurable set of criteria that the licensee can readily track and trend, as appropriate, to provide continuous monitoring of the implementation and effectiveness of the corrective action plan. These measures should form the acceptance criteria for closure and provide precursor indication of declining performance,” was closed in Section E8.4.

- SGRA Item C.1.3.f, “The licensee has anticipated and addressed potential conflicts of implementing the corrective action plan with existing facility operational (maintenance, engineering, etc.) practices, regulatory requirements, or personnel activities,” was closed in Section E8.4.

- SGRA Item C.1.3.g, “The corrective action plan contains guidance for the licensee to assess changing information or conditions to determine whether the licensee must modify the corrective action plan,” was closed in Section E8.4.

- SGRA Item C.1.3.h, “The licensee has developed training on both the lessons learned from the event analysis and root cause determination and the technical and administrative changes made to the facilities or practices that includes a discussion regarding why the changes are necessary,” was closed in Section E8.4.

- SGRA Item C.1.3.i, “The corrective action plan includes requirements to have self-assessments, and as necessary, independent assessments, of the implementation and effectiveness of the plan,” was closed in Section E8.4.

- SGRA Item C.1.3.j, “In cases where long-term actions remain to be accomplished, the licensee has clearly documented when the action will be complete, the basis for the delay in completing the action, and how the action will be tracked and trended to ensure completion," was closed in Section E8.4.

- SGRA Item C.1.3.k, "The licensee has established a predefined time frame following completion of the corrective actions during which they will continue to monitor the effectiveness of the corrective actions," was closed in Section E8.4.

- SGRA Item C.1.3.j, In cases where long-term actions remain to be accomplished, the licensee has clearly documented when the action will be complete, the basis for the delay in completing the action, and how the action will be tracked and trended to ensure completion, was closed in Section E8.4.

- SGRA Item C.1.3.k, The licensee has established a predefined time frame following completion of the corrective actions during which they will continue to monitor the effectiveness of the corrective actions was closed in Section E8.4.

- SGRA Item C.2.1.c, Effectiveness of licensee's independent review groups was closed in Section E8.4.

- SGRA Item C.3.1.d, Understanding of plant issues and corrective actions was closed in Section E8.4.

- SGRA Item C.4.f, Significant Hardware Issues Resolved was closed in Section E8.4.

- SGRA Item C.4.i, Maintenance backlog managed and impact on operation assessed was closed in Section E8.4.

- SGRA Item C.5.f, Significant enforcement issues have been resolved was closed in Section E8.4.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

#A. Bakken, Site Vice President
#M. Depuydt, Regulatory Compliance and Licensing
#M. Finissi, Director, Plant Engineering
#R. Gaston, Compliance Manager
#J. Gebbie, Engineering Programs
#S. Greenlee, Director, Design Engineering
#R. Godley, Director, Regulatory Affairs
#J. Kingseed, Assistant Director, Nuclear Fuel, Safety, and Analysis
W. Kropp, Director, Performance Assurance
#W. Lacey, Engineering
#M. Marano, Business Services
#R. Meister, Regulatory Affairs
#J. Molden, Director, Operations
#T. Noon, Director, Restart
#R. Powers, Senior Vice President
#M. Rencheck, Vice President, Nuclear Engineering
#R. Womack, Supervisor, Engineering Programs

# Denotes those present at the May 26, 2000, exit meeting.

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 92700:	Onsite Follow-up of Written Reports of Nonroutine Events at Power Reactor Facilities

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-315/316-2000013-01	NCV	Inadequate Surveillance Test Procedure Restoration Results in Spill in Residual Heat Removal Room
50-315/316-2000013-02	NCV	Failure to Comply With TS 3.0.3 Requirements With Two Essential Service Water Loops Inoperable
50-315/316-2000013-03	NCV	Failure to Submit Licensee Event Report for Condition Prohibited by TS
50-315/316-2000013-04	NCV	Emergency diesel generators declared inoperable due to inadequate protection of air intake, exhaust and room ventilation structures from tornado missile hazards
50-315/316-2000013-05	NCV	Failure to perform TS 4.0.5a, inspection of the boron injection tank.

Closed

50-315/316-2000013-01	NCV	Inadequate Surveillance Test Procedure Restoration Results in Spill in Residual Heat Removal Room
50-315/316-2000013-02	NCV	Failure to Comply With TS 3.0.3 Requirements With Two Essential Service Water Loops Inoperable
50-315/316-2000013-03	NCV	Failure to Submit Licensee Event Report for Condition Prohibited by TS
50-315/316-2000013-04	NCV	Emergency diesel generators declared inoperable due to inadequate protection of air intake, exhaust and room ventilation structures from tornado missile hazards
50-315/316-2000013-05	NCV	Failure to perform TS 4.0.5a, inspection of the boron injection tank.
EEI 50-315/316-98009-03	VIO	Apparent Failure to Consider Potential for Vortexing and Air Entrainment When Establishing the RWST Low-Low Level Setpoint

50-315/316-99021-02	URI	Review of TS requirements for ESW operability during design basis accident
50-315/99001-01	IFI	Residual heat removal system vibration
50-315/316-99024-01	IFI	Screening of Issues in Plant Databases for Inclusion in ECAP.
50-315/97011-03	LER	Operation Outside the Design Basis for ECCS and Containment Spray Pumps for Switch over to Recirculation Sump Suction
50-315/98016-02	LER	Non-Safety Related Cables Routed to Safety Related Equipment
50-315/99014-00	LER	Requirements of TS 4.0.5 Not Met for Boron Injection Tank Bolting
50-315/99015-00	LER	Radiation Monitoring System Not Tested in Accordance with TS Surveillance Requirements
50-315/99016-00	LER	TS Requirements for Source Range Neutron Flux Monitors Not Met
50-315/99017-00	LER	Improperly Installed Fuel Oil Return Relief Valve Renders Emergency Diesel Generator Inoperable Due to Personnel Error
50-315/99018-00	LER	Refueling Water Storage Tank Suction Motor Operated Valve Inoperable Due to Inadequate Design
50-315/99019-00	LER	Victoreen Containment High Range Radiation Monitors Not Environmentally Qualified to Withstand Post-LOCA Conditions
50-315/99020-00	LER	Emergency diesel generators declared inoperable due to inadequate protection of air intake, exhaust and room ventilation structures from tornado missile hazards
50-315/99021-00	LER	Generic Letter 96-01 Test Requirements Not Met in Surveillance Tests
50-315/20002-00	LER	Large bore piping not meeting the code of record results in safety systems being seriously degraded

Discussed

50-315/98053-00	LER	Use of inoperable substitute sub-cooling margin monitor
50-316/99003-00	LER	Fuses not installed for cable passing through containment penetration
50-316/20001-00	LER	Hole discovered in containment liner

LIST OF ACRONYMS

AES	Engineered Safety Features Exhaust Ventilation System
AFW	Auxiliary Feedwater System
AFX	Spent Fuel Pit Exhaust Ventilation System
AHU	Air Handling Unit
ASME	American Society of Mechanical Engineers
ATR	Administrative Technical Requirement
CCW	Component Cooling Water
CEP	Containment Electrical Penetration
CFR	Code of Federal Regulations
CR	Condition Report
CTS	Containment Spray System
CVCS	Chemical and Volume Control System
DCP	Design Change Package
D/G	Diesel Generator
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
ECAP	Electronic Corrective action program
EEI	Escalated Enforcement Issue
EHP	Engineering Head Procedure
ESF	Engineered Safety Features
ESRR	Expanded System Readiness Review
ESW	Essential Service Water
GIP	Generic Implementation Procedure
IFI	Inspection Followup Item
IST	In-Service Test
JO	Job Order
LBPRP	Large Bore Piping Reconstitution Project
LER	Licensee Event Report
LDCP	Limited Design Change Package
LOCA	Loss of Coolant Accident
LOOP	Loss of Off-Site Power
MC	Manual Chapter
MHP	Maintenance Head Procedure
MOV	Motor Operated Valve
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
ODCM	Offsite Dose Calculation Manual
ODE	Operability Determination Evaluations
OHI	Operations Head Instruction
OHP	Operations Head Procedure
OSO	Operations Standing Order
PDR	Public Document Room
PMI	Plant Manager's Instruction
PMP	Plant Manager's Procedure
PMSO	Plant Manager's Standing Order
PMT	Post Maintenance Testing

PPC	Plant Process Computer
RAM	Restart Action Matrix
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RG	Regulatory Guide
RWST	Refueling Water Storage Tank
SIDS	System Index Data-Base System
SGRA	Staff Guidelines for Restart Approval
SQUG	Seismic Qualification Utility Group
SRO	Senior Reactor Operator
SRSS	Square Root Sum of the Squares
STP	Surveillance Test Procedure
TDB	Technical Data Book
TS	Technical Specification
UFSAR	Updated final safety analysis report
VIO	Violation