February 4, 2005

Mr. William O'Connor, Jr.
Vice President
Nuclear Generation
Detroit Edison Company
6400 North Dixie Highway
Newport, MI 48166

SUBJECT: FERMI POWER PLANT, UNIT 2
NRC INTEGRATED INSPECTION REPORT 05000341/2004008

Dear Mr. O'Connor:

On December 31, 2004, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Fermi Power Plant, Unit 2. The enclosed report documents the inspection findings which were discussed on January 4, 2005, with you and other members of your staff.

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

Based on the results of this inspection, seven findings of very low safety significance, five of which involved violations of NRC requirements, were identified. However, because these findings were of very low safety significance and because the issues were entered into your corrective action program, the NRC is treating these violations as Non-Cited Violations in accordance with Section VI.A.1 of the NRC's Enforcement Policy. Additionally, a licensee identified violation is listed in Section 40A7 of this report.

If you contest the subject or severity of a Non-Cited Violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Fermi 2 facility.

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Sincerely,

/RA/

Eric R. Duncan, Chief
Branch 6
Division of Reactor Projects

Docket No. 50-341
License No. NPF-43

Enclosure: Inspection Report 05000341/2004008
w/Attachment: Supplemental Information

cc w/encl: N. Peterson, Manager, Nuclear Licensing
D. Pettinari, Corporate Legal Department
Compliance Supervisor
G. White, Michigan Public Service Commission
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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-341
License No:	DPR-43
Report No:	05000341/2004008
Licensee:	Detroit Edison Company
Facility:	Fermi Power Plant, Unit 2
Location:	6400 N. Dixie Hwy. Newport, MI 48166
Dates:	October 1 through December 31, 2004
Inspectors:	S. Campbell, Senior Resident Inspector T. Steadham, Resident Inspector M. Salter-Williams, Resident Inspector, Davis-Besse T. Bilik, Reactor Inspector R. Jickling, Emergency Preparedness Analyst W. Slawinski, Senior Radiation Specialist
Approved by:	E. Duncan, Chief Branch 6 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000341/2004008; 10/01/2004-12/31/2004; Fermi Power Plant, Unit 2; Fire Protection, Inservice Inspection Activities, Refueling Outage, Surveillance Testing, Event Followup.

This report covers a 3-month period of baseline resident inspection, and announced baseline inspections in the areas of emergency preparedness, radiation protection, and inservice inspection. The inspection was conducted by the resident inspectors and region-based specialist inspectors. Seven Green findings, five of which had associated Non-Cited Violations, were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process (SDP)." Findings for which the SDP does not apply may be "Green" or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. <u>NRC-Identified and Self-Revealed Findings</u>

Cornerstone: Initiating Events

- Green. A finding of very low safety significance was self-revealed when licensee personnel failed to adequately lubricate and prevent repetitive failures of the motor bearings for the north main turbine lube oil pump.

  This finding was more than minor because if left uncorrected, it would become a more significant safety concern. The finding was of very low safety significance because although the finding contributed to the likelihood of a reactor trip, it did not contribute to the likelihood that mitigating equipment or functions would be unavailable. No violation of regulatory requirements occurred. Immediate corrective actions included the installation of a motor lubricated in accordance with the vendor's lubrication instructions. The primary cause of this finding was related to the cross-cutting area of Problem Identification and Resolution. (Section 4OA3.2)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance for the failure to have a conduit junction box cover installed in a cable conduit located in a 3-hour fire barrier separating the cable spreading room and the main control room.

  This finding was more than minor because it was associated with the Protection Against External Factors attribute of the Mitigating Systems cornerstone and adversely impacted the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences since the fire boundary separating two fire zones was not maintained which could result in the loss of mitigating equipment if a fire was to propagate between the cable spreading room and the main control room. The finding was of very low safety significance because the automatic Halon system in the cable spreading room remained operable. A Non-Cited

Violation of Fermi-2 operating license condition 2.C(9) which required that the licensee maintain and implement their approved fire protection plan was identified. As part of their immediate corrective actions, the licensee initiated a fire watch until the covers were replaced. The primary cause of this finding was related to the cross-cutting area of Human Performance. (Section 1R05)

Cornerstone: Barrier Integrity

- Green. The inspectors identified a finding of very low safety significance involving a failure to correctly follow a procedure when a magnetic particle examination was not performed at a required reactor pressure vessel head-to-flange weld location.

  This finding was more than minor because the magnetic particle examination performed on an area other than the prescribed weld could affect the reactor coolant system barrier integrity since, if left uncorrected, it could become a more significant safety concern. Specifically, the failure to perform a required weld inspection on the correct weld location could have allowed undetected through-wall cracks to remain in service. Because this finding was not suitable for a significance determination process evaluation, in accordance with Inspection Manual Chapter 0612, Section 05.04.c, the finding was submitted for review by NRC management; and because there was no evidence of actual flaws, this finding was of very low safety significance. A Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was identified. Immediate corrective actions included stopping the examination and identifying the correct weld location. The primary cause of this finding was related to the cross-cutting area of Human Performance. (Section 1R08)

C Green. A finding of very low safety significance was self revealed when licensee personnel failed to implement the procedural guidance for the proper installation of the refueling shield bridge (cattle chute) which caused a fuel bundle to contact the shield bridge while the bundle was being transported from the reactor core to the spent fuel pool.

This finding was more than minor because it impacted the Barrier Integrity cornerstone and if left uncorrected and a fuel bundle struck the refueling shield bridge again, it could lead to the failure of the fuel bundle cladding and the potential release of fission products, which is a more significant safety concern. Because this finding only affected the fuel barrier, this issue was determined to be of very low safety significance. A Non-Cited Violation of Technical Specification 5.4.1.a was identified. Immediate corrective actions included properly repositioning the refueling shield bridge before transferring another fuel bundle. The primary cause of this finding was related to the cross-cutting area of Human Performance. (Section 1R20.2)

C Green. The inspectors identified a finding of very low safety significance when licensee personnel failed to follow procedures for the movement of the drywell head. During refueling outage 10, contractors moved the drywell head over a portion of the spent fuel pool in violation of the licensee’s procedures.
This finding was more than minor because if left uncorrected, the failure to follow safe load paths on the refuel floor could lead to a more significant safety concern since it would increase the likelihood of a load drop accident. Because this finding was not

suitable for a significance determination process evaluation, in accordance with Inspection Manual Chapter 0612, Section 05.04.c, the finding was submitted for review by NRC management. The finding was determined to be of very low safety significance because the reactor building crane used to move the drywell head was single failure-proof. A Non-Cited Violation of Technical Specification 5.4.1.a was identified. As part of the licensee's immediate corrective actions, this issue was entered into their corrective action program as Condition Assessment Resolution Document (CARD) 04-26765. The primary cause of this finding was related to the cross-cutting area of Human Performance. (Section 1R20.3)

- Green. The inspectors identified a finding of very low safety significance when licensee personnel failed to establish adequate procedures for cleaning the drywell basement trench which could cause inaccurate measurements in unidentified leakage.

  This finding was more than minor because if left uncorrected, it could delay leakage rate information to the operators which was a more significant safety concern. Because this finding was not suitable for a significance determination process evaluation in accordance with Inspection Manual Chapter 0612, Section 05.04.c., this finding was submitted for review by NRC management; and since this finding only affected the monitoring of the reactor coolant system integrity, it was determined to be of very low safety significance. No violation of regulatory requirements occurred. As part of the licensee's immediate corrective actions, the trench drain was thoroughly cleaned. (Section 1R22.2)

- Green. The inspectors identified a finding of very low safety significance when engineering personnel failed to perform a proper evaluation of a scaffold in contact with the torus. Subsequent evaluation of this finding determined that the licensee's procedure for performing the evaluation was inadequate.

  This finding was more than minor because the failure to properly perform the required evaluations to support scaffold variances could become a more significant safety issue if left uncorrected. The finding was of very low safety significance because it represented neither a degradation of the control room barrier nor an actual open pathway in the physical integrity of the reactor containment. A Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," was identified. As part of the licensee's immediate corrective actions, the scaffold in question was removed and all scaffold erection activities in safety-related areas was suspended pending re-evaluation. The primary cause of this finding was related to the cross-cutting area of Problem Identification and Resolution. (Section 4OA3.1)

B. Licensee-Identified Violations

A violation of very low safety significance, which was identified by the licensee, has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and corrective action tracking number is listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 2 operated at or near full power until November 5, 2004, when the unit was shutdown to conduct refueling outage 10 (RF-10). The plant was restarted and reached criticality on December 1, 2004. Mode 1 was entered on December 3, 2004. The plant operated at or near full power until December 4, 2004, when the plant automatically shutdown due to a malfunction on the automatic voltage regulator system for the main turbine generator. Following repairs, the reactor was restarted and reached full power on December 10, 2004. Power remained at or near full power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity (BI)

1R01 Adverse Weather (71111.01)

a. Inspection Scope

The inspectors reviewed licensee procedures and preparations for mitigating the effects of cold weather and high winds. The inspectors reviewed severe weather procedures, emergency plan implementing procedures related to severe weather, annunciator response procedures, and performed walkdowns. Additionally, the inspectors reviewed condition assessment resolution documents (CARDs) and verified that problems associated with adverse weather were entered into the licensee's corrective action program with the appropriate significance characterization.

These activities represented two inspection samples.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments (71111.04)

.1 Partial System Walkdown (71111.04Q)

a. Inspection Scope

The inspectors performed three partial system walkdowns of the following risk significant systems:

- division 2 battery room performed on October 5, 2004;
- general service water (GSW) system performed on October 19, 2004; and
- main steam isolation valves performed on November 29, 2004.

The inspectors selected these systems based on their risk significance relative to the reactor safety cornerstones. The inspectors reviewed operating procedures, system diagrams, Technical Specification (TS) requirements, Administrative TSs, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components were aligned correctly.

In addition, the inspectors verified that equipment alignment problems were entered into the corrective action program with the appropriate significance characterization.

These activities represented three inspection samples.

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown (71111.04S)

a. Inspection Scope

The inspectors performed two complete system walkdowns of the following risk significant systems:

- standby feedwater system performed November 15, 2004, through November 19, 2004; and
- division 2 residual heat removal system performed November 29, 2004, through December 30, 2004.

The inspectors reviewed operating procedures, system diagrams, TS requirements, and applicable sections of the Updated Final Safety Analysis Report (UFSAR) to ensure the correct system lineup. The inspectors verified acceptable material condition of system components, availability of electrical power to system components, and that ancillary equipment or debris did not interfere with system performance.

These activities represented two inspection samples.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Fire Protection Walkdowns (71111.05Q)

a. Inspection Scope

The inspectors performed four fire protection walkdowns of the following risk significant plant areas:

- cable spreading room;
- turbine building - second floor;
- cable tray room; and
- reactor building - first floor.

The inspectors verified that fire zone conditions were consistent with assumptions in the licensee's Fire Hazards Analysis. The inspectors walked down fire detection and suppression equipment, assessed the material condition of fire fighting equipment, and evaluated the control of transient combustible materials. In addition, the inspectors verified that fire protection-related problems were entered into the corrective action program with the appropriate significance characterization.

These activities represented four inspection samples.

b. Findings

Introduction: The inspectors identified a finding of very low safety significance (Green) with an associated Non-Cited Violation (NCV) of Fermi-2 operating license condition 2.C(9) for the failure to have a junction box cover installed in a cable conduit penetrating a 3-hour fire barrier separating the cable spreading room and the main control room.

Description: On October 14, 2004, the inspectors identified two missing conduit junction box covers from a two-inch cable conduit penetrating committed fire boundary E-15004 located in a wall of the cable spreading room. The conduit contained abandoned computer cables associated with the plant computer system. Fire Protection Evaluation, UFSAR Figure 9A-7, "Fire Protection Evaluation Reactor and Auxiliary Buildings Cable Spreading Area Plan," identified that the penetration was a 3-hour fire boundary.

The inspectors questioned the licensee if the missing covers impacted the fire rating for the penetration. After reviewing the configuration, the licensee determined that in order to maintain the required 3-hour rating, the conduit must be sealed within at least 3 feet on each side of the penetration. Since one of the missing covers was within 3 feet of the penetration, the licensee determined that the actual fire rating was less than 3 hours and declared the penetration inoperable.

Based on a review of maintenance activities, the inspectors determined that the conduit was without an adequate seal since at least May of 2000. The failure to have an intact 3-hour fire barrier as approved in the Fermi-2 Safety Evaluation Report through Supplement 6 affected a fire protection defense in depth feature intended to protect structures, systems, and components important to safety to minimize the effect of a fire. The licensee declared the seal inoperable on October 14, 2004, and added the penetration to the hourly fire watch rounds. As part of their immediate corrective actions, the licensee initiated a fire watch until the covers were replaced. The junction box covers were re-installed under Work Request (WR) 000Z041114, restoring the penetration to a fully operable and compliant condition.

Analysis: The inspectors determined that the failure to install the required junction box cover was a performance deficiency warranting a significance evaluation. The inspectors determined this finding to be greater than minor because it was associated with the

Protection Against External Factors attribute of the Mitigating Systems cornerstone and adversely impacted the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences since the fire boundary separating two fire zones was not maintained which could result in the loss of mitigating equipment if a fire was to propagate between the cable spreading room and the main control room.

The inspectors completed a significance determination of this issue using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process (SDP)," Appendix F, "Fire Protection Significance Determination Process." The inspectors assigned this finding to the Fire Confinement category and determined that the issue represented a moderate fire barrier degradation since it could affect more than the ability to reach and maintain cold shutdown conditions. However, although the penetration was degraded, the automatic Halon fire suppression system in the cable spreading room was available. Therefore, this issue screened out as Green.

Enforcement: Fermi-2 Facility Operating License NPF-43, Condition 2.C(9), required that Detroit Edison Company shall implement and maintain in effect all provisions of the approved fire protection program as described in Section 9.5.1 of the UFSAR as amended and approved in the Fermi-2 safety evaluation report through supplement 6. UFSAR 9.5.1.2.3.10 stated that the Fire Hazards Analysis in UFSAR Section 9A provides the barrier requirements for the floors, walls, and ceilings enclosing separate fire areas and for the other penetrations through those barriers. Fire Protection Evaluation, UFSAR Figure 9A-7, "Fire Protection Evaluation Reactor and Auxiliary Buildings Cable Spreading Area Plan," specified a 3 hour rated fire boundary between the cable spreading room and the main control room. Contrary to the above, from May 2000 until October 14, 2004, the licensee failed to fully implement and maintain the provisions of the approved fire protection program as required by Facility Operating License condition 2.C(9). Specifically, the 3-hour fire barrier between the cable spreading room and main control room was not intact as a result of a cable conduit located at committed fire boundary E-15004 without having the required conduit junction box cover installed. However, because the finding was of very low safety significance and because it has been entered into the corrective action program (CARD 04-24751), this violation is being treated as an Non-Cited Violation (NCV 05000341/2004008-01), consistent with Section VI.A of the NRC Enforcement Policy. As part of their immediate corrective actions, the licensee initiated a fire watch until the covers were replaced.

1R06 Flood Protection (71111.06)

a. Inspection Scope

The inspectors performed an inspection related to the licensee's precautions to mitigate the risk from external flooding events. The inspectors performed a walkdown of the following plant areas to assess the adequacy of watertight doors and verify that drains and sumps were clear of debris and were operable:

C residual heat removal complex; and
C Lake Erie shore barriers.

This activity represented one inspection sample.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors reviewed completed test reports and observed the performance of inspections for the division 2 emergency equipment cooling water heat exchanger

The inspectors selected this heat exchanger because its associated systems were risk significant in the licensee's risk assessment and was required to support the operability of other risk significant safety-related equipment. During these inspections, the inspectors observed the as-found condition of the heat exchanger and verified that no deficiencies existed that would mask degraded performance. The inspectors discussed the as-found condition as well as the historical performance of the heat exchanger with engineering department personnel and reviewed applicable documents and procedures.

In addition, the inspectors verified that heat sink problems were entered into the corrective action program with the appropriate significance characterization, and that completed corrective actions were adequate and appropriately implemented.

This activity represented one inspection sample.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

a. Inspection Scope

From November 8-10, 2004, the inspector reviewed the implementation of the licensee's inservice inspection (ISI) program to assess the effectiveness of monitoring degradation of the reactor coolant system (RCS) boundary, risk-significant piping system boundaries, and the containment boundary.

Specifically, the inspector observed licensee vendor personnel perform the following nondestructive examination (NDE) activities to evaluate compliance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements and to verify that indications and defects were dispositioned in accordance with the ASME code:

- ultrasonic examination of a reactor head meridional weld (weld 1-319E, code class 1); and

- magnetic particle examination of a reactor head-to-flange weld (weld 3-319, code class 1).

The inspector reviewed the ultrasonic report for the lower intermediate shell course weld (15-308B) completed on April 22, 2003. During this examination, the licensee identified a relevant indication found to be acceptable per ASME Code, Section XI, IWB-3112(b). The inspector conducted a review of this examination to confirm the licensee had correctly evaluated and dispositioned the indication in accordance with the ASME Code or an NRC approved alternative.

The inspector reviewed the pressure boundary weld records for welds FW-E41-5050-201/10-C1 and FW-E41-5050-20/201-C1. These welds were fabricated during replacement of the high pressure coolant injection (HPCI) system air operated pressure control valve E51-F035 (Class 2 component). The inspector also reviewed the pressure boundary weld records for welds FW-E51-5126-201/202 and FW-E51-5126-202/203. These welds were fabricated during replacement of the reactor core isolation cooling (RCIC) system air operated pressure control valve E51-F015 (Class 2 component). The inspector conducted this review to confirm that the welding process and welding examinations for these welds were performed in accordance with ASME code requirements or an NRC approved alternative.

The inspector performed a review of a sample of ISI-related problems that were identified by the licensee and entered into the corrective action program. The inspector reviewed these corrective action program documents to confirm that the licensee had appropriately described the scope of the problems. Additionally, the review included confirmation that the licensee had an appropriate threshold for identifying issues and had implemented effective corrective actions. The inspector performed these reviews to ensure compliance with 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” requirements. The specific corrective action documents that were reviewed are listed in the attachment to this report. In addition, the inspector verified that the licensee correctly assessed operating experience for applicability to the ISI group.

This activity represented one inspection sample.

b. Findings

Introduction: The inspectors identified a finding of very low safety significance (Green) with an associated Non-Cited Violation (NCV) of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” related to the inadequate magnetic particle examination (MT) of an ASME Code weld.

Description: On November 8, 2004, on the refueling floor, the inspector identified through direct observation that a licensee contract NDE examiner had not performed an MT on the prescribed area. The examiner was required to examine reactor pressure vessel (RPV) head-to-flange weld 3-319, an ASME Code class 1 weld, but instead incorrectly identified a “forged taper” adjacent to the weld as the intended inspection area.

A review of the drawing provided to the examiner during the pre-job brief showed that the area that had been examined was a forged taper about 7 inches above the actual weld centerline. The examiner failed to confirm the actual weld location by referencing the drawing prior to commencing his examination and had instead visually identified an area he believed to be the weld location. The examiner was subsequently shown the correct weld location, which was marked with equally spaced punch marks as well as radiographic datum numbers. The licensee documented this concern in CARD 04-25290.

Analysis: The inspector determined the failure to perform the MT of the prescribed weld was a performance deficiency warranting a significance determination. The inspector reviewed this finding against the guidance contained in IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Dispositioning Screening." In particular, the inspector compared this finding to the findings identified in IMC 0612, Appendix E, "Examples of Minor Issues," to determine whether the finding was minor and concluded that none of the examples listed in Appendix E accurately represented this example. As a result, the inspector compared this performance deficiency to the minor questions contained in IMC 0612, Appendix B, Section 3, "Minor Questions." The inspector concluded that the finding was greater than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," because the finding was associated with the BI cornerstone and the failure of the examiner to follow the Inservice Inspection instructions could, if left uncorrected, become a more significant safety concern. The inspector was concerned that the failure to perform a required weld examination on the correct weld location could have allowed undetected cracks to remain in service. Returning the plant to service with undetected cracks could increase the probability of an RCS break or rupture. The finding also affected the cross-cutting area of Human Performance because the licensee examiner failed to follow the procedure and performed the examination on the incorrect weld.

The inspector determined that the finding could not be evaluated using the SDP in accordance with NRC IMC 0609, "Significance Determination Process," because the SDP for the BI cornerstone applied only to degraded systems/components, not to deficiencies associated with the procedures that are designed to detect component degradation. Therefore, the finding was reviewed by NRC management in accordance with IMC 0612, Section 05.04c, who determined that this finding was of very low safety significance (Green) since there was no evidence of actual flaws once the correct weld was inspected.

Enforcement: 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings appropriate to the circumstances, and shall be accomplished in accordance with these instructions, procedures, or drawings. Surveillance Procedure 43.000.016, "Performance of ISI-NDE Inspections," Step 5.3, required that the prescribed NDE be performed on the components identified in Attachment 1. Attachment 1 directed an MT to be accomplished on weld 3-319. Contrary to the above, on November 8, 2004, a licensee examiner failed to perform an MT on the correct weld area. However, because this finding was of very low safety significance and because the issue was entered into the licensee's corrective action program (CARD 04-25290), this finding is being treated as a Non-Cited Violation

(NCV 05000341/2004008-02), consistent with Section VI.A.1 of the NRC Enforcement Policy. As part of the licensee's immediate corrective actions, upon discovery that the incorrect location had been selected for examination, the correct location was subsequently examined with satisfactory results.

1R12 Maintenance Rule Implementation (71111.12Q)

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the general service water system.

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. Specifically, the inspectors independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- appropriate work practices;
- identifying and addressing common cause failures;
- scoping of systems in accordance with 10 CFR 50.65(b);
- characterizing system reliability issues;
- tracking system unavailability;
- trending key parameters (condition monitoring);
- 10 CFR 50.65(a)(1) or (a)(2) classification and/or re-classification; and
- appropriate performance criteria for systems classified as (a)(2) and/or appropriate and adequate goals and corrective actions for systems classified as (a)(1).

In addition, the inspectors verified that maintenance effectiveness issues were entered into the corrective action program with the appropriate significance characterization.

This inspection activity represented one inspection sample.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation (71111.13Q)

a. Inspection Scope

The inspectors reviewed the licensee's evaluation and management of plant risk for the following maintenance and operational activity affecting safety-related equipment:

- 64B/11EA undervoltage logic functional testing on October 26, 2004.

This activity was selected based on its potential risk significance relative to the reactor safety cornerstones.

As applicable for the above activity, the inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's probabilistic risk analyst and/or shift technical advisor, and verified that plant conditions were consistent with the risk assessment. The inspectors also reviewed TS requirements and walked down portions of redundant safety systems, when applicable, to verify that risk analysis assumptions were valid and applicable requirements were met.

This inspection activity represented one inspection sample.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-Routine Plant Evolutions (71111.14)

Reactor Recirculation Pump “A” Speed Transient

a. Inspection Scope

The inspectors reviewed operator performance in coping with the events and circumstances surrounding the October 31, 2004, speed transient on the “A” reactor recirculation pump. The inspectors reviewed operator logs and plant computer data to determine what occurred, how the operators responded, and if operator response was in accordance with both the relevant procedures and training.

This activity represented one inspection sample.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

.1 Review of Selected Operator Workarounds

a. Inspection Scope

The inspectors evaluated the risk assessment of revised operator workarounds, letter TMSA-04-0022 dated March 19, 2004, to identify any potential effect on the functionality of mitigating systems or on the operators' response to initiating events:

The inspectors selected this issue to review as a potential operator work-around in order to understand how this task was accomplished and the potential effect on plant operations. The inspectors reviewed selected procedures and documents.

This activity represented one inspection sample.

b. Findings

No findings of significance were identified.

.2 Operator Workaround Aggregate Assessment

a. Inspection Scope

The inspectors reviewed the “Active Operations Challenge Index,” dated October 2004, and Nuclear Generation Memorandum NPOP-04-022, “Aggregate Assessment of Operator Work Arounds,” dated March 19, 2004. The inspectors evaluated the cumulative effect of operator work-arounds, control room deficiencies, and degraded conditions on equipment availability, initiating event frequency, and the ability of the operators to implement abnormal or emergency operating procedures. In particular, the cumulative effects of operator work-arounds on the following attributes were considered:

- the reliability, availability and potential for mis-operation of a system;
- the ability of operators to respond to plant transients or accidents in a correct and timely manner; and
- the potential to increase an initiating event frequency or affect multiple mitigating systems.

In addition, the inspectors verified that operator work-around issues were entered into the corrective action program with the appropriate significance characterization.

This activity represented one inspection sample.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

Engineering Design Package (EDP) 13231 for replacing the “D” residual heat removal pump motor was reviewed and selected aspects were discussed with engineering personnel. This document and related documentation were reviewed for adequacy of the safety evaluation, consideration of design parameters, implementation of the modification, post-modification testing, and that relevant procedures, design, and licensing documents were properly updated.

This activity represented one inspection sample.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed post maintenance testing activities associated with the following scheduled maintenance:

- WR 000Z043130, “High wear and vibration on standby liquid control (SLC) pump “B” gear box”;
- WR 000Z043989, “Replacement of the RCIC pump trip coil;” and
- WR 000Z044026; “Repair of the automatic voltage regulator.”

The inspectors reviewed the scope of the work performed and evaluated the adequacy of the specified post maintenance testing. The inspectors verified the post maintenance testing was performed in accordance with approved procedures, the procedures clearly stated acceptance criteria, and that the acceptance criteria were met. The inspectors interviewed operations, maintenance, and engineering department personnel and reviewed the completed post maintenance testing documentation.

In addition, the inspectors verified post maintenance testing problems were entered into the corrective action program with the appropriate significance characterization.

These activities represented three inspection samples.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

.1 Routine Refueling Outage Inspection Activities

a. Inspection Scope

The inspectors observed the licensee’s performance during RF-10 conducted between November 5, 2004, and December 3, 2004.

This inspection consisted of a review of the licensee’s outage schedule, safe shutdown plan and administrative procedures governing the outage, periodic observations of equipment alignment, and plant and control room outage activities. Specifically, the inspectors determined whether the licensee effectively managed elements of shutdown risk pertaining to reactivity control, decay heat removal, inventory control, electrical power control, and containment integrity.

The inspectors performed the following activities daily, during the outage:

- attended control room operator and outage management turnover meetings to verify the current shutdown risk status was well understood and communicated;

- performed walkdowns of the main control room to observe the alignment of systems important to shutdown risk;
- observed the operability of RCS instrumentation and compared channels and trains against one another;
- performed walkdowns of the turbine, auxiliary, and reactor buildings and the drywell to observe ongoing work activities to ensure work activities were performed in accordance with plant procedures and to verify procedural requirements regarding fire protection, foreign material exclusion, and the storage of equipment near safety-related structures, systems, and components were maintained;
- verified the licensee maintained secondary containment in accordance with TS requirements; and
- reviewed selected issues that the licensee entered into its corrective action program to verify identified problems were being entered into the program with the appropriate characterization and significance.

Additionally, the inspectors performed the following specific activities:

- observed the removal of the reactor missile shields;
- observed the control room staff perform the Unit 2 shutdown and initial cooldown;
- observed operators de-inert the drywell and torus;
- observed the moisture/separator removal;
- observed the operators align the residual heat removal system for shutdown cooling;
- toured at-power inaccessible areas;
- routinely toured the drywell including as-found and closeout inspections;
- verified shutdown cooling tagouts;
- verified completion of restart restraint items;
- observed control rod withdrawal to criticality, synchronization to the grid, and portions of the plant power ascension.

In particular, the inspectors reviewed the licensee's restart restraint process and verified the closure of selected issues. Documents reviewed during these inspection activities are listed at the end of this report.

These activities represented one inspection sample.

b. Findings

No findings of significance were identified.

.2 Fuel Movement

a. Inspection Scope

On November 9, 2004, a fuel bundle contacted the refueling shield bridge (cattle chute) while the bundle was being moved from the reactor core to the spent fuel pool. The

inspectors reviewed CARD 04-25319; Procedure 35.710.25, "RPV Internals;" and interviewed maintenance personnel to follow-up on the event.

This activity represented one inspection sample.

b. Findings

Introduction: A self-revealed finding of very low safety significance (Green) with an associated NCV of TS 5.4.1.a, “Procedures,” was identified when licensee personnel failed to implement the procedural guidance for the proper installation of the refueling shield bridge (cattle chute).

Description: On November 9, 2004, refueling operators discovered that fuel bundle JLG818 contacted the cattle chute while moving the bundle from core location 23-54 to the spent fuel pool. The cattle chute is a U-shaped trough lined with 6 inches of lead and is placed across the gap between the RPV flange and the inner edge of the refuel transfer canal. When in place, the cattle chute provides sufficient shielding to ensure continuous access to the drywell during fuel transfers. After the bundle contacted the cattle chute, the refueling operators initially stopped the movement and, following discussions, placed it into the proper spent fuel pool location. Subsequent investigation determined that the cattle chute was about 10 inches out of position. Licensee personnel initiated CARD 04-25319 and repositioned the cattle chute properly. The bundle was subsequently inspected. No damage was identified.

General Electric personnel installed the chute before the first fuel shuffle sequence. During interviews, these individuals stated they believed they had installed it correctly; however, during later discussions these individuals expressed some doubt whether it was properly installed. These individuals did not seek further guidance when they questioned the adequacy of their initial installation.

The inspectors reviewed the procedure and determined that Step 4.21.8 provided the instructions to position the cattle chute between the reactor flange and the fuel pool canal. This step also provided instructions for the cattle chute extension end to lay over the vessel flange and the opposite end of the cattle chute to fit into the pool canal. Step 4.21.2 of the procedure provided reference drawings for installing the cattle chute. Drawings 6C721-4858 and 6C721-2801 provided the reactor building fifth floor heavy load analysis pathway. For the remaining two drawings, DeCo File Drawings R6-196, “Refueling Channel Shield,” provided the dimensional details of the cattle chute and the chute extensions and R6-185, “Portable Radiation Shielding Chute,” provided the installation location of the cattle chute between the reactor and the spent fuel pool. A slot existed in the cattle chute to ensure that the cattle chute was aligned and installed properly. The procedure did not provide instructions to use this slot for proper seating of the cattle chute. A procedure enhancement to include this detail was planned as a corrective action.

Analysis: The inspectors determined that the failure to correctly implement Procedure 37.710.25 and DeCo drawing R6-185 for installing the cattle chute was a performance deficiency warranting a significance evaluation. The inspectors concluded the finding was more than minor in accordance with IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Disposition Screening,” because it was associated with the BI cornerstone; and if the problem were left uncorrected and a fuel bundle struck the cattle chute again, it could lead to the failure of the fuel bundle cladding and the potential release of fission products, which was a more significant safety concern.

The inspectors completed a significance determination of this issue using IMC 0609, “Significance Determination Process (SDP),” Appendix A, Attachment 1, “Cornerstones and Functions Degraded as a Result of a Deficiency,” and the inspectors checked “Fuel Cladding Barrier Degraded.” In the SDP Phase 1 Screening Worksheet for Initiating Event, Mitigating Systems, and BI,” the inspectors determined that since the finding only involved the fuel barrier, this finding was considered to be of very low safety significance (Green). The primary cause of this finding was related to the cross-cuting area of Human Performance since personnel failed to adhere to cattle chute installation procedures.

Enforcement: Technical Specification 5.4.1.a required written procedures be established, implemented and maintained in accordance with the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, dated February 1978. Appendix A, Section 2.I, “Refueling and Core Alterations,” of Regulatory Guide 1.33 required, in part, general operating procedures for refueling and core alterations. Step 4.21 of Procedure 37.710.25 provided instructions for installing the cattle chute before commencing refueling operations. Further DeCo File Drawing R6-185 provided a visual representation on how to properly install the cattle chute. Contrary to the above, on November 9, 2004, personnel who installed the cattle chute did not follow the instructions provided in both Procedure 37.710.25 and DeCo Drawing R6-185 to ensure proper positioning of the cattle chute before commencing the first fuel shuffle. As a result, while the refueling operators were relocating bundle JLG818 from core location 23-54 to the spent fuel pool, the bundle struck the cattle chute because the chute was 10 inches out of position, which was a violation of TS 5.4.1.a.

However, because this violation was of very low safety significance and because the licensee entered this issue into their corrective action program (CARD 04-25319), this violation is being treated as a Non-Cited Violation (NCV 05000341/2004008-03), consistent with Section VI.A of the NRC Enforcement Policy. Corrective actions to address this issue included repositioning the cattle chute properly and revising the procedure to provide improved guidance for installing the cattle chute.

.3 Movement of the Drywell Head Over the Spent Fuel Pool

a. Inspection Scope

The inspectors observed the removal of the drywell head at the beginning of RF-10 to ensure the lift was performed in accordance with established procedures. The inspectors reviewed the procedures used to control the activity including the identified safe load path for the drywell head. The inspectors reviewed other documents and interviewed licensee personnel to determine the circumstances surrounding the lift.

b. Findings

Introduction: The inspectors identified a finding of very low safety significance (Green) with an associated NCV of TS 5.4.1.a, “Procedures,” when licensee personnel deviated from the approved safe load path and moved the drywell head over a portion of the spent fuel pool.

Description: On November 7, 2004, the inspectors observed the removal of the drywell head to support refueling activities at the beginning of RF-10. The controlling procedure for the activity was 35.710.025, “Reactor Vessel Disassembly.” Prior to the lift, the inspectors reviewed this procedure, identified the safe load path for the head, and questioned personnel about the planned load path. The inspectors were informed that although the head would come close to the corner of the pool, the intended load path was in accordance with the procedure.

When personnel moved the head to its storage location, the inspectors witnessed the head traverse over southwest corner of the pool. After the lift, the inspectors discussed the event with the refueling senior reactor operator and other personnel. Initially, the licensee believed they had prior engineering approval to move the head over the pool; however, the inspectors later determined that no such approval existed.

Procedure 35.710.025 explicitly prohibited any deviation from the safe load path for the head in three separate procedure steps. Step 2.5.1 stated, “Observe the safe load handling path....” A caution statement at the beginning of Section 4.5, “Drywell Head Removal,” stated, “When lifting the drywell head, DO NOT deviate from the safe load handling path....” Step 4.5.6 stated, “Lift drywell head out of reactor cavity ensuring not to deviate from safe load handling path.”

Procedure MMA-07, “Hoisting, Rigging, and Load Handling,” Step 3.5.9 required that “load paths shall be adhered to as established by design documents.” Furthermore, Step 5.1.23 required that “a person other than the crane/hoist operator will monitor the load movement to ensure that the load remains in the intended safe load path.”

The safe load path for the head was specified in drawing 6C721-4856 and did not include any portion of the pool. Prior to refueling outage 2, this safe load path was re-evaluated in EDP 12175 to allow for movement of the head from the normal storage location on the refueling floor to an alternate location above the dryer/separator pit. This EDP did not modify the safe load path and, in fact, re-affirmed the requirement to not move the head over any portion of the pool as documented in safety evaluation 91-0033.

The licensee stated the drywell head had been moved over the southwest corner of the pool many times since refueling outage 2 because engineering had approved such a path in memorandum NE-PJ-91-0154, dated May 7, 1991. The inspectors determined this memorandum did not provide such approval and, in fact, stated, “The authorized travel path is documented on drawing 6C721-4856.” No revision of this drawing included any portion of the pool and EDP-12175, which the memorandum referenced,

specifically stated the head would not be moved over the pool. The inspectors concluded the licensee had moved the head on several prior occasions in violation of MMA-07.

Analysis: The inspectors determined that the failure to follow Procedure 35.710.025 for adherence to the safe load path was a performance deficiency warranting a significance evaluation. The inspectors concluded the finding was more than minor in accordance with IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Disposition Screening,” because if left uncorrected, the failure to follow safe load paths on the refuel floor would be a more significant safety concern because it would increase the probability of a load drop accident.

The inspectors determined that the finding could not be evaluated using the SDP in accordance with IMC 0609, “Significance Determination Process,” because the SDP for the BI Cornerstone applied only to degraded systems/components, not to deficiencies associated with the spent fuel pool. Therefore, the finding was reviewed by NRC management in accordance with IMC 0612, Section 05.04c, who determined that this finding was of very low safety significance (Green) since the licensee was using a single failure-proof crane to conduct the lift.

Enforcement: Technical Specification 5.4.1.a required that written procedures be established, implemented and maintained in accordance with the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, dated February 1978. Regulatory Guide 1.33, Appendix A, Section 2.k, “Preparation for Refueling and Refueling Equipment Operation,” required, in part, general operating procedures for reactor vessel disassembly and movement of the drywell head. Step 4.5.6 of Procedure 35.710.025 stated, “Lift drywell head out of reactor cavity ensuring not to deviate from safe load handling path.” The safe load path indicated on drawing 6C721-4856 excluded any portion of the spent fuel pool. Contrary to the above, on November 7, 2004, personnel responsible for moving the drywell head did not follow the instructions provided in Procedure 35.710.025 and drawing 6C721-4856 when they deviated from the approved safe load path and moved the drywell head over a portion of the spent fuel pool which was in violation of TS 5.4.1.a..

However, because this violation was of very low safety significance and because it was entered into the licensee’s corrective action program, this violation is being treated as a Non-Cited Violation (NCV 05000341/2004008-04), consistent with Section VI.A of the NRC Enforcement Policy. The licensee entered this issue into their corrective action program as CARD 04-26765. Since identification of this issue did not occur until after the refueling outage was complete, the licensee had just recently entered this issue into their corrective action program and has begun a review of the events surrounding this issue, as well as their procedures, to determine the necessary corrective actions.

.4 Forced Outage Due to Main Generator Automatic Voltage Regulator Failure

a. Inspection Scope

The inspectors observed the licensee's performance during the December 2004 forced outage due to the automatic voltage regulator failure which resulted in a turbine trip and reactor scram.

This inspection consisted of a review of the licensee's outage schedule, safe shutdown plan and administrative procedures governing the outage, periodic observations of equipment alignment, and plant and control room outage activities. Specifically, the inspectors determined whether the licensee effectively managed elements of shutdown risk pertaining to reactivity control, decay heat removal, inventory control, electrical power control, and containment integrity.

The inspectors frequently performed the following activities during the outage:

- attended control room operator and outage management turnover meetings to verify the current shutdown risk status was well understood and communicated;
- performed walkdowns of the main control room to observe the alignment of systems important to shutdown risk;
- observed the operability of RCS instrumentation and compared channels and trains against one another;
- performed walkdowns of the turbine, auxiliary, and reactor buildings to observe ongoing work activities to ensure that work activities were performed in accordance with plant procedures and to verify that procedural requirements regarding fire protection, foreign material exclusion, and the storage of equipment near safety-related structures, systems, and components were maintained; and
- verified that the licensee maintained secondary containment in accordance with TS requirements.

These activities represented one inspection sample.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

.1 Routine Inspection of Surveillance Tests

a. Inspection Scope

The inspectors reviewed the test results for the following four activities to determine whether risk significant systems and equipment were capable of performing their intended safety function and to verify that testing was conducted in accordance with applicable procedural and TS requirements:

- Procedure 24.202.01; HPCI flow test at 1025 psig;
- Job ID TH01040108; obtain GSW pumps performance data;
- Procedure 37.206.002; RCIC overspeed testing; and
- Hydrostatic test of the RPV and associated piping systems during restart from RF-10.

The inspectors reviewed the test methodology and test results to verify equipment performance was consistent with safety analysis and design basis assumptions. In addition, the inspectors verified surveillance testing problems were being entered into the corrective action program with the appropriate significance characterization.

These activities represented four inspection samples.

b. Findings

No findings of significance were identified.

.2 Changes in Unidentified Leakage Rates - Drywell Trench

a. Inspection Scope

The inspectors reviewed the circumstances surrounding changes in unidentified leakage rate calculations throughout the operating cycle. This included reviewing drawings for the drywell coolers and the drywell temperature thermocouple locations and the sump collection system located underneath the reactor vessel. The inspectors toured the sump during RF-10 to determine the condition of all inputs into the sump, particularly a 6-inch wide by 4 inch deep trench on the drywell floor that surrounded the biological shield. The inspectors interviewed operations and engineering personnel and observed operators conduct shiftily unidentified leakage rate calculations.

This activity represented one inspection sample.

b. Findings

Introduction: The inspectors identified a finding of very low safety significance (Green) for the failure to clean the drywell basement trench drain used to collect leakage outside the biological shield and direct the leakage to the drywell floor drain for RCS unidentified leakage measurement. No violations of regulatory requirements occurred.

Description: On September 22, 2004, drywell unidentified leakage increased above the administrative limit of 0.5 gpm, which was below the TS 3.4.4 limit of 5.0 gpm. The marked increasing trend was noted after completing a forced outage to repair the automatic voltage regulator on the main turbine generator on September 6, 2004. The licensee initiated CARD 04-24335 to document the increase.

On November 13, 2004, during an inspection inside the drywell, an NRC inspector noted that the drywell basement trench drain had accumulated a large amount of foreign material. The engineer documented on CARD 04-25529 that the accumulation was mud-like and noted a stain ring which would indicate that at some time in the past, the drain may have been plugged which could have impacted the accuracy of unidentified drywell leakage calculations. The system engineer who responded to the CARD documented that this appeared to be a long-term buildup of debris. Further, the engineer documented that during the cycle, plant unidentified leakage was erratic and partial obstruction of the trench could have contributed to this behavior. The engineer concluded that the actual impact of this debris on the RCS leakage calculation could not be determined without conducting a hydraulic model of the trench.

Regulatory Guide 1.45, “Reactor Coolant Pressure Boundary Leakage Detection Systems,” provided written guidance on leakage detection systems for identifying reactor coolant pressure boundary leakage. This guide stated the detection and monitoring of leakage of RCS into the drywell area was necessary. Separating leakage from identified and unidentified sources was necessary to provide prompt and quantitative information to the operators to permit them to take immediate corrective action should a leak be detrimental to plant safety. Further, the regulatory guide stated that leakage to the primary containment (drywell) from unidentified sources should be collected and the flow rate monitored with an accuracy of 1 gallon per minute (gpm). The inspectors determined that a clogged trench drain, used to collect leakages outside the biological shield, could represent a hold up volume adversely impacting unidentified leakage calculations. The inspectors determined that water would not be drained from the trench as a result of the mud-like deposits and instead the water would collect and eventually migrate to and through the door in the biological shield and into the floor drain sump. This would adversely impact the measurement of unidentified leakage as specified in Regulatory Guide 1.45.

Before the discovery of the issue, a formal preventative maintenance activity to clean out the trench after completing outages had not been developed. Typically, this activity was done by decontamination personnel. The licensee provided radiation work permit tracking form 2003-1108 that documented the trench was cleaned on April 18, 2003, during refueling outage 9. Due to the large accumulation of mud-like deposits, the inspectors questioned the adequacy of the trench cleaning. WR 000Z043464 was generated to clean and flush the trench, which was completed on November 20, 2004. A mode restraint was created to inspect and clean the drywell trench specifically following refueling outages.

Analysis: The inspectors determined the failure to establish a procedure to clean the drywell basement trench which could impact the ability to measure unidentified leakage was a performance deficiency warranting a significance evaluation. The inspectors concluded the finding was more than minor in accordance with IMC 0612, “Power

Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," since if the problem were left uncorrected, it could delay prompt and quantitative unidentified leakage rate information to the operators to permit them to take immediate corrective action should a leak be a significant impact to plant safety, which was a more significant safety concern.

The inspectors determined that the finding could not be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," because the SDP for the BI cornerstone applied only to degraded systems/components, not to deficiencies associated with the procedures, practices and processes that are designed to detect component degradation. Therefore, the finding was reviewed by NRC management in accordance with IMC 0612, Section 05.04c, who determined that this finding was of very low safety significance (Green) since there was no evidence of a degraded RCS boundary (FIN 05000341/2004008-05).

Enforcement: Since the drywell floor drain sump was a nonsafety-related system and was not relied upon in the licensee's accident analysis, no violation of regulatory requirements occurred.

The licensee entered this issue into their corrective action program as CARD 04-25529. Corrective actions included cleaning and flushing the trench per WR 000Z43464 and creating a preventive maintenance activity to inspect and clean the drywell trench following refueling outages.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the following temporary modification (TM) and verified the installation was consistent with design modification documents and the modification did not adversely impact system operability or availability.

- TM-03-0016; interim annunciator system.

The inspectors verified configuration control of the modifications were correct by reviewing design modification documents and confirmed appropriate post-installation testing was accomplished. The inspectors interviewed engineering and operations department personnel, and reviewed the design modification documents and 10 CFR 50.59 evaluations against the applicable portions of the TS and UFSAR.

This activity represented one inspection sample.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspectors reviewed Revision 29 of the Fermi Power Plant Emergency Plan to determine whether changes identified in Revision 29 reduced the effectiveness of the licensee's emergency planning, pending on-site inspection of the implementation of these changes.

This activity represented one inspection sample.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Plant Walkdowns and Radiation Work Permit Reviews

a. Inspection Scope

The inspectors selectively reviewed the licensee's access controls and survey data for the following work areas located within radiation and high radiation areas in the plant to determine if radiological controls, postings and barricades were adequate:

- Unit 2 drywell;
- Unit 2 turbine building (various areas);
- Unit 2 reactor building (various areas); and
- Unit 2 torus.

The inspectors reviewed the radiation work permits (RWPs) that governed access into the drywell for a variety of work activities, for torus diving incident to desludging, and for inspections/walkdowns in radiation and high radiation areas of the plant. The inspectors reviewed the radiological information to ensure the work control instructions and control barriers that had been specified were adequate and the electronic dosimetry alarm setpoints conformed with area conditions. The inspectors also walked down and surveyed with an NRC survey meter selected areas in the Unit 2 reactor and turbine buildings to verify radiological conditions were consistent with area postings and controls.

This activity represented one inspection sample.

b. Findings

No findings of significance were identified.

2OS2 As Low As Is Reasonably Achievable Planning and Controls (71121.02)

.1 Inspection Planning

a. Inspection Scope

The inspectors reviewed plant collective refueling outage exposure history, current refueling outage exposure trends, and ongoing outage activities in order to assess current dose performance and exposure challenges. This included determining the plant's current 3-year rolling average for collective exposure in order to provide a perspective of significance for any resulting inspection finding assessment.

The inspectors reviewed the Unit 2 RF-10 work and the associated work exposure projections, time/labor estimates, and/or historical dose data for the following seven work activities which were likely to result in the highest personnel collective exposures or were otherwise radiologically significant activities:

- control rod drive exchange;
- torus diving - desludge, inspect/repair torus coating;
- safety relief valves - remove and replace;
- scaffold activities in the steam tunnel and drywell;
- ISI in the steam tunnel and drywell;
- insulation activities in the steam tunnel and drywell; and
- refueling activities.

The inspectors determined site specific trends in collective dose based on plant historical exposure and source term data including historical Boiling Water Reactor Assessment and Control dose rate data. The inspectors reviewed procedures associated with maintaining occupational exposures as low as reasonably achievable (ALARA) and evaluated those processes used for RF-10 to develop dose projections, to determine time/labor estimates, and to track work activity exposures.

These activities represented four inspection samples.

b. Findings

No findings of significance were identified.

.2 Radiological Work Planning

a. Inspection Scope

The inspectors obtained the licensee's list of RF-10 work activities ranked by estimated exposure and reviewed the following radiologically significant work activities:

- refueling activities (RWP 04-1251);
- torus diving to desludge, inspect and repair coating (RWP 04-1160);

- routine mechanical maintenance in the drywell and steam tunnel (RWP 04-1106);
- control rod drive exchange (RWP 04-1113);
- safety relief valve removal and replacement (RWP 04-1115);
- scaffold activities in the steam tunnel and drywell (RWP 04-1105);
- ISI in the steam tunnel and drywell (RWP 04-1110); and
- insulation removal, repair and replacement (RWP 04-1108).

For each of the activities listed above, the inspectors reviewed the RWP, the Radiological Planning Checklists, the Pre-Job ALARA Review, and associated total effective dose equivalent ALARA evaluation, i.e., respirator evaluation, as applicable, along with exposure mitigation criteria. The reviews were performed in order to verify the licensee had established radiological engineering controls that were based on sound radiation protection principles in order to achieve occupational exposures that were ALARA. This also involved determining that the licensee had reasonably grouped the radiological work into activities that were based on historical precedence and/or industry data to allow for enhanced dose projections normalized across the industry.

The inspectors compared the exposure results achieved through the initial 13 days of the scheduled 24-day outage including the dose rate reductions and person-rem expended with the doses projected in the licensee’s ALARA planning for the above listed work activities and for other selected outage activities. Reasons for inconsistencies between intended (projected) and actual work activity doses as well as time/labor estimate differences were reviewed to determine if the activities were planned reasonably well and to ensure the licensee was cognizant of any work planning deficiencies.

The interfaces between radiation protection, maintenance, and scheduling groups were reviewed to varying degrees to identify potential interface problems. The integration of ALARA requirements into work procedures and RWP documents was evaluated to verify that the licensee’s radiological job planning would reduce dose.

The inspectors compared the person-hour estimates provided by maintenance planning and craft groups to the radiation protection ALARA staff with the actual work activity time expenditures in order to evaluate the accuracy of these time estimates.

The inspectors evaluated whether work activity planning included consideration of the benefits of dose rate reduction initiatives such as shielding provided by water filled components/piping, system flushing, hydrolazing and sequencing of scaffold, and shielding installation/removal along with logic-ties in the work scheduling process in order to maximize dose reduction.

Job Progress ALARA Reviews were reviewed by the inspectors for those outage jobs that approached their respective dose estimates or that were generated by the ALARA staff to document problems, to identify changes in work scope or dose estimates and/or otherwise to assess work progress. These reports were reviewed to verify the licensee could identify problems and address them as work continued.

These activities represented seven inspection samples.

b. Findings

No findings of significance were identified.

.3 Verification of Dose Estimates and Exposure Tracking Systems

a. Inspection Scope

The inspectors reviewed the licensee’s assumptions and basis for its collective outage exposure estimate and for individual job estimates, and evaluated the methodology and practices for projecting work activity specific exposures. This included evaluating both dose rate and time/labor estimates for adequacy compared to historical station specific or industry data.

The inspectors reviewed the licensee’s process for adjusting outage exposure estimates when unexpected changes in scope, emergent work or other unanticipated problems were encountered which could significantly impact worker exposures. This included determining that adjustments to estimated exposure (intended dose) were based on sound radiation protection and ALARA principles and not adjusted to account for failures to effectively plan or control the work. The frequency and scope of these adjustments were reviewed to evaluate the adequacy of the original ALARA planning process.

The licensee’s exposure tracking system was examined to determine whether the level of exposure tracking detail, exposure report timeliness, and exposure report distribution was sufficient to support control of outage work exposures. Radiation work permits were reviewed to determine if they covered an excessive number of work activities to ensure they allowed work activity specific exposure trends to be detected and controlled. During the conduct of exposure significant work, the inspectors evaluated if licensee management was aware of the exposure status of the work and would intervene if exposure trends increased significantly beyond exposure estimates.

These activities represented three inspection samples.

b. Findings

No findings of significance were identified.

.4 Job Site Inspections and ALARA Control

a. Inspection Scope

The inspectors observed the following three jobs that were being performed in high radiation areas that potentially represented significant radiological risk to workers:

- intermediate range monitor/source range monitor removal under-vessel;
- torus diving; and
- traversing in-core probe tubing support bracket installation and inspection.

The licensee’s use of ALARA controls for these work activities was evaluated using the following:

- The licensee's use of engineering controls to achieve dose reductions was evaluated as was the job coverage provided by the radiation protection staff to verify procedures and controls were consistent with the licensee's ALARA reviews.

- Job sites were observed to determine whether workers were cognizant of work area radiological conditions and utilized low-dose waiting areas and were effective in maintaining their doses ALARA by moving to the low-dose waiting area when subjected to temporary work delays.

The inspectors reviewed the radiation exposures of individual divers that were involved in torus desludging work to determine whether significant exposure variations existed that may be attributed to poor ALARA practices or to radiation protection staff work oversight or dose monitoring problems. The inspectors also reviewed selected whole body count results and the corresponding internal dose assessment results for several workers who had small intakes while working under-vessel during the outage to evaluate the adequacy of these ongoing assessments.

These activities represented three inspection samples.

b. Findings

No findings of significance were identified.

.5 Source Term Reduction and Control

a. Inspection Scope

The inspectors reviewed licensee records to understand historical trends and current status of plant source terms. The inspectors discussed the plant's source term with radiation protection and chemistry staffs to determine if the licensee has developed an adequate understanding of the input mechanisms and the methodologies and practices necessary to achieve reductions in source term.

The inspectors selectively reviewed exposure reduction initiatives taken for RF-10 such as system/component hydrolazing and flushing. The inspectors reviewed the status of stellite valve internals replacement for those valves identified in the licensee's 1998 Cobalt Reduction Plan so as to minimize the source term introduced into the core each cycle. The inspectors discussed with the licensee its water chemistry control initiatives relative to industry recommended practices and reviewed the effectiveness of its operating chemistry plan.

The inspectors discussed with the licensee its plans for future source term reduction initiatives which were in conceptual stages and the benefits of developing a long-term

strategy to maintain pace with the industry. The inspectors determined whether specific sources and priorities were being considered by the licensee for exposure reduction from source term initiatives.

These activities represented two inspection samples.

b. Findings

No findings of significance were identified.

.6 Radiation Worker Performance

a. Inspection Scope

Radiation worker and radiation protection technician performance was observed during work activities being performed in radiation areas and high radiation areas including various work activities ongoing in the Unit 2 reactor building and turbine building. The inspectors evaluated whether workers demonstrated the ALARA philosophy in practice by being familiar with the work activity scope, the tools to be used for the job, by utilizing low dose waiting areas and had knowledge of the radiological conditions and adhered to the ALARA requirements for the work activity. Job oversight, job support, and the communications provided by the radiation protection staff were also evaluated by the inspectors.

This activity represented one inspection sample.

b. Findings

No findings of significance were identified.

.7 Identification and Resolution of Problems

a. Inspection Scope

The inspectors discussed with the licensee's lead auditor the preliminary results of an ongoing quality assurance department audit of the radiation protection program to assess the licensee's ability to identify and correct problems.

The inspectors verified that identified problems were entered into the corrective action program for resolution and that they had been properly characterized, prioritized, and were being addressed. This included ALARA program critique items and lessons learned from the licensee's previous Unit 2 RF-09.

CARDs generated during the first 13 days of the refueling outage that were related to the radiation protection program were selectively reviewed by the inspectors and licensee staff members were interviewed to verify follow-up activities were being conducted in a timely manner commensurate with their importance to safety and risk using the following criteria:

- initial problem identification, characterization, and tracking;

- disposition of operability/reportability issues;
- evaluation of safety significance/risk and priority for resolution;
- identification of repetitive problems;
- identification of contributing causes; and
- identification and implementation of effective corrective actions.

The licensee's corrective action program was also reviewed to determine if repetitive deficiencies in problem identification and resolution had been addressed, as applicable.

These activities represented two inspection samples.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification (71151)

Cornerstones: Initiating Events, Mitigating Systems, and BI

.1 Reactor Safety Strategic Area

a. Inspection Scope

The inspectors sampled the licensee's submittals for the performance indicators (PIs) listed below. The inspectors used PI definitions and guidance contained in Revision 2 of Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," to verify the accuracy of the PI data. The following two PIs were reviewed:

- RCS leakage; and
- heat removal unavailability.

The inspectors reviewed selected applicable conditions and data from logs, licensee event reports and CARDs from January 1, 2003, through December 31, 2003, for each PI area specified above. The inspectors independently re-performed calculations where applicable. The inspectors compared the re-calculated information to the information required for each PI definition in the guideline to ensure that the licensee reported the data correctly.

These activities represented two inspection samples.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

As discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify they were being entered into the licensee's corrective action system at an appropriate threshold,

adequate attention was being given to timely corrective actions, and adverse trends were identified and addressed.

b. Findings

No findings of significance were identified.

.2 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a screening review of each item entered into the licensee’s corrective action program to identify trends that might indicate the existence of a more significant safety issue. The inspectors considered repetitive or closely related issues that may have been documented by the licensee outside the normal corrective action program, such as in:

- trend reports or PIs;
- major equipment problem lists;
- repetitive and/or rework maintenance lists;
- departmental problem/challenges lists;
- system health reports;
- quality assurance audit/surveillance reports;
- self assessment reports;
- maintenance rule assessments; or
- corrective action backlog lists.

The inspectors verified the licensee was identifying issues at an appropriate threshold and entering them into their corrective action program by comparing those issues identified by the NRC during the conduct of the plant status and inspectible area portions of the program with those issues identified by the licensee.

This activity represented one inspection sample.

b. Issues

The inspectors identified a potential degrading trend with the licensee's ability to resolve equipment-related problems. The inspectors identified five notable equipment issues that occurred in 2004 in which the licensee expended significant effort in resolving several times. These examples include:

- The automatic voltage regulator failed on September 4, 2004, and on December 4, 2004. Each occurrence resulted in a turbine trip and reactor scram.
- The diesel fire pump experienced high coolant temperatures which required the pump to be shut down on July 13, 2004, and August 26, 2004.
- The HPCI outboard steam isolation valve failed to stroke closed on April 8, 2004, and August 12, 2004.
- The north main turbine lube oil pump motor bearings failed on February 8, 2004; April 24, 2004; and October 5, 2004.
- The west station air compressor exhibited high lube oil temperatures which required the compressor to be shut down on March 18, 2004; August 20, 2004; and December 22, 2004.

Due to the diverse nature of the above equipment (digital equipment, pumps, valve, air compressor) the inspectors were concerned that a more generic issue with problem identification and resolution could exist.

4OA3 Event Followup (71153)

.1 (Closed) URI 05000341/20004007-01: Review of Torus/Scaffold Interaction Operability Evaluation

a. Inspection Scope

The inspectors opened this unresolved item to document an NRC-identified issue with a scaffold in contact with the torus. The inspectors reviewed the licensee's operability evaluation to assess the scaffold's impact on the operability of the torus. The inspectors reviewed design basis documents to ensure the evaluation was consistent with the current licensing basis.

b. Findings

Introduction: The inspectors identified a finding of very low safety significance (Green) with an associated NCV of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," when licensee personnel failed to establish appropriate procedures to properly evaluate scaffolding variances.

Description: As described in Section 1R15.1 of inspection report 05000341/2004007, on September 14, 2004, the inspectors identified a scaffold that was not in compliance with the licensee's documented rattlespace requirements. On September 17, 2004, after prompting by the inspectors, the licensee removed the scaffold and discovered a 2 foot long horizontal member erected between and in direct contact with both the torus and the torus room wall. Since this member could have placed a significant stress on the torus had the torus moved, during a design basis event, such as a loss-of-coolant-accident, the licensee completed a past operability evaluation.

The scaffold was originally approved by engineering personnel in August 2004. Further, the evaluations that followed the inspectors' questions on September 14, 2004, also determined the scaffold to be acceptable. In both cases, the licensee considered the effect of the scaffold impacting the torus during a seismic event to be negligible and, thus, acceptable. Neither evaluation considered the effect of the horizontal member in question becoming wedged between the torus and the torus room wall in the event of a design basis accident or other plant transient. The licensee later concluded that the failure to consider the effects of the torus being restrained by the scaffold was a programmatic failure in their philosophy of evaluating scaffolds for rattlespace variances. The inspectors determined the licensee missed an opportunity to correct this programmatic inadequacy when the inspectors raised a similar concern on June 29, 2004, as documented in CARD 04-22915.

The inspectors forwarded the evaluation to Region III structural experts who concurred with the licensee's conclusion that the torus would have remained operable during a design basis accident or other plant transient. The licensee concluded that although torus shell deformation could have occurred, there remained a large margin between the maximum calculated local stresses for the torus shell and the allowable stresses specified by ASME code limits for deformation of the shell prior to a rupture.

The licensee entered this issue into their corrective action program as CARD 04-24282. The licensee concluded their existing scaffold procedure MMA08 did not provide adequate guidance on evaluating rattlespace variances because the effects of equipment movement during a transient were not considered.

Analysis: The inspectors determined that the failure to perform a proper evaluation of this scaffold was a performance deficiency warranting a significance determination. The inspectors concluded the finding was more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," because the failure to properly perform the required evaluations to support scaffold variances could become a more significant safety concern if left uncorrected.

The inspectors completed a significance determination of this issue using IMC 0609, "Significance Determination Process (SDP)," Appendix A, Attachment 1, "SDP Phase 1 Screening Worksheet for IE (Initiating Events), MS (Mitigating Systems), and BI Cornerstones." The inspectors concluded that this finding affected the BI cornerstone because the torus (e.g. containment barrier) was degraded; however, since the finding represented neither a degradation of the control room barrier nor an actual open pathway in the physical integrity of the reactor containment, this finding was considered to be of very low safety significance (Green).

The primary cause of this finding was related to the cross-cutting area of Problem Identification and Resolution since licensee personnel had multiple prior opportunities to identify and correct this problem.

Enforcement: 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," required, in part, that activities affecting quality be prescribed by procedures of a type appropriate to the circumstances. Contrary to the above, the licensee failed to have appropriate procedures to perform adequate rattlespace variance evaluations for scaffolds erected in proximity to safety-related equipment. As a result, in August 2004, a scaffold was erected which jeopardized the operability of the torus during a design basis event. However, because this violation was of very low safety significance and because it was entered into the licensee's corrective action program, this violation is being treated as a Non-Cited Violation (NCV 05000341/2004008-06), consistent with Section VI.A of the NRC Enforcement Policy.

Once identified, the licensee entered this issue into their corrective action program as CARD 04-24282 and performed an operability evaluation which verified the operability of the torus. In addition, the licensee immediately suspended all scaffold erection in safety-related areas pending a comprehensive review of all scaffolds with approved variances utilizing the lessons learned from this issue. The licensee then revised scaffold procedure MMA08 to require a more thorough evaluation of all scaffold variances.

.2 Failure of North Main Turbine Lube Oil Pump

a. Inspection Scope

The inspectors reviewed the CARD and work requests and interviewed engineering personnel to determine the causes of the repetitive failure of the north main turbine lube oil pump motor.

b. Findings

Introduction: A self-revealed finding of very low safety significance (Green) was identified for the failure to maintain adequate lubrication of the north main turbine lube oil pump upper motor bearing that caused repetitive failures of the pump. No violation of regulatory requirements occurred.

Description: The turbine main lube oil system has three pumps, a north and a south pump supplied by alternating current (AC) power and an emergency oil pump supplied by direct current (DC) power. One AC pump is normally in operation with the alternate pump in automatic control in the event the operating pump malfunctions. The DC emergency oil pump supplies oil to the turbine bearings if the north and the south pumps lose power. This pump is used to prevent damage to the turbine bearings while the turbine is coasting to a stop and not for normal operation. Pump operation is alternated between the north and the south pump on a quarterly basis to ensure a run time of about 9 months. The licensee conducted preventive maintenance on the non-operating pump that included lubricating the pump bearing.

On March 7, 2003, a loud grinding noise was heard due to the failure of the north main turbine lube oil pump motor, serial number 5030/4, upper bearing, which was documented on CARD 03-11250. The cause of this failure was that the pump had run for 10 months which was greater than the allowed run time of 9 months recommended by the vendor. The 9-month run time was exceeded due to unexpected forced outages that occurred during the operating cycle which interrupted the planned quarterly swap and consequently the ability to conduct preventive maintenance. On April 21, 2003, mechanics installed a rebuilt spare motor, serial number V444 370 01, in accordance with preventive maintenance task N147020100.

On February 8, 2004, the north main turbine lube oil pump tripped due to inadequate lubrication of the upper bearing as documented on CARD 04-20460. Because some of the run time was on the south main turbine lube oil pump, the total run time for the north pump was about 6 months. The licensee determined that the cause of this failure was due to not performing Step 4 of PM N918030100 which was the instruction to inject 40 strokes of grease while rotating the shaft. This step was not performed because the mechanics believed the refurbished motor had been previously lubricated by the vendor.

On February 8, 2004, mechanics installed the lube oil pump, Serial Number 5030/4, that had been rebuilt following the March 7, 2003 failure per WR 000Z040371.

On April 24, 2004, the north main turbine lube oil pump motor experienced another similar failure. This failure was documented on CARD 04-21802 and closed to CARD 04-22164 written to establish the performance improvement plan since the system entered 10 CFR 50.65 (a)(1) maintenance rule monitoring from the repetitive failures. The pump had operated only 11 weeks. Upon disassembly, an inspection of the lubricated areas of the motor determined the motor bearings were well lubricated. The licensee determined the cause of the failure was the bearing clearances were too wide between the inner and outer races. Too wide a clearance caused the bearings to heat up only the grease at pump speed causing a rapid degradation of the grease. The inspectors reviewed WR 000Z040371 and found no instruction to verify axial thrust that established the bearing clearances. Mechanics installed a new motor, serial number V444 370 01, per WR 000Z41197 on April 25, 2004. This was the spare motor rebuilt by the vendor following the March 7, 2003, failure.

On October 5, 2004, the licensee noticed an increase in bearing vibration data and shutdown and switched pump operation to the south main turbine lube oil pump. CARD 04-24607 was written and the investigation from this CARD determined inadequate lubrication of the upper bearing caused the elevated vibration. For this failure, the mechanics again believed the vendor had lubricated the rebuilt spare and did not lubricate the underside of the bearing in the motor.

From these events and with assistance from the pump vendor, the licensee discovered the need to maintain lubrication in the underside of the bearing by injecting 40 strokes of grease to prevent bearing damage. Typically, mechanics used this method when implementing Step 4 of PM N918030100, which was completed during refueling outages. This requirement was discussed in vendor manual publication C51F11, "Installation & Maintenance Cage and Wound Rotor Impack Range Frames 355-450." The manual was recently discovered while researching the causes of the October 5, 2004, failure. Previously, the licensee used vendor manual VMTI-1.6.19.2, "A.C. Main Lubricating Oil Pump (Motor English Electric AEI)," that did not discuss the lubrication of the underside of the bearings.

Failures occurred on the spare pump, serial number V444 370 01, because it had been in storage and Step 4 of PM N918030100 had not been performed. Also, mechanics erroneously assumed that adding lubrication to the underside of the bearing was completed by the vendor during rebuild. Lessons learned from the February 8, 2004, failure to prevent the October 5, 2004, failure were not incorporated because the licensee did not include in CARD 04-20460 a corrective action to train maintenance personnel on the need to lubricate the underside of the bearing of the spare pump.

Analysis: The inspectors determined that not maintaining adequate lubrication of the motor bearings for the north main lube oil pump that resulted in multiple failures of the motor within a year was a performance deficiency warranting a significance evaluation. The inspectors concluded the finding was more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," because if the problem were left uncorrected and inadequate lubrication practices occurred simultaneously on the south main turbine lube oil pump, the finding would lead to a failure of the lube oil system and plant scram, which was a more significant safety concern.

The inspectors completed a significance determination of this issue using IMC 0609, "Significance Determination Process (SDP)," Appendix A, Attachment 1, "SDP Phase 1 Screening Worksheet for Initiating Events, Mitigating Systems, and BI Cornerstones." The inspectors concluded that this finding affected the Initiating Event cornerstone. Although the finding contributed to the likelihood of a reactor trip, the finding did not contribute to the likelihood that mitigating equipment or functions would be unavailable. Therefore, this finding was considered to be of very low safety significance (Green) (FIN 05000341/2004008-07). The primary cause of this finding was related to the cross-cutting area of Problem Identification and Resolution since licensee personnel had multiple opportunities to identify and correct this problem.

Enforcement: Since the main turbine lube oil system was a nonsafety-related system, no violation of regulatory requirements occurred.

The licensee entered this issue into their corrective action program as CARD 04-22164 when the system entered 10 CFR 50.65 (a)(1) maintenance rule monitoring. Immediate corrective actions included the installation of a motor lubricated in accordance with the vendor's lubrication instructions. Long-term corrective actions included revising procedures to ensure the motor bearings were properly lubricated in accordance with recent vendor manual instructions and the training of maintenance personnel to ensure that the underside of the bearings were adequately lubricated for spare motors.

.3 (Closed) URI 05000341/2004002-04: Failure of Nitrogen Inerting Primary Containment Isolation Valves to Meet IST Stroke Times.

This URI involved reviewing the root cause analysis for the failures of primary containment pneumatic divisions 1 and 2 supply outboard and inboard primary containment valves T4901F465 and 468, respectively, to meet ISI stroke time acceptance criteria. The licensee disassembled the valves and found debris in the internals of the valve operators. Previous failures dating back to 1995 indicated that these valves were mechanically bound/stuck due to an inappropriate pipe sealant compound being used on the non-interruptible instrument air system to stop fitting leaks. During RF-10, a relief valve, pressure regulator, and tubing normally connected to the valves were disassembled to determine whether these components were the source of the debris. Debris removed from these items was bagged and sent to a laboratory for analysis. Completion of these tests were scheduled between February and March 2005. Consequently, final root cause has not been determined at this time. No recent failures of the valves have occurred since the valves were replaced in October 2003. This item is closed.

.4 Review of Licensee Events and Degraded Conditions

a. Inspection Scope

The inspectors monitored plant status on a daily basis and responded to licensee events and degraded conditions as appropriate. The inspectors monitored licensee performance to evaluate whether the licensee appropriately resolved the event or issue. Specifically, the inspectors independently verified the licensee's actions to address the following:

- failure of station service transformer 68 during the week of October 31, 2004;
- fuel channel bowing indications identified during RF-10;
- reactor scram on an automatic voltage regulator trip on December 4, 2004;
- Various snubber failures discovered from testing during RF-10;
- Annunciator system failure and resulting notification of unusual event on December 26, 2004;
- continued follow-up on URI 05000341/2004007-06, HPCI outboard steam isolation valve failure to close; and
- continued follow-up on URI 05000341/2004007-05, emergency diesel generator 12 blower failure.

These activities represented seven inspection samples.

b. Findings

No findings of significance were identified.

4OA4 Cross-Cutting Aspects of Findings

.1 A finding described in Section 1R05 of this report had, as a primary cause, a human performance deficiency, in that maintenance personnel failed to follow a procedure by not installing the cover plate on an electrical conduit junction box.

.2 A finding described in Section 1R08 of this report had as a primary cause a human performance deficiency in that the licensee examiner failed to follow a procedure by not performing an MT on the prescribed weld location.

.3 Two findings described in Section 1R20 of this report had as a primary cause a human performance deficiency, in that vendors, who conducted refueling activities, failed to follow procedures by not installing a refueling shield bridge correctly before beginning fuel transfers and deviating from a safe load path by lifting the drywell head over the spent fuel pool.

.4 Two findings described in Section 4OA3 of this report had as a primary cause a problem identification and resolution aspect to the deficiency, in that the licensee failed to identify the impact of scaffolding contacting the torus and the licensee failed to correct multiple inadequate lubrication practices on the north main turbine lube oil pump motors.

4OA6 Meetings

.1 Exit Meetings

The inspectors presented the inspection results to Mr. W. O'Connor and other members of licensee management at the conclusion of the inspection on January 4, 2005. The inspectors asked the licensee whether any material examined during the inspection should be considered proprietary. All proprietary information was returned to the licensee.

.2 Interim Exit Meetings

Interim exit meetings were conducted for:

- ISI (IP 71111.08) with Mr. D. Cobb on November 10, 2004;
- Occupational Radiation Safety ALARA program inspection during the licensee's Unit 2 RF-10 refueling outage with Messrs. W. O'Connor and D. Cobb on November 19, 2004; and
- Emergency Preparedness inspection with Mr. K. Morris on December 20, 2004.

4OA7 Licensee-Identified Violations

The following violation of very low significance (Green) was identified by the licensee and is a violation of NRC requirements which met the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

Cornerstone: Mitigating Systems

Technical Specification 5.4.1 required, in part, that maintenance affecting safety-related equipment be properly performed in accordance with procedures appropriate to the circumstances. Contrary to this requirement, the licensee did not consistently replenish the oil sample and purge volumes after sampling the "B" standby liquid control (SLC) pump gearbox due in part to inadequate procedural guidance. As a result, on November 2, 2004, the licensee discovered an insufficient quantity of oil in the "B" SLC pump gearbox to support operability. Subsequent evaluations determined that the pump was inoperable from July 24, 2004, through November 2, 2004, when the normal oil level was restored. This violation is not greater than green because the "A" SLC pump remained operable. This issue was entered into the licensee's corrective action program as CARD 04-25097.

ATTACHMENT: SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

W. O'Connor, Jr., Vice President Nuclear Generation
D. Cobb, Plant Manager
A. Brooks, Performance Engineering, NDE Level III
D. Craine, General Supervisor, Radiological Engineering
T. Dong, Manager, Performance Engineering
H. Higgins, Radiation Protection Manager
R. Libra, Director Nuclear Engineering
K. Morris, Emergency Preparedness Supervisor
D. Noetzel, Manager Nuclear System Engineering
J. Pendergast, Licensing Engineer
N. Peterson, Nuclear Licensing Manager
M. Philippon, Operations Manager
J. Priest, General Supervisor, Radiation Protection Operations

NRC

E. Duncan, Chief, Division of Reactor Projects, Branch 6

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000341/2004008-01	NCV	Cover Not Installed in a Cable Conduit Junction Box Penetrating a Fire Barrier
05000341/2004008-02	NCV	RPV Head Weld Examination Done in the Wrong Location
05000341/2004008-03	NCV	Fuel Bundle Struck Refueling Shield Bridge Due to Improper Installation
05000340/2004008-04	NCV	Lift of Drywell Head Over the Spent Fuel Pool
05000341/2004008-05	FIN	Failure to Clean Drywell Trench Drain
05000341/2004008-06	NCV	Failure to Perform Rattle Space Variance Evaluations for Scaffold
05000341/2004008-07	FIN	Multiple Failures of the North Main Turbine Lube Oil Pump Due to Inadequate Lubrication of Motor Bearings

Closed

05000341/2004008-01	NCV	Cover Not Installed in a Cable Conduit Junction Box Penetrating a Fire Barrier
05000341/2004008-02	NCV	RPV Head Weld Examination Done in the Wrong Location
05000341/2004008-03	NCV	Fuel Bundle Struck Refueling Shield Bridge Due to Improper Installation
05000340/2004008-04	NCV	Lift of Drywell Head Over the Spent Fuel Pool
05000341/2004008-05	FIN	Failure to Clean Drywell Trench Drain
05000341/2004008-06	NCV	Failure to Perform Rattle Space Variance Evaluations for Scaffold
05000341/2004008-07	FIN	Multiple Failures of the North Main Turbine Lube Oil Pump Due to Inadequate Lubrication of Motor Bearings
05000341/2004007-01	URI	Operability of Torus Impacted by Scaffold
05000341/2004002-04	URI	Failure of Nitrogen Inerting Primary Containment Isolation Valves to Meet IST Stroke Times

Discussed

None.

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety but rather that selected sections of portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

1R01 Adverse Weather

- Procedure 27.000.04, Rev. 28; "Freeze Protection Lineup Verification"
- Procedure 32.000.07, Rev. 33; "Reactor Building Crane Operation"
- Procedure 20.000.02, Rev. 32; "Tornado"

1R04 Equipment Alignment

- Drawing 6M721-5726; General Service Water System Functional Operating Sketch, Revision BM
- Drawing 6M721-5726-1; General Service Water System Functional Operating Sketch, Revision 0
- System Operating Procedure 23.131; General Service Water System, Revision 83
- UFSAR 9.2.1, General Service Water System
- Procedure 24.204.02, Rev. 29; "Residual Heat Removal Valve Lineup and System Filled Verification"
- Procedure 23.205, Rev. 87; "Residual Heat Removal System"
- Drawing 6M721-5706-1, Rev. AA; "Residual Heat Removal Division II Functional Operating Sketch"
- DC-5003, Rev. E; "Emergency Diesel Generator Loads Calculation"
- DC-0835, Rev. E; "System Voltage Study"
- Procedure 35.000.217, Rev. 37; "Maintenance Lubrication"
- Drawing 6I721-2201-04, Rev. P; "Schematic Diagram Residual Heat Removal Pump D"
- Drawing 6I721-2201-02, Rev. 0; "Schematic Diagram Residual Heat Removal Pump B"

1R05 Fire Protection

- CARD 04-24751; "Penetration Sealed on One Side Only" (NRC-Identified Issue)

1R06 Flood Protection

- Job ID 022304040712; Perform 43.000.001 Shore Barrier Surveillance, September 24, 2003
- UFSAR Section 2.4.2.2; Flood Design Consideration, Revision O

1R07 Heat Sink Performance

- Design Basis Document P44-00; Emergency Equipment Water System, Revision C
- Drawing 6M721-5753; Emergency Equipment Cooling Water System Division II, Revision AY
- Procedure 23.127; Reactor Building Closed Cooling Water/Emergency Equipment

Cooling Water, Revision 93

- Procedure 47.207.02; Emergency Equipment Cooling Water Division 2 Heat Exchanger Performance Test, Revision 33
- EPRI NP-7752; Heat Exchanger Performance Monitoring Guideline, December 1991

1R08 ISI

- 39.NDE.002; Magnetic Particle Examination; Revision 22; dated February 18, 2000
- 39.NDE.001; Liquid Penetrant Examination; Revision 22; dated February 22, 2000
- GE-UT-300; Procedure for Manual Examination of Reactor Vessel Assembly Welds in Accordance with PDI; Revision 5; dated August 31, 2004
- Documentation Number: B11-01-A-001-DB-002, 232-913; Closure Head Machining and Welding Drawing; Revision 4; dated July 26, 1972
- Drawing Number 6M721-5360-5; Inservice Inspection Detail Dwg. Reactor Vessel Category B-A and B-H Components Reactor Building Unit #2; dated December 15, 1984
- 43.000.016; Performance of ISI-NDE Inspections; Revision 27; dated October 17, 2003
- WR: 000Z022442; Reactor Core Isolation Cooling (RCIC) Cooling Water to Lube Oil Cooler E5100B002 Pressure Control Valve; dated February 29, 2004
- WR: 000Z032413; High Pressure Coolant Injection (HPCI) Pressure Control Valve; dated April 3, 2004
- RF 09-05; Summary Report, RPV Shell Weld; dated April 28, 2003
- CARD 04-25290; Near Miss for Incorrect Weld Inspection; dated November 8, 2004
- CARD 03-14913; Procedure 43.401.605 References Wrong Step in Acceptance Criteria; dated March 31, 2003
- CARD 03-16365; Weld Configuration Prevents Application of Newly Qualified Techniques Required by 10 CFR 50.55a for Dissimilar Welds UT Exams; dated April 3, 2003
- CARD 03-16367; Evaluate Deleting the Crevice Corrosion Cracking Degradation Mechanism from Recirc Inlet Nozzles in the RIISI Program; dated April 9, 2003
- CARD 03-16383; Evaluation of Ultrasonic Indication in RPV Weld 15-308B; dated April 25, 2003

1R12 Maintenance Rule Implementation

- Design Calculation-5750, Appendix O; Hydraulic Analysis Reports for GSW System - First Run: Pumps Operating on Design Curve - 0 percent degraded, Revision O
- Maintenance Rule Functional Failure Evaluations; P4100 GSW System, June 2002 through September 2004.

1R13 Maintenance Risk Assessment and Emergent Work

- Protected System Form MOP05002; Div. 2 CS, RHR/RHRSW, EDG 13 & 14, 120, and 345 kV
- WR H888030100; “Clean, Inspect and Re-Lube Motor Bearings”
- File No. TMSA-04-0093; “Risk Assessment for the Week of October 25, 2004"
- Plan of the Day for October 25, 2004

1R16 Operator Workarounds

- Operations Department Expectations ODE-006; Open Operations Department Challenges, October 2004
- TMSA-03-0059; Risk Assessment of Revised Operator Work Arounds, March 19, 2004
- NPOP-04-0022; Aggregate Assessment of Operator Work Arounds, March 19, 2004

1R17 Permanent Modifications

- Selected ETAP program inputs for residual heat removal pump “B”
- EDP-13231, Rev. 0; “Replacement of Residual Heat Removal Pump D Motor with Spare”
- DC-0835, Rev. E; “System Voltage Study”
- DC-5003, Rev. E; “Emergency Diesel Generator Loads Calculation”

1R19 Post Maintenance Testing

- WR 000Z043989, “RCIC failed to trip using manual trip pushbutton. Troubleshoot/repair”

1R20 Refueling & Outage Activities

- Procedure 35.710.025, Rev. 1; “Reactor Vessel Disassembly”
- MMA07, Rev. 13; “Hoisting, Rigging, and Load Handling”

1R22 Surveillance Testing

- Procedure 23.131; General Service Water System, Revision 83
- Johnston Pump Company Performance Test Set-up and Test Data Sheet, December 17, 1996
- Procedure 37.206.002, Rev. 4; “Reactor Core Isolation Cooling Turbine Overspeed Test with Motor Unit”
- STR 2004-000715

1R23 Temporary Plant Modifications

- TM 03-0016, Rev. 0; “Installation of an Interim annunciator system to support the replacement opf the annunciator sequence of event recorder systems with the new visual annunciator system per EDP 32523.”
- TM 04-0013, Rev. 0; “Transfer of annunciator windows...to the interim annunciator system and add provisions to reduce the effects of possible field induced AC voltage.”

1EP4 Emergency Action Level and Emergency Plan Changes

- Fermi Power Plant Emergency Plan; Revisions 28 and 29

2OS1 Access Control to Radiologically Significant Areas

- RWP 04-1009; Perform Walkdowns, Inspections and Supervisory Tours; Revision 3
- RWP 04-1106; Routine Mechanical Maintenance in Drywell & Steam Tunnel; Revision 0
- RWP 04-1127; Drywell Under-Vessel Replacement of IRMs and SRMs; Revision 0
- RWP 04-1160; Torus Diving; Revision 0

2OS2 ALARA Planning and Controls

- Plant Technical Procedure 63.000.200; ALARA Reviews; Revision 17
- Plant Technical Procedure 63.000.100; Radiation Work Permits; Revision 21
- Radiation Protection Conduct Manual MRP-05; ALARA/RWPs; Revision 5
- Historical Fermi 2 Outage Dose Information for RF-01 through RF-09
- Daily RF-10 Dose Data/Graphs and RWP Activity Reports for November 15-19, 2004
- RWP 04-1251 (Revision 0); Associated Radiological Planning Checklist and ALARA Review; Perform Refuel Activities on RB-5
- RWP 04-1160 (Revision 0); Associated Radiological Planning Checklist and ALARA Review; Torus Diving - Desludge, Inspect/Repair Coating Under Water
- RWP 04-1106 (Revision 0); Associated Radiological Planning Checklist and ALARA Review; Routine Mechanical Maintenance in the Drywell and Steam Tunnel
- RWP 04-1113 (Revision 0); Associated Radiological Planning Checklist and ALARA Review; Control Rod Drive Exchange
- RWP 04-1115 (Revision 0); Associated Radiological Planning Checklist and ALARA Review; Safety Relief Valves - Remove and Replace
- RWP 04-1105 (Revision 0); Associated Radiological Planning Checklist and ALARA Review; Scaffold Activities in the Steam Tunnel and Drywell
- RWP 04-1110 (Revision 0); Associated Radiological Planning Checklist and ALARA Review; In-Service Inspections in the Drywell and Steam Tunnel
- RWP 04-1108 (Revision 0); Associated Radiological Planning Checklist and ALARA Review; Insulation Removal, Repair and Replacement
- Job Progress ALARA Review for RWP 04-1121; E1100 and E4100 System Component Repairs and Maintenance in the Drywell and Steam Tunnel; dated November 13, 2004
- Job Progress ALARA Reviews for RWP 04-1115; Safety Relief Valve Removal and Replacement; dated November 8 and 14, 2004
- Job Progress ALARA Reviews for RWP 04-1160; Torus Diving - Desludge, Inspect/Repair Torus Coating; dated November 9 and 15, 2004
- Job Progress ALARA Review for RWP 04-1151; Torus Hatch Removal; dated November 15, 2004
- RWP 04-1160 Access Detail (Daily Dose) Report for November 5-17, 2004
- Attachment 1 to Form 63.000.100; Respirator Evaluation Worksheet; RWP 04-1119 for Work on Valve B2100F0101B; dated November 16, 2004
- Attachment 1 to Form 63.000.100; Respirator Evaluation Worksheet; RWP 04-1023 for Work Inside E1100F031A/B Systems; dated October 23, 2004
- Fermi 2 Cobalt Reduction Plan; dated November 1998
- Shift Manager Log Entry Summaries Related to System/Component Flushes for Various RF-10 Outage and Pre-Outage Dates
- Historical Fermi 2 BWR Radiation Assessment and Control (BRAC) Survey Data and BRAC Point Graphs for RF-01 through RF-10

- Summary Data of Level 1 and Level 2 Personnel Contaminations for RF-10 through November 17, 2004
- Whole Body Count Reports for Selected Individuals for RF-10 through November 18, 2004
- RF-09 Lessons Learned Report and Critique Item Summary (undated)
- CARD 04-25661; Weaknesses in Radiological Surveys; dated October 16, 2004
- CARD 04-25626; Radiation Protection Boundary Weaknesses; dated November 15, 2004
- CARD 04-25280; RWP Violation; dated November 8, 2004
- CARD 04-25570; Worker Exited Protected Area and Took TLD Home; dated November 14, 2004
- CARD 04-25701; Worker Violation of RWP Requirements; dated November 17, 2004
- CARD 04-25717; Investigate PDE4 Dose Alarms; dated November 17, 2004

4OA1 Performance Indicator Verification

- Reactor core isolation cooling maintenance rule out of service hours from January 1, 2004 through December 31, 2004
- Reactor core isolation cooling maintenance rule functional failure evaluations from January 1, 2004 through December 31, 2004

4OA3 Event Follow-Up

- CARD 04-24282; "Scaffolding Touching the Torus"
- Past operability evaluation for CARD 04-24282
- CARD 04-24040; "Reactor SCRAM on AVR Relay Trip"
- TM 04-0020, Rev. A
- CARD 04-26443; "Reactor SCRAM on AVR Trip"
- ARP 4D53, Rev. 9; "AVR General Alarm"

LIST OF ACRONYMS USED

ALARA	As Low As Is Reasonably Achievable
ASME	American Society of Mechanical Engineers
BI	Barrier Integrity
CARD	Condition Assessment Resolution Document
CFR	Code of Federal Regulations
EDP	Engineering Design Package
GSW	General Service Water
HPCI	High Pressure Coolant Injection
IMC	Inspection Manual Chapter
ISI	Inservice Inspection
MT	Magnetic Particle Examination
NCV	Non Cited Violation
NDE	Nondestructive Examination
NRC	Nuclear Regulatory Commission
PI	Performance Indicator
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
RF-10	Tenth Refueling Outage for Fermi-2
RPV	Reactor Pressure Vessel
RWP	Radiation Work Permit
SDP	Significance Determination Process
SLP	Standby Liquid Control
TM	Temporary Modification
TS	Technical Specifications
UFSAR	Updated Final Safety Assessment Report
WR	Work Request