November 1, 2004

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 05000341/2004007

Dear Mr. O'Connor:

On September 30, 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 September 29, 2004, 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, three findings of very low safety significance, all 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.

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.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

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

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

Enclosure: Inspection Report 05000341/2004007
w/Attachment: Supplemental Information

cc w/encl: N. Peterson, Manager, Nuclear Licensing
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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-341
License No:	DPR-43
Report No:	05000341/2004007
Licensee:	Detroit Edison Company
Facility:	Fermi Power Plant, Unit 2
Location:	6400 N. Dixie Hwy. Newport, MI 48166
Dates:	July 1 through September 30, 2004
Inspectors:	S. Campbell, Senior Resident Inspector T. Steadham, Resident Inspector W. Slawinski, Senior Radiation Specialist D. McNeil, Senior Operator License Examiner
Approved by:	E. Duncan, Chief Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000341/2004007; 07/01/2004-09/30/2004; Fermi Power Plant, Unit 2; Surveillance Testing; Problem Identification and Resolution; Event Follow-Up.

This report covers a 3-month period of baseline resident inspection and announced baseline inspections in radiation protection and operator licensing. The inspection was conducted by the resident inspectors, a Region III senior radiation specialist inspector, and an operator licensing examiner. Three Green findings associated with three Non-Cited Violations (NCVs) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, “Significance Determination Process” (SDP). Findings for which the SDP does not apply may be “Green” or be assigned a severity level after NRC management review. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance for the failure to incorporate the requirements and acceptance limits contained in applicable design documents into the surveillance tests for the High Pressure Coolant Injection (HPCI) and the Reactor Core Isolation Cooling (RCIC) systems.

  This finding was determined to be more than minor because if left uncorrected, it would become a more significant safety concern since surveillance testing would not have ensured that the HPCI and RCIC design functions were able to be accomplished if system performance degraded. The finding was of very low safety significance because there was no actual loss of safety function of either system. A Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XI, “Test Control,” was identified. Immediate corrective actions included ensuring that both systems remained operable through analysis or testing. (Section 1R22.1)

- Green. The inspectors identified a finding of very low safety significance for the failure to quarantine degraded components inside the motor operator for the HPCI Turbine Steam Supply Outboard Containment Isolation Valve after the valve failed to close during surveillance testing activities. The primary cause of this finding was related to the cross-cutting area of human performance.

  This finding was determined to be more than minor because if left uncorrected, it could become a more significant safety concern since the failure to quarantine degraded components could impede the identification of root causes for conditions adverse to quality and prevent the implementation of appropriate corrective actions to prevent their recurrence. The finding was of very low safety significance because the finding was not a design or qualification deficiency resulting in a loss of function per Generic Letter 91-18; did not represent an actual loss of safety function of a system or the loss of safety function of a train of equipment; and was not potentially risk-significant due to a seismic, fire, flooding, or severe weather initiating event. A Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” was identified. As part of the licensee’s immediate corrective actions, the wiring that was inappropriately discarded was retrieved. (Section 4OA2.2)

- Green. A finding of very low safety significance was self-revealed when a compression fitting on a seal cooling line for Residual Heat Removal (RHR) Pump “A” separated rendering the pump inoperable. The pump was operating in the shutdown cooling mode at the time of the event. Design control measures during plant construction were inadequate in that a ferrule on a compression fitting was constructed of carbon steel instead of stainless steel, as required.

  The finding was determined to be more than minor because the finding increased the likelihood of a loss of decay heat removal, affected the licensee’s ability to add reactor coolant system inventory, and degraded the licensee’s ability to establish an alternate core cooling path. The finding was of very low safety significance because the makeup capability of the control rod drive pumps, which were in service at the time of the event, exceeded the leakage rate through the failed seal cooling line. A Non-Cited Violation of 10 CFR 50, Appendix B, Criterion III, “Design Control,” was identified. Immediate corrective actions included replacing the carbon steel ferrule with a stainless steel ferrule. (Section 4OA3.1)

B. Licensee-Identified Violation

One 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 numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 2 began this inspection period at full power and remained at or near full power until August 8, 2004, when a plant shutdown commenced in accordance with Technical Specifications (TS) requirements due to a blower failure on Emergency Diesel Generator (EDG) 12.

On August 17, 2004, operations personnel commenced a reactor startup and declared the reactor critical. Full power was achieved on August 19, 2004.

Unit 2 continued to operate at or near full power until September 3, 2004, when an automatic reactor scram occurred as a result of a main generator automatic voltage regulator failure. On September 5, 2004, operators commenced a reactor startup and declared the reactor critical. Full power was achieved on September 8, 2004.

Unit 2 operated at or near full power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, Barrier Integrity

1R05 Fire Protection (71111.05Q)

a. Inspection Scope

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

- Reactor Building First Floor Steam Tunnel;
- Auxiliary Building Basement, T-Room;
- Reactor Building - Fourth Floor;
- Reactor Building - Basement, Corner Rooms; and
- Main Control Room.

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.

b. Findings

No findings of significance were identified

1R11 Licensed Operator Requalification (71111.11)

.1 Quarterly Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

On September 8, 2004, the inspectors observed an operations crew during the annual requalification examination in mitigating the consequences of events in Scenarios SS-OP-904-0004, "Instrument Failure, Uncoupled Rod, Loss of General Service Water, Reactor Pressure Vessel Flooding," and SS-OP-904-0009, "Instrument Failure, Jet Pump Failure, and Secondary Containment/Radiation Release Emergency Operating Procedure," on the simulator. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of abnormal and emergency procedures;
- control board manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications.

The crew's performance in these areas was compared to pre-established operator action expectations and successful critical task completion requirements. This activity represented two inspection samples.

b. Findings

No findings of significance were identified.

.2 Biennial Written Examination and Annual Operating Test Results (71111.11B)

a. Inspection Scope

The inspectors reviewed the overall pass/fail results of Job Performance Measure (JPM) operating tests, and simulator dynamic operating tests (required to be given per 10 CFR 55.59(a)(2)) administered by the licensee during calendar year 2004. The overall results were compared with the Significance Determination Process in accordance with NRC Manual Chapter 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process (SDP)."

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12Q)

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the Reactor Core Isolation Cooling (RCIC) 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;
- characterizing system reliability issues;
- trending key parameters (condition monitoring);
- 10 CFR 50.65(a)(2) classification; and
- appropriate performance criteria.

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

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 four maintenance and operational activities affecting safety-related equipment:

- Inadvertent half containment isolation due to failure of Channel "A" high steam tunnel temperature instrumentation, Condition Assessment Resolution Document (CARD) 04-15588;
- Division 1 RHR safety system outage;
- Forced outage due to EDG 12 blower failure; and
- Division 1 Ultimate Heat Sink and Emergency Equipment Cooling Water (EECW) safety system outage.

These activities were selected based on their potential risk significance relative to the reactor safety cornerstones.

As applicable for each of the above activities, 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.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

.1 Routine Review of Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following seven CARDs to ensure that either the condition did not render the involved equipment inoperable or result in an unrecognized increase in plant risk, or the licensee appropriately applied TS limitations and appropriately returned the affected equipment to an operable status:

- CARD 04-23247, High Pressure Coolant Injection (HPCI) Maximum Operating Pressure;
- CARD 04-23362, RCIC Maximum Operating Pressure;
- CARDs 04-23647 and 04-04-23689, E4150F003 Environmental Qualification Operability Evaluation;
- CARD 04-23258, Mechanical Draft Cooling Tower Fan Inlet Grating Covered with Plywood;
- CARD 04-24103, EDG 13 Air Intake Filter/Silencer Supports;
- CARD 04-24134, E41N061D Failure; and
- CARD 04-24282, Scaffolding Too Close to the Torus.

b. Findings

Introduction

During a walkdown of the torus room on September 14, 2004, the inspectors identified a scaffold that was not in compliance with the licensee's documented rattlespace requirements. Procedure MMA08, "Scaffolding," required an engineering review for all scaffolds within 3 inches horizontally of any safety-related structure or component. The inspectors noted two scaffolds that were within 3 inches of the torus with one of them having a horizontal member in direct contact with the torus.

Both scaffolds had approved seismic variances so the inspectors requested a copy of the evaluation. However, since MMA08 did not require the seismic variance to be documented and it was approved through "engineering judgement," no documented evaluation was available. The licensee concluded that the scaffolding did not have sufficient rigidity to affect torus movement during a transient or seismic event and thus would have a negligible impact on the torus. The licensee based their conclusion on their understanding that the only scaffold member in contact with the torus was the upper portion of an unrestrained vertical member.

The inspectors questioned the licensee on the suitability of such an evaluation because plant Design Specification 3071-031, Appendix H, specified a rattlespace value of 3 inches which included "seismic, safety relief valve discharge and thermal expansion displacements." Upon review of the licensee's 10 CFR 50.59 screening form for Procedure MMA08, the inspectors identified that the procedure's rattlespace criteria considered only the seismic effects of scaffolding interacting with safety-related components. The licensee did not review the dynamic interactions between the torus and the scaffolding that would occur during a design basis event, such as a loss-of-coolant accident.

The inspectors determined that the scaffold contacting the torus would place more than a negligible load on the torus during a transient which could impact the torus response harmonics and thus impact the torus attached-piping evaluations. The licensee concluded that they had not fully analyzed the scaffold, immediately removed the scaffolding on September 17, 2004, and entered this issue into their corrective action program as CARD 04-24282.

During scaffold removal, the licensee discovered a 2-foot long horizontal member erected between and in direct contact with both the torus room wall and the torus. During a transient, this member could have restricted the horizontal movement of the torus, placing a significant localized stress on the torus.

As a result of this discovery, the licensee suspended all scaffold erections in safety-related areas. The licensee developed a process to formally document the evaluation performed to support a variance from the MMA08 rattlespace requirements. Utilizing this new process, the licensee re-evaluated all scaffolds with seismic variances and found no significant deficiencies. This is an Unresolved Item (URI 05000341/2004007-01) pending the inspectors' review of the final past operability determination of the torus.

1R19 Post Maintenance Testing (71111.19Q)

.1 Inadequate Reactor Core Isolation Cooling Pump Acceptance Criteria

a. Inspection Scope

The inspectors reviewed the scope of the work performed for Work Requests (WRs) 000Z033972, 000Z032665, and 000Z040116 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 the acceptance criteria were met. The inspectors interviewed operations, maintenance, and engineering department personnel and reviewed the completed post maintenance testing documentation. This constituted one sample.

b. Findings

No findings of significance related to post maintenance testing were identified; however, one finding having very low safety significance (Green) associated with inadequate acceptance criteria is discussed in Section 1R22.1 of this report.

.2 Routine Review of Post Maintenance Testing

a. Inspection Scope

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

- WR G290030100 for Division 1 Essential Equipment Cooling Water Room Cooler, T4100B034, repair;
- WR W8400401P4 for EDG 12 safety system outage;
- WR 000Z042219 for RHR Pump "A" seal water cooling line repair;
- WR 000Z042311 for RHR Pump "C" seal water cooling line repair;
- WR 000Z042333 and 000Z973751 for HPCI turbine outboard steam isolation valve, E4150F003, repair;
- WR 000Z042279 for EDG 12 repair;
- WR 000Z020632 for HPCI turbine inboard steam isolation valve, E4150F002, refurbishment;
- WR 000Z042377 for E4150F003 bypass valve packing adjustment; and
- WR 000Z042638 for Division 2 torus level instrumentation isolation valve, E41F402, limit switch adjustment.

The inspectors reviewed the scope of the work performed and evaluated the adequacy of the specified post maintenance testing. The inspectors verified that the post maintenance testing was performed in accordance with approved procedures, that 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 that post maintenance testing problems were entered into the corrective action program with the appropriate significance characterization.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors observed the licensee's performance during the August 2004 forced outage due to the EDG 12 blower failure. These inspection activities represented one inspection sample.

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 frequently 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 reactor coolant system 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.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

.1 Inadequate High Pressure Coolant Injection and Reactor Core Isolation Cooling Pump Acceptance Criteria

a. Inspection Scope

The inspectors observed the HPCI pump time response and operability test performed on July 13, 2004. The inspectors reviewed the test results to determine whether risk significant systems and equipment were capable of performing their intended safety function and to verify testing was conducted in accordance with applicable procedural and TS requirements. The inspectors also reviewed the test procedure to ensure that acceptance criteria were appropriate.

These inspection activities represented one inspection sample.

b. Findings

Introduction

The inspectors identified a finding of very low safety significance (Green) when licensee personnel failed to establish appropriate test criteria to ensure that the HPCI and RCIC systems could perform their intended safety function. A Non-Cited Violation (NCV) of 10 CFR 50, Appendix B, Criterion XI, “Test Control,” associated with the HPCI system was also identified.

Description

The inspectors reviewed the acceptance criteria in Procedure 24.202.01, “HPCI Pump Time Response and Operability Test,” to determine whether the Inservice Testing (IST) acceptance criteria was adequate to ensure HPCI could perform its safety function at design pressure. The inspectors determined that the HPCI pump IST acceptance criteria could have allowed pump performance to degrade below the level where the system was able to perform its safety function. NRC Information Notice 97-90, “Use of Non-Conservative Acceptance Criteria in Safety-Related Pump Surveillance Tests,” had previously alerted licensees to this potential test acceptance criteria deficiency.

In January 1998, General Electric Nuclear Energy (GENE) performed an analysis to determine the effects of revising the Safety Relief Valve (SRV) setpoint tolerance from plus or minus one percent (+/- 1%) to plus or minus three percent (+/- 3%) and issued the results in GENE report NEDC-32789P. One of the effects of revising this tolerance was that the maximum reactor pressure that the HPCI and RCIC systems were required to inject against increased from 1146 pounds per square inch gauge (psig) to 1169 psig. Based on the results of this analysis, the licensee determined the required HPCI and RCIC pump speeds increased to 4060 revolutions per minute (rpm) and 4535 rpm, respectively, to inject at the higher pressures.

To meet these higher speed requirements, the licensee set the HPCI and RCIC turbine governor high speed setting to 4100 rpm and 4600 rpm, respectively. The effect of this speed setting on the HPCI pump, for example, was that the governor control logic would limit the turbine’s speed to no more than 4100 rpm even if the demand flow rate required higher speeds. The inspectors questioned whether the high speed setting was sufficient to ensure that HPCI could perform its intended safety function if pump performance were to degrade to the minimum allowable IST acceptance criteria.

The licensee initiated CARD 04-23296 to determine if the IST acceptance criteria for the HPCI pump was appropriate. In the interim, the inspectors performed an independent calculation and determined the current performance of the HPCI pump and existing turbine speed limiter setting was adequate to ensure the system design function would be met. However, the inspectors determined that a 1 percent degradation from the current level of pump performance could prevent the HPCI system from achieving its design flow rate at design pressure.

Concerned that a similar condition could exist with the RCIC pump, the inspectors reviewed the most recent performance data of the RCIC pump and determined it also was not adequate to ensure the system design function would be met. As a result of the inspectors’ calculations, the inspectors questioned the operability of both the HPCI and RCIC systems.

To evaluate the conditions, the licensee initiated CARDs 04-23362 and 04-23363 and performed an operability determination in accordance with Generic Letter 91-18. The licensee's calculations generally agreed with the inspector's insofar as the RCIC pump could not meet the required pressure and flow using the last recorded test data and that a 1 percent decrease in HPCI pump performance would have prevented it from achieving its design flow rate at design pressure.

The licensee discussed the results of their RCIC operability determination with the inspectors and noted that the turbine speed was recorded using an unreliable tachometer. Since the RCIC pump was not scoped in the IST program, an accurate recording of pump speed using a digital tachometer was neither performed nor required by the surveillance test. Since the recorded turbine speed could not be used for the purposes of determining operability, the licensee performed a RCIC test 2 days later using IST-quality instruments.

Both the licensee's and the inspectors' analyses of the new test data confirmed that the current performance of the RCIC pump was adequate to ensure that the system design function would be met in the current (as-found) condition. However, the inspectors determined that a 2 percent degradation from the current level of pump performance could prevent the RCIC system from achieving its design flow rate at design pressure.

Existing surveillance test procedure acceptance criteria was non-conservative because it allowed HPCI pump performance degradation up to a nominal 10 percent from vendor curve test performance. Similarly, a substantial degree of RCIC pump performance degradation from the vendor curve test performance was noted. This degradation level would result in the inability of both pumps to achieve their design flow rate against a reactor pressure of 1169 psig. Therefore, the inspectors concluded that corrective actions were required to ensure the capability of both systems to achieve their design functions.

Analysis

The inspectors determined that the failure to incorporate the requirements and acceptance limits contained in applicable design documents into the surveillance tests for the HPCI and RCIC systems 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 was left uncorrected and pump performance were to degrade further, it could lead to the undetected inability to meet design basis requirements which is a more significant safety concern since the surveillance test would not have ensured that system design functions were maintained.

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 B [Barrier Integrity] Cornerstones." The inspectors concluded that this finding affected the Mitigating Systems cornerstone. However, since the finding did not result in an actual loss of safety function per Generic Letter 91-18, this finding was considered to be of very low safety significance (Green).

Enforcement

10 CFR 50, Appendix B, Criterion XI, “Test Control,” requires, in part, that tests be performed in accordance with written procedures which incorporate the requirements and acceptance limits contained in applicable design documents. Contrary to this requirement, following a safety relief valve setpoint tolerance revision in January 1998, the acceptance criteria contained in Procedure 24.202.01, Revision 80, “HPCI Pump Time Response and Operability Test,” was inadequate to ensure that the HPCI system could perform its safety function at design conditions. 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 05000341/2004007-02) consistent with Section VI.A of the NRC Enforcement Policy.

Since RCIC is not relied upon in the licensee’s accident analysis, the requirements of 10 CFR 50, Appendix B, did not apply to the RCIC system and no violation of regulatory requirements was identified.

Once identified, the licensee entered this issue into their corrective action program as CARDs 04-23362 and 04-23363 and performed an operability determination which verified that both systems were currently operable. On September 30, 2004, the licensee revised the applicable acceptance criteria for HPCI. Although the RCIC surveillance test had not been revised by the end of this inspection period, the inspectors verified that the licensee’s planned corrective actions included the applicable revisions before the next scheduled RCIC pump test.

.2 Routine Surveillance Test Reviews

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 24.206.01, RCIC flow test at 1025 psi;
• Procedure 24.107.13, Standby feedwater pump flow test;
• NUREG-1482 requirement to stroke test E4150F600 following packing adjustment; and
C Procedure 24.000.02, Shiftly, daily and weekly required surveillances, Attachment 1, eight hour -- mode 1,2,3 -- control room, reactor coolant system operational leakage.

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.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the following two temporary modifications:

- TM 04-0020, install three temporary auctioneering diodes in series with 'G05' 24-volt power supply connections at the input terminals of the rectifier converter electronic control modules, and

- TM 04-0013, upgrade the annunciator system.

The inspectors verified the installations were consistent with design modification documents and the modifications did not adversely impact system operability or availability. 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, operations, and maintenance department personnel and reviewed the design modification documents.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS3 Radiation Monitoring Instrumentation and Protective Equipment (71121.03)

.1 Inspection Planning

a. Inspection Scope

The inspectors reviewed the Fermi 2 Updated Final Safety Analysis Report (UFSAR) to identify applicable radiation monitors associated with measuring transient high and very high radiation areas including those used in remote emergency assessment. The inspectors identified the types of portable radiation detection instrumentation used for job coverage of high radiation area work including instruments used for underwater surveys, fixed Area Radiation Monitors (ARMs) used to provide radiological information in various plant areas and Continuous Air Monitors (CAMs) used to assess airborne radiological conditions and consequently work areas with the potential for workers to receive a 50 millirem or greater Committed Effective Dose Equivalent (CEDE).

Contamination monitors, whole body counters, and those radiation detection instruments utilized for the release of personnel and equipment from the Radiologically Restricted Area (RRA) were also identified.

These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

.2 Walkdowns of Radiation Monitoring Instrumentation

a. Inspection Scope

The inspectors conducted walkdowns of selected ARMs in the Turbine and Reactor Buildings to verify they were located as described in the UFSAR, were optimally positioned relative to the potential source(s) of radiation they were intended to monitor, and to verify that control room instrument readout and high alarm setpoints for those ARMs were consistent with UFSAR information. Walkdowns were also conducted of those areas where portable survey instruments were calibrated/repaired and maintained for Radiation Protection (RP) staff use to determine if those instruments designated "ready for use" were sufficient in number to support the RP program, had current calibration stickers, were operable, and were in good physical condition. Additionally, the inspectors observed the licensee's instrument calibration units and the radiation sources used for instrument checks to assess their material condition, and discussed their use with RP staff to determine if they were used adequately. Licensee personnel were also observed performing source checks of selected instruments as they were logged out for use.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

.3 Calibration and Testing of Radiation Monitoring Instrumentation

a. Inspection Scope

The inspectors selectively reviewed radiological instrumentation associated with monitoring transient high and/or very high radiation areas, instruments used for remote emergency assessment and for post accident sampling, and radiation monitors used to identify personnel contamination and for assessment of internal exposures to verify that the instruments had been calibrated as required by the licensee's procedures, consistent with industry and regulatory standards. The inspectors also reviewed alarm setpoints for selected ARMs to verify that they were established consistent with the UFSAR and TSs, as applicable. In particular, the inspectors reviewed calibration procedures and the most recent calibration records and/or source characterization/output verification documents for the following radiation monitoring instrumentation and instrument calibration equipment:

- Containment Area High Range Radiation Monitors (both divisions);
- Traversing In-Core Probe (TIP) Room ARM (channel 12);
- Torus Room ARM (channel 14);
- Refuel Floor High Range ARM (channel 18);
- First Floor Drywell ARM (channel 45);
- Small Article Monitors (SAMs) used at the main and alternate RRA egresses;
- J. L. Shepherd Instrument Calibrators (box, panoramic and beam calibrators);
- Electrometer and the associated ion chambers used for measuring the output of the instrument calibrators (vendor calibration);
- Portable survey instruments used for underwater surveys (three instruments);
- Standup (Fastscan) and Chair Type Whole Body Counters;
- Portal Monitors used at the Primary Access Portal (two instruments); and
- Personnel Contamination Monitors used at the main and alternate RRA egresses (four units).

The inspectors discussed the operability and maintenance of the Post Accident Sampling System (PASS) with chemistry staff and reviewed PASS surveillance records for 2004 to determine if system function was demonstrated consistent with the licensee's chemistry procedures. Contingency plans for obtaining highly radioactive samples of reactor coolant which the licensee implemented as a commitment following the issuance of Amendment No. 150 to its Facility Operating License (which eliminated the TSs requirements for the PASS) were also reviewed to ensure the licensee's commitments were being met.

The inspectors determined what actions were taken when, during calibration or source checks, an instrument was found significantly out of calibration or exceeded as-found acceptance criteria. Should that occur, the inspectors verified that the licensee's actions would include a determination of the instrument's previous use and the possible consequences of that use since the prior calibration. The inspectors also reviewed the licensee's 10 CFR Part 61 source term information to determine if the calibration sources used were representative of the plant source term and that difficult to detect nuclides were scaled into whole body count dose determinations.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

.4 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed licensee CARDs and any special reports that involved personnel contamination monitor alarms due to personnel internal exposures to verify identified problems were entered into the corrective action program for resolution. Licensee audits and CARDs were also reviewed to verify deficiencies and problems with radiological instrumentation, the radiation monitoring system, or Self-Contained Breathing Apparatus (SCBA) were identified, characterized, prioritized, and resolved effectively using the corrective action program.

The inspectors reviewed corrective action program reports related to exposure-significant radiological incidents that involved radiation monitoring instrument deficiencies since the last inspection in this area, as applicable. Members of the RP staff were interviewed and corrective action documents were reviewed to verify that follow-up activities were being conducted in an effective and timely manner commensurate with their importance to safety and risk based on the following:

- 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 inspectors determined if the licensee's self-assessment and/or audit activities were identifying and addressing repetitive deficiencies or significant individual deficiencies in problem identification and resolution.

These reviews represented three inspection samples.

b. Findings

No findings of significance were identified.

.5 Radiation Protection Technician Instrument Use

a. Inspection Scope

The inspectors selectively verified that calibrations for those instruments recently used and for those designated for use had not lapsed. The inspectors reviewed instrument logs to verify that response checks of portable survey instruments were completed prior to instrument use and upon return of the instrument to the storage area after use, as required by the licensee's procedure. The inspectors also discussed instrument calibration methods and source response check practices with RP staff and observed staff complete instrument operability checks prior to use.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

.6 Self-Contained Breathing Apparatus Maintenance/Inspection and User Training

a. Inspection Scope

The inspectors reviewed aspects of the licensee's respiratory protection program for compliance with the requirements of Subpart H of 10 CFR Part 20, and to determine if SCBAs were properly maintained and ready for emergency use. The inspectors reviewed the status, maintenance and surveillance records of SCBAs staged and ready for emergency use in various areas of the plant and assessed the licensee's capability for refilling and transporting SCBA air bottles to and from the control room and Operations Support Center (OSC) during emergency conditions. The inspectors verified that all control room staff designated for the active on-shift duty roster, including those individuals on the station's fire brigade, were trained, respirator fit-tested, and medically certified to use SCBAs. Additionally, the inspectors reviewed SCBA qualifications for the emergency response organization's radiological emergency team and for the damage control and rescue team to determine if a sufficient number of staff were qualified to fulfill emergency response positions to meet the requirements of 10 CFR 50.47. The inspectors also reviewed respiratory protection training lesson plans to assess their overall adequacy and for compliance with Subpart H, and to verify that personal SCBA air bottle change-out was adequately addressed.

The inspectors walked down the bottled air supply rack and spare air bottle stations located outside the main control room, and inspected SCBA equipment maintained in the control room and SCBA equipment staged for emergency use in various areas of the plant. During the walkdowns, the inspectors examined several SCBA units to assess their material condition, to verify that air bottle hydrostatic tests were current, and to verify that bottles were pressurized to meet procedural requirements. The inspectors reviewed records of SCBA equipment inspection and functional testing and observed an RP technician complete a functional test to determine if these activities were performed consistent with the procedure and the equipment manufacturer's recommendations. The inspectors also ensured that the required, periodic air cylinder hydrostatic testing was documented and up to date, and that the Department of Transportation required retest air cylinder markings were in place for several randomly selected SCBA units. Additionally, the inspectors reviewed vendor training certificates for those individuals involved in the repair of SCBA pressure regulators to determine if those personnel who performed maintenance on components vital to equipment function were qualified. The most recent vital component (regulator) test records were reviewed by the inspectors for selected SCBA equipment currently designated for emergency use.

These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification (71151)

Cornerstone: Public Radiation Safety

.1 Radiation Safety Strategic Area

a. Inspection Scope

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

- Radiological Effluent TS/Offsite Dose Calculation Manual Radiological Effluent Occurrence

The inspectors reviewed the licensee's CARD database and selected CARDs generated between January 2003 and September 2004 to identify any potential occurrences such as unmonitored, uncontrolled or improperly calculated effluent releases that may have impacted offsite dose. The inspectors reviewed gaseous effluent summary data, the results of associated offsite dose calculations, and quarterly PI verification records generated between July 2003 and August 2004 to determine if indicator results were accurately reported. Additionally, the inspectors discussed with the RP technical staff its methods for quantifying effluents and determining effluent dose.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

Paragraph 40A2.1 does not represent an inspection sample. Paragraphs 40A2.2 and 40A2.3 each represent one inspection sample.

.1 Routine Review of Identification and Resolution of Problems

Introduction

As discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's corrective action program at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed.

a. Effectiveness of Problem Identification

(1) Inspection Scope

The inspectors reviewed CARDs 04-23247, 04-23296, 04-22915, 04-24282, 04-23647, and 04-23668 to verify that the licensee's identification of the problems were complete, accurate, and timely, and that the consideration of extent of condition review, generic implications, common cause, and previous occurrences was adequate.

(2) Issues

HPCI and RCIC Test Acceptance Criteria (CARDs 04-23247 and 04-23296)

As described in Section 1R22.1 of this report, the inspectors identified non-conservative acceptance criteria for both the RCIC and HPCI pumps. The licensee entered the original concern into their corrective action program as CARD 04-23247, but failed to identify the non-conservative acceptance criteria without the inspectors' involvement whereupon they initiated CARD 04-23296. Licensee management stated that CARD 04-23247 would have been closed with no further evaluation had the inspectors not questioned the licensee's conclusion; therefore, the inspectors considered this to be an example of ineffective problem identification.

Torus Room Scaffold (CARDs 04-22915 and 04-24282)

CARDs 04-22915 and 04-24282 both relate to scaffolds installed within 3 inches of the torus. On April 2, 2004; June 29, 2004; and July 2, 2004; the inspectors brought to the licensee's attention several scaffolds that were too close to the torus. In each case, the licensee corrected the deficiencies, but failed to identify the significance of scaffold being too close to the torus.

As documented in Section 1R15.1 of this report, the inspectors identified a scaffold that was contacting the torus during a walkdown on September 14, 2004. The licensee's original conclusion that the scaffold was acceptable was based, in part, on an inadequate identification of the actual scaffold configuration. Upon closer examination of the configuration, the licensee determined that a past operability concern of the torus existed.

These four instances represented four missed opportunities for the licensee to identify the programmatic scaffolding deficiencies that were ultimately discovered as a result of the inspectors' involvement; therefore, the inspectors considered this to be an example of ineffective problem identification.

Degraded Wires on HPCI Valve E4150F003 (CARDs 04-23647 and 04-23668)

The licensee discovered missing insulation on a portion of the HPCI steam line below the limit switch compartment for the motor operator and initiated CARD 04-23668 to document the issue. Heat generated from the exposed piping may have caused overheating of the wire insulation inside the limit switch compartment. As a result, the licensee conducted an extent of condition review that included an inspection of the reactor building steam tunnel, which was the location of the valve.

The licensee identified insulation concerns with three other valves. First, the HPCI injection valve into the feedwater line, E4150F006, had insulation removed to prevent steam voiding and HPCI pressure transients during HPCI operation. Appropriate design control processes were used to remove this insulation and since this line was not a steam line, the missing insulation would have no impact on the wires. Second, the licensee discovered missing insulation on the bypass valve, E4150F600, around E4150F003. The licensee justified this condition in that the limit switch compartment was too far from the uninsulated valve and determined not to impact the wires. Finally, missing insulation was discovered on the RCIC steam supply line. The licensee concluded that because of the location of this pipe, this did not impact the wires inside the valve.

The inspectors conducted a steam tunnel closeout to identify any additional examples of missing insulation while operators heated up the plant for startup. During the walk down, the inspectors discovered damaged insulation on the bypass line around E4150F003. No evaluation of this damaged insulation was included in the CARD. In this example, the licensee missed an opportunity to address the extent of condition related to thermal aging of wires caused by missing and/or damaged insulation in the steam tunnel. If this was not identified, further aging of the replaced wires could have occurred. The licensee initiated CARD 04-23707 and repaired the insulation under a minor work request.

b. <u>Prioritization and Evaluation of Issues</u>

(1) <u>Inspection Scope</u>

The inspectors reviewed CARDs 04-23247, 04-23296, and 04-24282. The inspectors considered the licensee’s evaluation and disposition of performance issues, evaluation and disposition of operability issues, and application of risk insights for prioritization of issues.

(2) <u>Issues</u>

<u>HPCI and RCIC Test Acceptance Criteria (CARDs 04-23247 and 04-23296)</u>

The licensee initiated CARD 04-23247 to evaluate HPCI’s ability to meet its design function. The inspectors determined that although the licensee appropriately prioritized this issue, their evaluation was inadequate. Specifically, several licensee engineers reviewed this issue and determined that no issue existed with either HPCI or RCIC because the effects of the SRV setpoint relaxation had been previously identified, evaluated, and incorporated.

When the licensee discussed their conclusions to CARD 04-23247 with the inspectors, the inspectors questioned if the revised speed requirements accounted for pump degradation and if the IST acceptance criteria were non-conservative. The licensee evaluated this scenario, determined that the acceptance criteria for HPCI could be non-conservative, and initiated CARD 04-23296 to document the issue. The inspectors concluded that evaluating the effects of pump degradation against the maximum required pump speed was required to thoroughly evaluate the issue identified in CARD 04-23247. Therefore, the inspectors considered this to be an example of an inadequate evaluation of an issue.

Torus Room Scaffold (CARDs 04-22915 and 04-24282)

As documented in Section 1R15.1 of this report, the inspectors identified a scaffold in contact with the torus during a walkdown on September 14, 2004 that had been previously evaluated and approved by engineering. After the inspectors questioned the scaffold, the licensee re-evaluated it and incorrectly concluded that the scaffold did not have sufficient rigidity to affect torus movement.

The inspectors discussed their concerns with the evaluation with the licensee on four occasions over the next 3 days before the licensee removed the scaffolding to alleviate the inspectors' concerns. Before removing the scaffold, the licensee took detailed pictures to document the as-found condition which identified a 2-foot long horizontal member essentially wedged between the torus and a concrete wall.

The licensee determined that this member could have significantly impacted the torus and that those effects were not properly reviewed in their prior evaluations. This issue identified a weakness in their scaffolding program in that only seismically-induced movement of scaffolding interacting with plant equipment was analyzed. This issue represented a scenario where the torus could have been restrained from normal movement during a transient which was previously outside the scope of the licensee's evaluations. Because of the inspector involvement that was required to resolve this concern, the inspectors considered this to be an example of an inadequate evaluation of an issue.

.2 Selected Issue Followup Inspection: Preservation of Evidence During Troubleshooting of E4150F003 Failure

a. Inspection Scope

The inspectors evaluated the licensee's methods for determining the cause of the failure of HPCI Turbine Steam Supply Outboard Containment Isolation Valve E4150F003 to close on August 12, 2004 as discussed in Section 4OA3.3 of this report. This included the preservation of evidence pertaining to the as-found condition of components associated with the valve.

b. Findings

Introduction

The inspectors identified a finding of very low safety significance (Green) and a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," when licensee personnel could not determine with certainty the cause of the failure of HPCI valve E4150F003 to close because multiple procedure requirements for quarantining degraded components necessary to evaluate the evidence were not implemented. The inspectors discovered the licensee failed to implement these requirements when components associated with the valve were discarded while troubleshooting the problem.

Description

On August 12, 2004, the licensee conducted stroke time testing of HPCI Turbine Steam Supply Outboard Containment Isolation Valve E4150F003 and the valve failed to close on demand. The licensee initiated CARD 04-23647 to identify this issue in their corrective action program as this was the third such failure within a year. Following the failure, the inspectors accompanied the licensee into the reactor building steam tunnel to observe the as-found condition of the electrical wires inside the limit switch compartment of the motor-operated valve. The wires had experienced a significant color change indicative of excessive temperature exposure from nearly black to brown since the valve was last inspected during the refueling outage conducted in April 2003. Further, numerous cracks were identified in the neoprene layer and corrosion on the torque switch metal surfaces had become more pronounced since the previous inspection.

An Emergent Issues Team (EIT) was formed to investigate the cause of the failure. Through interviews with the team members, the inspectors discovered that the team focused solely on the breaker logic and the control room pushbutton used to close the valve as the potential cause of the failure. Consequently, efforts to replace the breaker and the pushbutton were initiated under WR 000Z973751 and WR 000Z042337. Subsequently, after multiple engineering reviews of the condition, the licensee initiated WR 000Z042333 to conduct an inspection of the valve components that included wires, splices, and terminal boards inside the limit switch compartment.

Work Conduct Manual MWC05, “Troubleshooting,” provided a standard systematic approach for performing troubleshooting on equipment problems, including failures. For complex repair activities, the manual provided instructions to use form MWC05002, “Complex Troubleshooting Datasheet.” Specifically, MWC05002 contained a checklist questioning whether equipment quarantining was required. The inspectors reviewed the final WR package and were unable to locate a completed copy of Form MWC05002. Further, Quality Assurance Conduct Manual (QACM) MQA 11, “Condition Assessment Resolution Document,” provided instructions for quarantining equipment or components to aid in a subsequent investigation. While conducting reportability and operability screening of the CARD, operations personnel were required to determine whether Enclosure C, “Quarantine of Area, Equipment and Records Guidelines,” should be implemented. However, none of these forms or the enclosure was used. Instead, the EIT team leader communicated the need to preserve all as-found components while disassembling the motor actuator on the valve.

Through interviews with the EIT members, the inspectors determined that although the need to quarantine the electrical components was communicated to all electrical crews, maintenance crews discarded the wires and the terminal boards as radwaste. The inspectors discovered this while questioning the location of these components needed to conduct a thorough past operability evaluation. In response, the EIT leader initiated efforts to locate these items and found the components in several radwaste bags. To radiologically release the items, the components were decontaminated using a spray foam cleaner. The components were sent to various test facilities and independent analysts for evaluation.

A preliminary evaluation report of the degraded wires was issued on September 16, 2004 which concluded that evidence handling should be improved. These improvements included 1) as-found state of the specimens must be preserved in that greater care must be taken in the removal of the cable leads to limit damage, 2) more photographs should be taken, and 3) precise notes must be taken regarding incidental damage during removal. The analyst also recommended against chemically cleaning the specimen since the chemical oxidizes exposed conductors and could severely damage the surface conditions of insulation and jacket polymers.

Analysis

The inspectors determined that the failure to quarantine the degraded components associated with E4150F003 as required by QACM MQA 11 and MWC05 was a performance deficiency warranting a significance evaluation in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening." This finding was determined to be more than minor because if left uncorrected, it could become a more significant safety concern since the failure to quarantine degraded components could impede the identification of root causes for conditions adverse to quality and prevent the implementation of appropriate corrective actions to prevent their recurrence. The inspectors determined that the Mitigating Systems cornerstone was affected by the finding. The inspectors determined that the failure to quarantine these degraded components also affected the cross-cutting area of human performance since, despite the existence of multiple barriers such as procedure requirements and multiple communications among electrical maintenance crews to retain the items, the licensee lost control of the degraded components. Using IMC 0609, Appendix A, "Significance Determination of Reactor Findings for At-Power Situations," Attachment 1, "SDP Phase 1 Screening Worksheet for IE, MS, and B," because the finding was not a design or qualification deficiency resulting in a loss of function per Generic Letter 91-18; did not represent an actual loss of safety function of a system or the loss of safety function of a train of equipment; and was not potentially risk-significant due to a seismic, fire, flooding, or severe weather initiating event; the finding was considered to be of very low safety significance (Green).

Enforcement

10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions," requires, in part, that for significant conditions adverse to quality licensees must determine the cause and take action to prevent recurrence. QACM MQA 11, "Condition Assessment Resolution Document," provided instructions for quarantining equipment or components to aid in a subsequent root cause investigation. Enclosure C, "Quarantine of Area, Equipment and Records Guidelines," of MQA 11 provided the requirements for preserving areas, equipment and records involved in an event in a post-event state to ensure against loss of information until an appropriate root cause analysis has been performed. Further, for

complex repair activities, MWC05, “Troubleshooting,” provided instructions to use form MWC05002, “Complex Troubleshooting Datasheet.” Form MWC05002 was a checklist questioning whether equipment quarantining was required.

Contrary to the above, licensee personnel could not discount potential wiring degradation as the cause of the safety-related HPCI Turbine Steam Supply Outboard Containment Isolation Valve E4150F003 failure on August 12, 2004 because multiple procedure requirements for quarantining degraded components necessary to evaluate the evidence were not implemented. The components were discarded as radwaste while repairing the wiring inside the motor actuator and later retrieved and tainted with chemical spray foam. As a result, the ability to determine the cause and implement corrective actions to prevent recurrence of the failure was adversely impacted. The licensee entered this issue into their corrective action program as CARD 04-24490. 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 05000341/2007004-03) consistent with Section VI.A of the NRC Enforcement Policy.

As part of the licensee’s immediate corrective actions, the discarded wiring was retrieved for analysis.

.3 Selected Issue Follow up Inspection: RP Issues Regarding RHR Pump A Seal Cooling Line Leak

a. Inspection Scope

The inspectors reviewed CARDs 04-23596 and 04-23597 involving several communication issues between operations and radiation protection personnel while responding to a seal cooling line leak on RHR Pump A as described in Section 4OA3.1 of this report. The review included interviews with operations and radiation personnel and a review of several Radiation Work Permits (RWPs) and procedures. The inspectors considered the licensee’s evaluation and disposition of performance issues.

b. Findings

No findings of significance were identified

4OA3 Event Followup (71153)

.1 Separation of Seal Cooling Line on Residual Heat Removal Pump A During Shutdown Cooling Operation

a. Inspection Scope

On August 10, 2004, while RHR Pump “A” was in shutdown cooling operation, the seal cooling line connected on the discharge pipe of the pump separated from a compression fitting to the seal cooler. This failure caused the loss of about 7 gallons per minute (gpm) of reactor coolant for a short period of time until the pump was secured and the line was isolated. The inspectors reviewed the circumstances surrounding this event.

b. Findings

Introduction

A self-revealed finding of very low safety significance (Green) and an associated Non-Cited Violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," was identified when the licensee failed to maintain appropriate design control while assembling the RHR Pump "A" seal cooling line during initial plant construction. Design control measures were inadequate in that a ferrule installed on a compression fitting was constructed of carbon steel instead of stainless steel, as required. This resulted in the failure of the seal cooling line during shutdown cooling operations.

Description:

On August 9, 2004, RHR Pump "A" was placed in service in accordance with Procedure 23.208, Section 5.2, "Operating RHR System in the Shutdown Cooling Mode," to remove decay heat from the reactor during a forced outage. On August 10, 2004, the pump was shutdown when an operator identified that a 3/4-inch seal cooling line connected on the pump discharge pipe had separated from a compression fitting. The piping directed a small portion of water from the RHR discharge to the cyclone separator and to a seal cooler before returning to the RHR pump suction. Operators secured the pump and isolated the leak, estimated to be about 7 gpm, and started Division 1 RHR Pump "C" to maintain shutdown cooling. At the time of discovery, control rod drive flow was in operation and of sufficient capacity (60 gpm) to make up for any lost inventory due to the leak. The licensee initiated CARD 04-23582 to document the compression fitting failure.

Work Request 000Z042219 was initiated to repair the seal cooling line. Mechanics disassembled the compression fitting and identified a carbon steel ferrule, that being composed of a relatively soft material, could not "bite" into the harder stainless steel 3/4-inch line. Investigation into the work history of the pump revealed that Field Modification Request S-4168, dated June 11, 1982, which was a design change document used during plant construction, revised the tubing material from A-179 carbon steel to ASTM A-269 Type 304 stainless steel. Several revisions to the Field Modification Request occurred; however, only the tubing was replaced. The fittings, including the ferrule, remained carbon steel. The licensee evaluated seal cooling lines for Division 1 RHR Pump "C"; Division 2 RHR Pumps "B" and "D"; and the four core spray pumps, since the core spray pumps had a similar design. A carbon steel fitting was found on RHR Pump "C". Operators shut down this pump and mechanics replaced the fitting with stainless steel.

Analysis

The inspectors determined that failing to maintain proper design control during fabrication of the RHR Pump "A" and "C" seal cooling systems was a performance deficiency warranting a significance evaluation. The inspectors concluded that the finding was more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," because the finding was associated with the Design Control 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 availability of the RHR system was affected. Additionally, since the ferrule was installed during plant construction, the inspectors considered whether this was considered an old design issue. Per Manual Chapter 0305, “Operating Reactor Assessment Program,” Section 06.06, “Treatment of Old Design Issues,” this finding was not considered an old design issue because it was identified through a self-revealing event.

The inspectors completed a significance evaluation of this issue using IMC 0609, “Significance Determination Process,” Appendix G, Attachment 1, “Shutdown Operations Significance Determination Process Phase 1 Operational Checklist for Both PWRs and BWRs.” In Attachment 1 of Appendix G, the inspectors used Checklist 5, “BWR Hot Shutdown: Time to Boil <2 Hours RHR in Operation (RCS Pressure <RHR Cut-in Permissive),” and answered “No” to Sections I.C(1) and II.C since the operability of the RHR system was adversely affected by the event. Based on these answers, the inspectors concluded that the finding increased the likelihood of a loss of decay heat removal, affected the licensee’s ability to add RCS inventory, and degraded the licensee’s ability to establish an alternate core cooling path due to the failure of RHR Pump “A” warranting a SDP Phase 2 evaluation by a Region III Senior Reactor Analyst (SRA). The SRA concluded that the 7 gpm leak from the RHR pump seal cooling line was well within the makeup capability of the operating control rod drive pump. Therefore, the operating RHR pump could continue to run indefinitely without operator intervention because the leak was limited to 7 gpm. Therefore, the finding was determined to be of very low safety significance (Green).

<u>Enforcement</u>:

10 CFR 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures shall be established and implemented for the selection and review for suitability of materials, parts, and equipment that are important to the safety-related functions of structures, systems, and components.

Contrary to the above, on August 10, 2004, while investigating the failure of a compression fitting on RHR Pump “A”, the licensee discovered that Field Modification Request S-4168, dated June 11, 1982 revised the tubing material from A-179 carbon steel to ASTM A-269 Type 304 stainless steel. However, the modification was not adequately implemented and a carbon steel ferrule was installed instead of a stainless steel ferrule, as required. The licensee entered this issue into their corrective action program as CARD 04-23582. 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 05000341/2007004-04), consistent with Section VI.A of the NRC Enforcement Policy.

As part of the licensee’s immediate corrective actions, licensee personnel replaced the carbon steel ferrule with a stainless steel ferrule and conducted an extent of condition review.

.2 Failure of Emergency Diesel 12 Blower During Testing

a. Inspection Scope

On August 6, 2004, the inspectors responded to and reviewed the circumstances surrounding the failure of the blower on EDG 12 during post maintenance testing. The inspection included interviews with operations, engineering and maintenance personnel and a review of associated WRs and testing procedures.

b. Findings

On August 2, 2004, EDG 12 was removed from service to conduct a planned safety system outage and complete an 18-month preventive maintenance activity. On August 6, 2004, the licensee completed the outage and conducted post maintenance testing in accordance with Procedure 24.307.46, "EDG Fast Start Followed by Load Reject." Following the load reject test, licensee personnel who were inside the room heard an unexpected increase in EDG speed. Operators in the EDG switchgear room heard the same noise and manually tripped the EDG. CARD 04-23549 was written. The inspection covers were removed and the aluminum blower lobes were discovered to be in direct contact with the blower housing.

The blower design was such that the outside air flow passed through the turbocharger compressor to the blower. The blower was connected to the upper crankshaft using a flex gear. As the EDG operated, the blower, which was constructed of six aluminum helical lobes, three upper and three lower, rotated to provide scavenging air for engine combustion during low load operation.

Work Requests 000Z042275, 000Z042275, and 000Z042279 were initiated to repair the blower, clean and overhaul the engine. During investigation, the licensee determined aluminum particles from the damaged blower were dispersed throughout the engine making the repair efforts extensive. The licensee recognized that the 7-day outage time permitted by TS 3.8.1, Action A, would have been exceeded before completing these activities. On August 8, 2004, the licensee telephonically requested from NRC enforcement discretion for an extension of the allowed outage time to 14 days. The request was denied and plant was shutdown on August 9, 2004, to comply with the TSs.

Mechanics completed the work requests and the licensee conducted several EDG tests. The EDG satisfied all post maintenance and TS testing and was declared operable. Parts from the damaged blower were sent to the vendor and other offsite testing facilities. A root cause team was formed which used fault tree analysis and change analysis methods to determine the root cause. The team eliminated several potential failure modes. The team determined the blower seals may have failed. The cause of the seal failure had not been determined at the end of the inspection period. The root cause team conducted a maintenance history review and discovered that this blower was obtained from the Crystal River Nuclear Plant and installed in June 2003 to correct the blower lobe clearances being out-of-tolerance.

This is an Unresolved Item (URI 05000341/2004007-05) pending the inspectors' review of the final root cause, corrective actions, and the inspectors' determination whether a performance deficiency contributed to the failure.

.3 High Pressure Coolant Injection Outboard Steam Isolation Valve Failure to Fully Close

a. Inspection Scope

On August 12, 2004, a HPCI valve failed to stroke fully closed as required during testing. The inspectors reviewed CARDs, work orders, procedures, design calculations, and other documents; and interviewed engineering, maintenance, operations, and work control personnel to follow up on the circumstances surrounding this event.

b. Findings

During HPCI pump time response and operability test 24.202.01 on August 12, 2002, HPCI Turbine Outboard Steam Isolation Valve E4150F003 passed the stroke time test in the open direction but failed to fully close. As documented in NRC Inspection Report 05000341/2004004, the inspectors identified a failure that occurred on April 7, 2004, that was attributed to improper installation of the auxiliary contacts for the breaker associated with the valve motor.

Following the August 12, 2004 failure, the inspectors accompanied the licensee into the reactor building steam tunnel to observe the as-found condition of the electrical wires inside the limit switch compartment of the motor-operated valve. During this inspection, the wires had a significant color change of all leads from nearly black (normal) to brown since the valve was last inspected during the refueling outage conducted in April 2003. Further, numerous cracks were identified in the neoprene layer and corrosion on the torque switch metal surfaces had become more pronounced since the inspection conducted during the last refueling outage.

The licensee determined that the most probable cause of the repetitive failures was an unidentified fault in either the breaker cubicle or the open pushbutton which provided the seal-in contact in the closed direction. To address this potential root cause, maintenance personnel replaced the entire breaker cubicle and the open pushbutton. Furthermore, the licensee replaced the power and control cables from the valve to the first pull box in the steam tunnel where the valve was located. Consequently, the licensee concluded that the fault was bounded by the installation of the new components and wiring.

Considering the as-found condition of the internal wires, the inspectors questioned the environmental qualification of the actuator's motor because it's activation energy, 1.02 eV, was between that of the cable's jacket, 0.65 eV, and the insulation, 1.24 eV. Although the inspectors continued to question the environmental qualification of the motor, they determined that further testing, as proposed by the licensee, was required to confirm the as-left motor qualification.

As described in Section 4OA2.2 of this report, the licensee shipped the internal wires to an outside testing facility for analysis. In addition, both the old breaker and pushbutton were shipped to another testing facility for a failure analysis. The results of these analyses will be used to better evaluate the as-found environmental qualification of the motor actuator as well as the cause for the failure of the valve to close on demand. This is an Unresolved Item (URI 05000341/2004007-06) pending the inspectors' review of the final root cause, corrective actions, and determination if a performance deficiency contributed to this event.

.4 Fermi 2 Scram Due to Main Generator Automatic Voltage Regulator (AVR) Failure

a. Inspection Scope

Fermi 2 automatically scrammed on September 3, 2004, when the main generator automatic voltage regulator (AVR) failed. This caused a turbine trip, which initiated a reactor scram. The scram was uncomplicated and all systems operated as expected and designed. The resident inspectors responded to the site and evaluated the event using the guidance in Management Directive 8.3, "NRC Incident Investigation Program."

b. Findings

No findings of significance were identified.

4OA4 Cross-Cutting Aspects of Findings

A finding described in Section 4OA2.2 of this report had, as its primary cause, a Human Performance deficiency, in that station personnel failed to properly quarantine degraded electrical components inside the motor actuator for E4150F003. Quarantining of these items was necessary since the valve had failed to close three times within a year and a probable cause may have included thermally aged wiring inside the actuator. This error destroyed evidence and may have invalidated the root cause and associated corrective actions.

4OA6 Meetings

.1 Exit Meetings

The inspectors presented the inspection results to Mr. O'Connor and other members of licensee management at the conclusion of the inspection on September 29, 2004. The inspectors asked the licensee whether any material examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Interim Exit Meetings

Interim exit meetings were conducted for:

- Occupational radiation safety program for radiation monitoring instrumentation and protective equipment with Mr. K. Hlavaty on September 17, 2004.

- Licensed Operator Requalification 71111.11B with Mr. R. Duke on

October 4, 2004, via telephone.

4OA7 Licensee-Identified Violation

The following violation of very low significance was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Manual, NUREG-1600, for being dispositioned as a Non-Cited Violation.

Cornerstone: Mitigating Systems

10 CFR 50, Appendix B, Criterion III, “Design Control,” requires, in part, that design changes shall be subject to design control measures commensurate with those applied to the original design and be approved by the organization that performed the original design. Contrary to this requirement, on June 14, 2004, contractors covered approximately one fourth of the Mechanical Draft Cooling Tower inlet fan grating with plywood as part of a foreign material exclusion plan for ongoing work on the RHR reservoir complex roof without an evaluation to ensure continued operability of the Ultimate Heat Sink (UHS), a safety-related system. The licensee entered this issue into their corrective action program as CARD 04-23258.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
W. O'Connor, Jr., Vice President Nuclear Generation
D. Cobb, Plant Manager
D. Craine, General Supervisor, Radiological Engineering
R. Duke, Licensed Operator Requalification Training Group Lead
K. Hlavaty, Maintenance Manager and Acting Plant Manager
H. Higgins, Radiation Protection Manager
R. Libra, Director Nuclear Engineering
K. Morris, Emergency Preparedness Supervisor
N. Peterson, Nuclear Licensing Manager
M. Philippon, Operations Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000341/2004007-01	URI	Operability of Torus Impacted by Scaffold
05000341/2004007-02	NCV	Inadequate Test Acceptance Criteria for HPCI and RCIC
05000341/2004007-03	NCV	Failure to Quarantine Degraded Electrical Components
05000341/2004007-04	NCV	Use of Incorrect Fitting in RHR Seal Cooling Line
05000341/2004007-05	URI	EDG 12 Blower Failure
05000341/2004007-06	URI	Environmental Qualification of E4150F003

Closed

05000341/2004007-02	NCV	Inadequate Test Acceptance Criteria for HPCI and RCIC
05000341/2004007-03	NCV	Failure to Quarantine Degraded Electrical Components
05000341/2004007-04	NCV	Use of Incorrect Fitting in RHR Seal Cooling Line

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.

1R05 Fire Protection

Fire Protection Procedure, Revision 11; Inspection of Penetration Fire Stops
CARD 04-22974; Fire Penetration Fallen Out Around Pipe
UFSAR 9A.4.1.3; Basement Corner Rooms, Zone 2, El. 540 ft 0 in. And 562 ft 0 in.
UFSAR Figure 9A-1, Revision 12; Fire Protection Evaluation Plot Plan
UFSAR Figure 9A-2, Revision 12; Fire Protection Evaluation Reactor Building Subbasement Plan
UFSAR Figure 9A-3, Revision 12; Fire Protection Evaluation Reactor and Auxiliary Buildings Basement Plan
UFSAR 9A.4.1.9; Fourth Floor, Zone 8, El. 659 ft 6 in
UFSAR Figure 9A-9; Fire Protection Evaluation Reactor and Auxiliary Buildings Fourth Floor (Elevation 659.5 ft)

1R11 Licensed Operator Requalification

Evaluation Scenario SS-OP-904-0004, Revision 0; Instrument Failure, Uncoupled Rod, Loss of GSW, RPV Flooding
Evaluation Scenario SS-OP-904-0009, Revision 0; Instrument Failure, Jet Pump Failure, and SC/RR EOP

1R12 Maintenance Rule Implementation

TSR-29900, Rev. 0; “Reset RCIC Turbine Speed Control”; Dated June 10, 1998
RCIC surveillance test data from March 8, 2000, through March 4, 2004
RCIC Maintenance Rule Functional Failure Evaluations from January 1, 1994 though July 28, 2004
RCIC Maintenance Rule Scoping Sheet
MMR Appendix E; Maintenance Rule SSC Specific Functions
MMR Appendix F; Maintenance Rule Performance Criteria

1R13 Maintenance Risk Assessment and Emergent Work

Drawing 6I721-2095-18, Revision R; NSSS System Main Steam Line Outboard Isolation Valves
Drawing 6I721-2155-16, Revision H; Reactor Protection System Testability Modification
Drawing 6I721-2155-15, Revision I; Reactor Protection System Testability Modification
Drawing 6I721-2095-14, Revision M; Nuclear Steam Supply Shut-off System Trip, System A
Drawing 6I721-2095-17, Revision U; NSSS System Main Steam Line Inboard Isolation Valves B2103F022A, B, C & D

1R15 Operability Evaluations

CARD 04-23363; “HPCI flow at design pressure may not meet design requirement”; Dated July 27, 2004 (NRC-identified issue)
Job ID 0249040713; “Perform 24.202.01 section 5.1 HPCI pump/flow test and valve stroke at 1025 psig”; Performed on July 13, 2004
Drawing 6M721-5860, Rev. “D”; “Process diagram high pressure coolant injection system”
UFSAR Figure 6.3-1, Rev. 11; “High pressure coolant injection system process diagram”
Design Calculation DC-501, Vol. I, Rev. E; “High pressure coolant injection system hydraulic analysis”; Dated March 27, 1997
Byron Jackson Test No. T-33034-1; Certified HPCI main pump performance curve; Dated June 22, 1972
Byron Jackson Test No. PC-36658; Certified HPCI booster pump performance curve; Dated March 20, 1987
Procedure 46.202.001, Rev. 32; “HPCI turbine governor control system calibration”
CARD 04-23247; “Review of OE 40876 for applicability to Fermi”; Dated July 20, 2004 (NRC-identified issue)
TSR-29900, Rev. 0; “Reset RCIC turbine speed control”; Dated June 10, 1998
RCIC surveillance test data from March 8, 2000, through March 4, 2004
CARD 00-10133; “Drawing error: 6M721-5859 does not show the correct maximum pressure across the pump for the RCIC pump”; Dated April 27, 2000
DBD E51-00, Rev. C; “Reactor Core Isolation Cooling System”
CARD 98-14557; “Untimely identification of engineering actions to support TS change (licensing amendment) NRC-98-0011 NANL-98-0083"; Dated June 24, 1998
Design Calculation DC-0502, Rev. E; “RCIC hydraulic analysis”; Dated July 7, 1998
UFSAR Section 5.5.6; “Reactor Core Isolation Cooling System”
CARD 04-23542; “—5860, Process Diagram HPCI”; Dated August 6, 2004 (NRC-Identified Issue)
CARD 04-23258; “Unanalyzed Condition - MDCT Fan Inlet Grating Covered”; Dated July 21, 2004
EDP-32813, Rev. C
MMA-17, Rev. 4 and 5; “Foreign Material Exclusion (FME)”
Design Specification 3071-517, Rev. E; “RHR Complex Fermi 2"
Design Calculation DC-0182, Vol. I, Rev. E; “RHRSW Mechanical Draft Cooling Towers - Post LOCA Analysis of UHS”
Vendor Manual VMB9-2.0, Rev. G; “RHR Cooling Tower”
CARD 04-22867; Discrepancies Noted During NRC Inspection of Torus Room
CARD 04-24031; Evaluate MMA08 for Process Improvement With Regards to PSE Seismic Variance Review
CARD 04-24282; Scaffolding Touching the Torus
CARD 04-22915; Scaffold Within Three Inches of Torus
Work Control Conduct Manual MWC10, Revision 0; Work Package Preparation
Maintenance Conduct Manual MMA08, Revision 8; Scaffolding
Licensing/Safety Engineering Conduct Manual MLS14, Revision 5; Changes, Tests and Experiments
50.59 Screen Number 01-0203, Revision A; Conduct Manual MMA08 Revision Number 4
UFSAR 3.7.2.15.3; Interconnecting Category I and Other Structures

Detroit Edison Specification 3071-031, Appendix H, Revision 5; Rattlespace Criteria for Reactor/Auxiliary Building, Drywell, Steam Tunnel/Mezzanine and RHR Complex
Design Calculation Number DC-6102, Revision A; Seismic Analysis of Scaffolding
Design Specification 3071-226, Revision J; Purchase and Installation of Concrete Anchors
Drawing 6M721N-2090-4, Revision P Equipment Foundations - RHR Complex Anchor Bolt Schedule
Drawing 6M721N-2090-6, Revision AB; Equipment Foundation Details RHR Complex
Design Calculation DC-3224, Revision C, Volume I; Class 1E Equipment Qualification Review, System: E41
Environmental Qualification Central File EQ0-EF2-018D, Revision 0
CARD 03-16014; Charred Field Wires in MOV
Environmental Qualification Central File EQ1-EF2-044, Revision E
CARD 04-23695; Terminal Block Aging
CARD 03-17189; Iso Mimic Not Indicating Correct
Equivalent Replacement Evaluation 32423, Revision A
Work Request 000Z020631; Need to Replace Grease in MOV Actuator During Cycle 09 - E4150F003

1R19 Post Maintenance Testing

Job ID 0268040301; “Perform partial surveillance [24.206.01] for PMT”; Performed on March 4, 2004.
Job ID 0268040302; “Perform 24.206.01 RCIC system pump operability and valve test @ 1000 psig”; Performed on March 3, 2004.
Vendor Manual VMR4-4.0, Rev. B; “RCIC Pump”
CARD 04-23362; “RCIC flow at design pressure may not meet design requirement”; Dated July 27, 2004 (NRC-identified issue)
UFSAR Figure 5.5-6, Rev. 9; “Reactor core isolation cooling system process diagram”
GENE NEDC-32789P, “Enrico Fermi Energy Center Unit 2 Safety/Relief Valve Setpoint Tolerance Relaxation Analyses”; Dated January 1998
GE-NE-189-12-0292, Rev. 2; “Study for Power Uprate Technical Report - Reactor Core Isolation Cooling System”; Dated March 14, 1995
WR 000Z040116; “Troubleshoot problem with EGM, setup/checkout”
CARD 01-12462; Suspect outboard bearing bad on Div 1 EECW Pump Room Cooler
Work Request 00Z011696; Suspect outboard bearing bad on Div 1 EECW Pump Room Cooler
Maintenance Procedure 35.000.224, Revision 31; Alignment and Tension Adjustment of V-Belt Driven Equipment
Work Request T582030100; Div 1 EECW Pump Room Cooler; Check bolts, replace belts, power wash cooling coils
Reliance Electric Vendor Manual VMS25-9.1, Revision E; Standard Integral Horsepower Induction Motors
Work Request G290030100; Div 1 EECW Pump Room Cooler; Check bolts, inspect belts, power wash coils, lube & clean Div 1 EECW Pump Room cooler
CARD 03-22531; Vibration data indicates bearing fault on the Div 1 EECW Room Cooler motor.
CARD 03-22219; Noise and high vibration on Div 1 EECW Room Cooler
CARD 03-01655; Abnormal Noise/Vibration Div 1 EECW Pump Room Cooler

CARD 03-22666; Replace Div 1 EECW Room Cooler motor with upgraded motor
PIS T41000B034; EECW Pump Room Cooling Unit
Work Request 000Z020632; Replace Grease in MOV Actuator
Engineering Change Request 4921-1, Revision A; Specifying when an LLRT test must be performed and adding electrical maintenance instruction
Surveillance Scheduling and Tracking, Job ID 0980030328; Perform 43.401-300 LLRT Type C General St
Surveillance Scheduling and Tracking, Job ID 0386030420; Perform 43.401.306 LLRT for X-11
Work Request E452000100; Install Blank In Flow Element E41N400 Flanges to LLRT Test E4150F002
Work Request A876010100; Check Grease, Stem Lube, Packing Torque and Stroke Test MOV
Work Request 000Z945331; Retorque Safety-Related Valves as Listed.
Work Request 000Z971097; Install a Second "B" Contact on SB-2 Operator
Work Request 000Z042638; F402 Failed to Indicate Closed During 24.408.04
Work Request 000Z042333; Valve Failed to Stroke Closed Again
Work Request 000Z973751; Replace MCC 2PB-1 Position 9B (E4150F003)
Work Request 000Z042337; E4150F003 Failed to Stroke Closed. Investigate at MCC
Quality Control Report 04-IR-0685; Terms and Raychem in Limit Switch Compartment
Procedure 35.306.006, Revision 32; Motor Operated Valve Setup Verification
CARD 04-23647; Valve Failed to Stroke Closed Again
Drawing 6I721-2225-01, Revision T; Schematic Diagram HPCI System Logic Circuit Part 2
Drawing 6I721-2221-04, Revisions A through AA; Schematic Diagram HPCI System - Steam Supply Line Outboard Isolation Valves E4150F003, E4150F600
Procedure 35.306.016, Revision 8; Motor Control Center Cubicle Replacement
CARD 04-23680; Work Request Sequence Unexpected Occurrence
Specification 3071-128-ET; Use of Raychem Nuclear Grade Heat-Shrink Insulation Materials
Procedure 24.202.05, Revision 38; HPCI System Shutdown Valve Operability Test

1R20 Refueling & Outage Activities

General Operating Procedure 22.000.05, Revision 39; Pressure/Temperature Monitoring During Heatup and Cooldown
System Operating Procedure 23.205, Revision 86; Residual Heat Removal System
Drawing 6M721-5706-1, Revision AA; RHR Div II Functional Operating Sketch
Drawing 6M721-5706-2, Revision W; RHR Div I Functional Operating Sketch
Drawing 6M721-5813-2, Revision K; ISI Classification Boundary Drawing RHR Div I ISI-E11-2
Drawing 6M721-5813-1, Revision K; ISI Classification Boundary Drawing RHR Div II ISI-E11-1
FO 04-01 Engineering Actions Tracking Task 20; NRC Walkdown Concerns

1R22 Surveillance Testing

UFSAR Section 6.3; “Emergency Core Cooling Systems”
WR 0249040713; “Perform 24.202.01 Section 5.1 HPCI Pump/Flow Test & Valve Stroke at 1025 psig”
Procedure 24.202.01, Rev. 80; “HPCI Pump Time Response and Operability Test at 1025 psig”
ASME XI, 1980 edition, subsection IWP; “Inservice Testing of Pumps in Nuclear Power Plants
DER 90-0280; “VEN SIL 351 REV 2, HPCI and RCIC Turbine Control System Calibration”; Dated April 20, 1990
CARD 04-23188; “NRC RI identified issue - lack of bi-directional IST testing on some HPCI valves”; Dated July 15, 2004 (NRC-identified issue)
CARD 04-23296; “IST acceptance criteria for HPCI pump”; Dated July 23, 2004 (NRC-identified issue)
WR E647040100; “Calibrate HPCI pump discharge flow loop”
WR E663040100; “Calibrate HPCI turbine governor speed loop”
WR E648030100; “Calibrate HPCI pump discharge pressure loop”
DBD E41-00, Revision C; “High Pressure Coolant Injection System”
Drawing 6I721-2671-14, Revision P; Primary Containment Temperature Pressure & Level Measurement Instrument Loop - Division II
Drawing 6M721-5741, Revision AH; Primary Containment Monitoring System Functional Operating Sketch
Work Request 000Z010925; RHR SDC Suction Thermal Relief Line Iso Valve
Engineering Support Conduct Manual MES49, Revision 2; Evaluation and Control of Leakage from Class 1, 2, and 3 Piping Systems
Drawing 6M721-3050-1, Revision U; Piping Isometric - Drywell Floor Drain Sump Pump Discharge Piping - Reactor Building
OSRO Meeting Minutes #1027, September 22, 2004
Drawing 6M721-5710-2, Revision AC; Sump Pumps System Functional Operating Sketch
OSRO Presentation on Unidentified Drywell Leakage
CARD 04-06255; Willful Procedure Violation
CARD 04-23790 and CRD 04-23790-01; Inconsistent Guidance in MMA11 with regard to stroke time testing and packing reconsolidation
MOP03, Enclosure F, Revision 15; Periodic Valve Packing Tightness Checks
Work Request 000Z013570 and Work Request 00Z013570, Revision 3; HPCI Steam Supply Otbd ISO Byb Valve, Bad Grease
Surveillance Performance Form, Job 0255030328; Perform 24.202.05 Sec 5.2 and 5.3 HPCI Local Valve Position Indication Verification & Lsft
PIS R1400S002B; 4160 V Metalclad Switchgear Diesel Generator Bus No. 12EB
Surveillance Performance 1252040531; Perform 24.107.03 Sec-5.1 SBFW Valve Operability/Lineup Verification
UFSAR 10.4.8; Standby Feedwater System
UFSAR Figure 10.4-11, Revision 10; Standby Feedwater System P&ID
Vendor Manual Review, VMS23-1, Revision A, Ingersoll-Rand Company; Standby Feedwater Pumps, Motor and Lubricating Oil System

Procedure 24.000.02; Shiftly Daily, and Weekly Required Surveillances, Attachment 1; Eight Hour – Mode 1,2,3 – Control Room, Reactor Coolant System Operational Leakage, Revision 111
Engineering Support Conduct Manual MES 49; Evaluation and Control of Leakage from Class 1, 2, and 3 Piping System, Revision 2
Work Request 000Z010925; E1100-F086 RHR SDC Suction Thermal Relief Line
Schematic Diagram 6I721-2671-14; Primary Containment Pressure & Temperature Pressure & Level Measurement Instrument Loop – Division II
Functional Operating Sketch 6M721-5741; Primary Containment Monitoring System
Onsite Safety Review Organization Meeting Minutes, dated September 22, 2004
Piping Isometric 6M721-3050-1; Drywell Floor Drain Sump Pump Discharge Piping – Reactor Building-

1R23 Temporary Plant Modifications

TM 04-0020, Rev. 0; “Install three temporary ‘auctioneering’ diodes in series with ‘G05' 24-volt power supply connections at the input terminals of the rectifier converter electronic control modules.”

2OS3 Radiation Monitoring Instrumentation and Protective Equipment

Updated Final Safety Analysis Report; Chapters 11 and 12; Revisions 8 - 11, as applicable
Fermi 2 Radiation Protection Instrument Daily and Weekly Check Lists; dated September 7, 2004
Plant Technical Procedure 64.611.504; Area Radiation Monitor System Channels 1-5, 7-14 and 18-48 Calibration/Functional Test; Revision 13
Plant Technical Procedure 66.000.242; Calibration of NNC Gamma 60 Portal Monitor; Revision 1
Plant Technical Procedure 66.000.247; Calibration of IPM9D Monitor; Revision 0
Plant Technical Procedure 66.000.304; Verification of Gamma Calibrator Dose Rates; Revision 5
Plant Technical Procedure 66.000.245; Calibration of the NE Small Articles Monitor; Revision 1
System Health Reports; Process Radiation Monitoring; 4th Quarter 2003 - 2nd Quarter 2004
NNC Calibration Data Forms; Instrument Nos. 960060 and 960059; dated June 16 and July 7, 2004, respectively
Dositec AR-20 Calibration Form; Instrument No. 30255; dated April 27, 2004
AMP-100 Calibration Forms; Instrument Nos. 5000-152 and 5001-147; dated April 27 and September 7, 2004, respectively
IPM9D Calibration Forms; Instrument Nos. 296, 302, 298, and 300; dated March 19, 2004; March 12, 2004; May 17, 2004; and May 14, 2004; respectively
Plant Technical Procedure 64.120.040; Containment Area High Range Radiation Monitor Division 1 and Division 2 Calibration Data; both dated March 17, 2003
Plant Technical Procedure 64.611.504; Area Radiation Monitoring System Channels 1-5, 7-14 and 18-48 Calibration/Functional Test Data Sheets, Channels 12, 14, 18 and 45; dated April 10, 2003; August 19, 2004; August 19, 2004; and May 19, 2004; respectively

Radcal Corporation Calibration Report; Model 2025AC Radiation Monitor (No. 4007) with Model 20X5-3 (No. 21135), Model 20X5-180 (No. 7498), and Model 20X5-1800 (No. 9959) Ion Chambers; dated June 10, 2004
SAM-11 Calibration Forms; Instrument Nos. 281 and 312; dated March 2, 2004 and January 6, 2004, respectively
Whole Body Counter Calibration Records; Standup Counter and Chair Counter; dated July 23, 2004 and September 7, 2004, respectively
Scaling Factor Report and Associated Analysis Results; dated August 26, 2003
Post Accident Sampling System Surveillance Records; CHS-AUX-09 and Associated Data and Analysis Results; dated January 29, 2004 and July 29, 2004
Chemistry Technician Qualification Matrix for Post Accident Sampling System; dated April 15, 2004
Nuclear Training Lesson Plan No. LP-RC-853-0001; Post Accident Sampling; Revision 4
CARD 03-21185 and Associated Root Cause Investigation; Inability to Complete Surveillance by Required Time; dated August 15, 2003
CARD 03-16467; Generate Work Package to Repack P3400F015 (PASS System); July 2, 2003
CARD 004-23810; Target Rock Valve Does Not Open; August 23, 2004
Radiological Emergency Team and Damage Control and Rescue Team Rosters and Team Qualification Matrix; September 2004
Fermi 2 Respiratory Protection Qualification Matrix; September 2004
Selection, Training and Qualification Program Description; QP-ER-665; Emergency Response Organization; Revision 26
Self-Contained Breathing Apparatus Maintenance and Inspection Logs; January 2003 - August 2004
Self-Contained Breathing Apparatus Inventory; September 8, 2004
Plant Technical Procedure; 65.000.707; Inspection of MSA Respiratory Equipment; Revision 10
Lesson Plan No. LP-GN-509-0100; Respiratory Protection - Airborne Area Work Controls and Devices; Revision 3
Lesson Plan No. LP-GN-509-0200; Respiratory Protection - Self Contained Breathing Apparatus; Revision 2
Lesson Plan No. LP-GN-509-0300; Respiratory Protection - Self Contained Breathing Apparatus and Emergency Breathing Air; Revision 3
Mine Safety Appliance Certificates for Six Members of the Radiation Protection Staff; July 2001 and July 2004
CARD Database Listings Related to Radiological Instrumentation, SCBAs, and the Radiation Monitoring System; January 2003 - August 2004
CARD 03-11147; Notification of Fire Brigade Qualifications; February 2, 2003
CARD 04-22320; Reactor Building Vent Exhaust Radiation Monitor Calibration Failure; May 25, 2004

4OA1 Performance Indicator Verification

NEI 99-02; Performance Indicators; dated November 19, 2001
Gaseous Effluent Summary Data for Selected Periods in 2004 Including Chemistry Analysis Data and Drywell Purge Calculations
Performance Indicator Effluent Dose Summary Data for 3rd Quarter 2003 - 2nd Quarter 2004

4OA2 Identification and Resolution of Problems

Equivalent Replacement Evaluation 31955, Revision 0; Replacement of Nebula EP with MOV Long Life grease
September 16, 2004, EPRI Plant Support Engineering Report; Field Cable and Lead Evaluation for MOV E1450F003
CARD 04-23534; EDG #12 Output Breaker EB3 could not be opened locally or from MCR
CARD 02-16956; E1150-F007A Failed to Open as Expected During Performance of 24.204-01
Work Request 000Z000583; Perform Set Point Change on UV Relays per TSR 31005
Work Request 000Z943469; Metalclad Switchgear Diesel Generator Bus No. 12EB, Refurbish Breaker 12EB - EB3
Work Request 000Z943473; Metalclad Switchgear Diesel Generator Bus No. 12 EB, Refurbish Breaker 12EB - EB2
Surveillance Procedure 42.302.01, Revision 40; Channel Functional Test of Division 1 4160 Volt Bus 64B Undervoltage Circuits
Drawing 6I721-2572-28Q; 4160V Ess Busses #64B and 64C Load Shedding Strings
CARD 04-23596; RP/Operations work control associated with RHR water leak
CARD 04-23597; RWP 04-1001 Violation
Fact Finding Notes for ED Dose Rate Alarm
Radiation Protection Conduct Manual MRP04, Revision 13; Accessing and Working in the Radiologically Restricted Area
Fermi 2 Safety Handbook, Section 1, Revision 7; General Rules and Practices
Fermi 2 Safety Handbook, Section 6, Revision 7; Fall Protection Program
Summary of Events / Time Line; “A” RHR Pump Leak
Fermi 2 Daily Plant Status, Tuesday, August 31, 2004
RP Shift Log; 08/10/04, Shift 0700-1900
Radiological Survey 02649-R04; Start-up/Shut-down Cooling Div 1
Radiological Survey 02661-R04; Follow-up Leak Seal Line Rupture to “A” Pump
Radiological Survey 02523-R04; —5
Radiological Survey 02173-R04; Monthly #5
Radiological Survey 02052-R04; Survey
Radiation Work Permit 04-1001, Revision 0;

4OA3 Event Followup

Reactor Engineering Standing Order #10-02, Revision 1; Thermal Limits
TMSA-03-0074, RPV Coolant Heatup Rate Without SDC in Mode 4
Fermi 2 Operator Log, 8/8-17/2004 and 09/02-03/2004
Surveillance Procedure 43.204.001, Revision 28; RHR Division 1 Leakage Monitoring Test
Forced Outage 04-01 Daily Status Report
General Electric Field Disposition Instruction 107-33800, Revision 0; RHR and Core Spray Pumps
Group Trend E11DT2497; RHR HX Inlet Temperature Div 1
Group Trend G33DT2810; RPV Bottom Drain Temperature
Group Trend C11DF1052; Control Rod Drive Flow
Group Trend N20CB5151; HFP Center Pump Status

Group Trend N20CB5152; HFP East Pump Status
Group Trend N20CB5153; HFP West Pump Status
Group Trend B21CP6601; RPV Pressure Average
Borg-Warner Specifications, June 70; Borg-Warner High Pressure Heat Exchangers Model NX-0625-H
ARP 2D105, Revision 12; Reactor Building Corner Rooms HPCI Room Flood Level
CARD 04-23582; RHR pump “A” seal line has double ended shear
Design Change Notice S-4168, Revision C; Vendor Drawing 2C-4970, E11-00 RHR Pump Pis #3-02-C002 A,B,C, and D
GE Letter Report GENE-0000-0022-8680-00, DRF 0000-0007-1008, Revision 0, dated November 2003; RHR and Core Spray Pump Mechanical Seal Operations at High Temperature
CARD 03-12097; EECW Flow to E1102C002B(D) Seal Coolers Measured Under SOE 03-01 Significantly Reduced from Flows Observed in SOE 95-11
Drawing 6M721-5706-2, Revision W; Residual Heat Removal Div 1 FOS
PIS E1102C002A; RHR Pump A, Replace Fitting on seal water line from pump outlet to cyclone separator.
NRC 0609, Appendix G; Shutdown Operations, Significance Determination Process
NRC 0609, Appendix G, Attachment 1; Shutdown Operations, Significance Determination Process, Phase 1 Operational Checklists for Both PWRs and BWRs
DTE Energy Report September 16, 2004, EDG 12 Blower Failure
CARD 04-23549; EDG 12 Fast Start Load Reject Issues
Crystal River Blower Storage Certification dated June 4, 2003
Preliminary Notification of Event or Unusual Occurrence; PNO-III-04-009; Fermi Shuts Down to Repair EDG
TMSA-04-0063 dtd August 7, 2004; Risk Associated with NOED Request for EDG 12 AOT Extension from 7 to 14 Days
Surveillance Procedure 24.307.46, Revision 8; EDG 12-Fast Start Followed by Load Reject
Detroit Edison’s ltr dtd 08/08/2004 to NRC, NRC-04-0061; Request for Enforcement Discretion with Respect to the TS Limiting Condition for Operation Related to EDG -12
8-Point Trend; TEMGRP01 dtd 08/06/2004
Point Trend; TEMGRP01 dtd 08/19/2003
CARD 04-23450; Different Gasket Materials for the Same Stock Code
Surveillance Performance Job 12198040806; Perform 42.302.11 4160 V Bus 64C Div 1, Undervoltage Circuits, C/Func.
CARD 04-23644; HP/Wrong Equipment
CARD 04-24119; AVR General Alarm (4D53) Due to Field Current DC Failure
Drawing 6SD721-2530-12, Revision AP; One Line Diagram 260/130V BOP Battery 2PC Distribution
Operations Training ST-OP-315-0055-001, Revision 12; Main Generator and Excitation
Drawing 6I721-2349-17, Revision H; Wiring Diagram Automatic Voltage Regulator Cabinet #H11P630
Drawing 6I721-2731-05, Revision I; Schematic Diagram - Generator #2 Differential Relaying & Control Unit #2
Drawing 6I721-2349-13, Revision B; Schematic Diagram - Control Input and Output Auto Voltage Regulator Static Excitation System
ARP 4D65, Revision 11; Generator Protective Relaying Operated
CARD 04-24040, Reactor Scram on AVR Relay Trip

Control Room Status Checklist, 09/03/2004
Post-Scram Data and Evaluation 2345; 09/03/2004
0645 Shift Manager’s Meeting Agenda; 09/10/04
Forced Outage 04-02 Daily Status Reports; 09/05/2004 and 09/06/2004
Drawing 6I721-2349-12, Revision 0; Schematic Diagram Power Supp & Meas Circuits Auto Voltage Regulator Static Excitation Sys.
CARD 04-24023, AVR General Alarm and Trip of AVR Channel A
Fermi 2 Operator Log, selective dates from 07/04/1996 to 09/03/2004
CARD 02-13671; Control Room Received 4D53 - AVR General Alarm due to Channel 47 - Field Current DC Failure
Post-Scram Data and Evaluation 0228; 09/04/2004
Fermi 2 Sequential Events Recorder Log 09/03/2004 - 09/04/2004
ARP 4D53, Revision 9; AVR General Alarm

LIST OF ACRONYMS USED

ARM	Area Radiation Monitor
CARD	Condition Assessment Resolution Document
CAM	Continuous Air Monitor
CEDE	Committed Effective Dose Equivalent
CFR	Code of Federal Regulations
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
EECW	Emergency Equipment Cooling Water
EDG	Emergency Diesel Generator
EIT	Emergent Issues Team
HPCI	High Pressure Coolant Injection
IMC	Inspection Manual Chapter
IST	Inservice Testing
JPM	Job Performance Measure
NCV	Non Cited Violation
NRC	Nuclear Regulatory Commission
OSC	Operational Support Center
PASS	Post Accident Sampling System
PI	Performance Indicator
QACM	Quality Assurance Conduct Manual
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RP	Radiation Protection
rpm	revolutions per minute
RRA	Radiologically Restricted Area
SAM	Small Article Monitor
SCBA	Self-Contained Breathing Apparatus
SDP	Significance Determination Process
TIP	Traversing Incore Probe
TM	Temporary Modification
TS	Technical Specification
UFSAR	Updated Final Safety Assessment Report
WR	Work Request