July 24, 2003

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

SUBJECT: FERMI, UNIT 2
NRC INTEGRATED INSPECTION REPORT 50-341/03-006

Dear Mr. O'Connor:

On June 30, 2003, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Enrico Fermi, Unit 2. The enclosed report documents inspection findings which were discussed on July 11, 2003, with you, Mr. Cobb, and other members of your staff. The findings were also discussed with S. Stasek and others on July 22, 2003.

The inspection examined activities conducted under your license as they relate to safety and 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.

This report documents three self-revealing findings and one inspector identified finding of very low safety significance (Green). Three of these issues were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating these issues as Non-Cited Violations (NCV) consistent with Section VI.A of the NRC Enforcement Policy.

If you contest any NCV in this report, 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 copies to the Regional Administrator, Region III; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington DC 20555-0001; and to the Resident Inspector at the Fermi 2 Nuclear Power Plant.

Since the terrorist attacks on September 11, 2001, NRC has issued five Orders and several threat advisories to licensees of commercial power reactors to strengthen licensee capabilities, improve security force readiness, and enhance controls over access authorization. In addition to applicable baseline inspections, the NRC issued Temporary Instruction 2515/148, "Inspection of Nuclear Reactor Safeguards Interim Compensatory Measures," and its subsequent revision, to audit and inspect licensee implementation of the interim compensatory measures required by order. Phase 1 of TI 2515/148 was completed at all commercial nuclear power plants during calender year 2002 and the remaining inspection activities for Fermi were completed on February 28, 2003. The NRC will continue to monitor overall safeguards and security controls at Fermi.

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

Sincerely,

/ RA /

Mark A. Ring, Chief
Branch 1
Division of Reactor Projects

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

Enclosure: Inspection Report 50-341/03-06

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

REGION III

Docket No:	50-341
License No:	DPR-43
Report No:	50-341/03-06
Licensee:	Detroit Edison Company
Facility:	Enrico Fermi, Unit 2
Location:	6400 N. Dixie Hwy. Newport, MI 48166
Dates:	April 1 through June 30, 2003
Inspectors:	S. Campbell, Senior Resident Inspector T. Steadham, Resident Inspector P. Pelke, Reactor Engineer, Branch 1 R. Alexander, Radiation Specialist M. Holmberg, Reactor Inspector J. Ellegood, Resident Inspector, Perry R. Powell, Senior Resident Inspector Perry G. Pirtle, Physical Security Inspector P. Harris, Operations Engineer, NRR
Approved by:	Mark Ring, Chief Branch 1 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000341-03-006, Detroit Edison Company, on 4/01/03-6/30/03, Fermi 2 Nuclear Power Station. Maintenance Risk Assessments and Emergent Work Evaluation, Post Maintenance Testing, Surveillance Testing, Radioactive Material Control Program, Event Followup.

This report covers a 3-month period of baseline resident inspection and announced baseline inspections on April 11 and May 22, 2003. The inspection was conducted by regional inspectors and the resident inspectors. Four Green findings were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual 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. Inspection Findings

Cornerstone: Initiating Events

- Green. A self revealed finding was identified for inappropriate maintenance on the breaker for the west station air compressor on December 20, 1999. This led to excessive cycling of the breaker and failure of the compressor coupling during compressor shutdown on March 13, 2003. This issue was considered more than minor because it affected an attribute and objective of the Initiating Events Cornerstone. No violation of regulatory requirements occurred because the issue occurred on plant support equipment, which was nonsafety-related.

  A phase 2 risk assessment was performed and it was determined that the issue had low safety significance (Green) due to a low initiating event frequency of a Loss of Instrument Air with one of three station air compressors unavailable. (Section 1R19)

- Green. The inspectors identified a Non-Cited Violation for failing to comply with 10 CFR 50.71(e). This violation was related to not updating the Updated Safety Analysis Report (USAR) with the effects of a safety analysis and evaluation performed by the licensee in support of approved license amendment 114, which allowed average reactor coolant system temperature to be between 200 and 212 degrees Fahrenheit for conducting in-service leakage tests and hydrostatic tests while in Mode 4. Specifically the licensee's safety analysis and evaluation credited all control rods as fully inserted to ensure additional shutdown margin during these testing evolutions, but this condition (all rods inserted) was not incorporated into the USAR or licensee procedures.

  This finding was more than minor because this condition permitted the conduct of control rod scram testing during hydrostatic and system leakage testing, which was different from the initial condition that all control rods were fully inserted as assumed in the analysis and evaluation for approved license amendment 114. The licensee provided this initial condition in their amendment request and the NRC staff used this initial condition in the safety evaluation for approval of license amendment 114. The failure to comply with 10 CFR 50.71(e) caused inappropriate pressure testing procedure changes and created a situation where less shutdown margin than originally evaluated and accepted by the NRC staff was present at times when these activities were conducted concurrently. As such, the performance of scram time testing during hydrostatic or system leakage testing would have required prior NRC approval in accordance with 10 CFR 50.59. The issue was considered of very low safety significance because the finding did not contribute to the likelihood of a primary or secondary system loss of coolant accident, it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available nor did it increase the likelihood of a fire or internal/external flood. Adequate shutdown margin had always been maintained. (Section R22)

- Green. A self-revealed violation of 10 CFR 50, Appendix B, Criterion V, occurred on April 28, 2003, when operators improperly shifted reactor protection system busses and caused a loss of shutdown cooling. 10 CFR 50, Appendix B, requires, in part, that activities affecting quality shall be accomplished in accordance with prescribed procedures. The licensee developed Procedure 23.316 to shift reactor protection system busses and require depression of the open pushbutton for E1150-F008. Contrary to these requirements, the operator failed to depress the pushbutton and caused a loss of shutdown cooling.

  The inspectors determined that the violation is more than minor using the guidance provided in NRC Inspection Manual Chapters 0609 and 0612. Chapter 0609 provides guidance on evaluation of the significance of findings for a shutdown reactor. In accordance with Appendix G, Table 1 of 0609, loss of shutdown cooling is a finding of very low safety significance (Green). (Section 4AO3)

Cornerstone: Public Radiation Safety

- Green. A self-revealing violation of 10 CFR 20.1802 was identified, when the licensee failed to maintain control of a measurable amount of licensed radioactive material (i.e., external radioactive contamination on a lanyard) identified during whole body counting of a contractor.

  The finding was more than minor because it was associated with the “Program and Process” and “Human Performance” attributes of the Public Radiation Safety Cornerstone and affected the cornerstone objective in ensuring adequate protection of public health and safety from exposure to radioactive materials released into the public domain. The whole body count vendor operator’s apparent lack of knowledge as to the requirements to control measurable amounts of radioactive material, which was exacerbated by less than adequate radiation protection oversight of the vendor and procedure deficiencies, led to the unrestricted release of measurable radioactive material. However, this finding, associated with the licensee’s radioactive material control program, was of very low safety significance in that public radiation exposure was not greater than 0.005 rem and the licensee did not have more than five radioactive material control occurrences (in the previous 8 quarters). An associated Non-Cited Violation of 10 CFR 20.1802 was identified for the failure to control licensed radioactive material in an unrestricted area and not in storage. (Section 2PS3)

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

Violations of very low safety significance, which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. These violations and corrective action tracking numbers are listed in Section 40A7 of this report.

REPORT DETAILS

Summary of Plant Status

At the beginning of the inspection, the plant was shutdown during a planned refueling outage (RF09). At the conclusion of the outage, operators commenced an orderly plant startup. The reactor was critical on May 7, 2003 at 12:28 p.m. Startup continued and the plant entered Mode 1 on May 9, 2003 at 4:17 p.m. On May 14, 2003, the plant reached 100 percent power where it remained at or near for the duration of the inspection except for two pre-planned down powers to approximately 75 percent for rod pattern adjustments on May 19, 2003 and May 30, 2003.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignments (71111.04Q)

.1 Partial Walkdowns

a. Inspection Scope

The inspectors performed partial walkdowns of accessible portions of risk-significant, mitigating systems during times when the systems were of increased importance due to redundant divisions or other related equipment being unavailable. The walkdowns were performed to verify proper alignment of valves, control switches and clear Control Room annunciator alarms. The inspectors reviewed associated piping and instrumentation drawings, condition assessment resolution documents (CARDs) and used the system operating procedures lineup to verify the system standby alignment. The inspectors used the documents to determine standby readiness of the system.

- Control Rod Drive (C1100)
- Non-Interruptable Air (P5002)
- Station Air (P5001)

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments (71111.04S)

Semiannual Inspection

a. Inspection Scope

The inspectors performed a complete walkdown of the Division 1 residual heat removal (RHR) E1100 system. This system was selected because of its risk-significance in the licensee's probabilistic risk assessment. The inspection reviewed the following:

- Appropriate plant procedures
- Drawings
- Updated Final Safety Analysis Report (UFSAR) to identify proper system alignment
- System training manual
- Maintenance work requests
- Configuration Control Condition Reports
- Outstanding design issues and operator work arounds
- Control room logs

The inspectors used the documents to verify valves were positioned correctly and did not exhibit leakage that would impact the valve's function, availability of electrical power, proper labeling, lubrication and cooling of major equipment, and functionality of hangers and support systems.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q)

a. Inspection Scope (71111.05Q)

The inspectors toured the following areas to determine whether combustible hazards were present, fire extinguishers were properly filled and tested, the CARDOX units were operable, hose stations were properly maintained, and if the fire hazard analysis drawings were correct:

- Cable Tray Area, Auxiliary Building Elevation 583.5 ft., Zone 2 (UFSAR Section 9A.4.2.3)
- Division I Switchgear Room, Zone 4 (UFSAR Section 9A.4.2.5)
- Auxiliary Building Basement Elevation 551 ft and 562 ft., Zone 1 (UFSAR Section 9A.4.2.2)
- Cable Spreading Room, Elevation 630 ft., 6 in., Zone 7 (UFSAR Section 9A.4.2.8)
- Control Room, Elevation 643 ft, 6 in., Zone 9 (UFSAR 9A.4.2.10)
- EDG 11 Room (UFSAR Section 9A.4.3.1)
- EDG 12 Room (UFSAR Section 9A.4.3.1)

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors observed the RHR Division 2 heat exchanger performance test. The inspectors verified if deficiencies masked degraded performance. Also, the inspectors verified whether any potential common cause heat sink performance problems may have increased plant risk. Finally, the inspectors verified that the licensee had adequately identified and resolved heat sink performance problems (corrosion, fouling, silting) that could result in affecting multiple heat exchangers in mitigating systems and thereby increase risk.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

a. Inspection Scope

The inspectors evaluated the implementation of the licensee's inservice inspection program for monitoring degradation of the reactor coolant system boundary and risk-significant piping system boundaries, based on review of records and in-process observation of nondestructive examinations.

From April 6, 2003 through April 8, 2003, the inspectors observed:

- Automated ultrasonic examination of a reactor vessel nozzle-to-vessel weld 4-316C conducted from the General Electric remote trailer located just outside the reactor building
- Manual ultrasonic examination of a control rod drive cap-to-nozzle weld N-9 near the reactor vessel at the 638 foot elevation in containment
- Automated ultrasonic examination data review and analysis for the reactor vessel longitudinal weld 15-308B and recirculation system riser nozzle-to-safe end weld 2-303G, observation conducted on the second floor of the Availability Improvement Building

From April 7, 2003 through April 11, 2003, the inspectors reviewed:

- Repair and replacement records for a weld repair made on a cracked stiffener plate for the Division 2 RHR heat exchanger and two Code replacement activities (outlet relief valve on the RHR heat exchanger and the control valve for the hydrogen recombiner)
- Disposition of Code recordable indications identified during surface examinations of welds in the RHR and recirculation system piping

The inspectors reviewed records and activities observed for conformance with requirements in the American Society of Mechanical Engineers Code, Sections III, V, IX, and XI.

The inspectors' review of records occurred on the second floor of the Availability Improvement Building located within the site Protected Area.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

On June 18, 2003, the inspectors observed Shift 1 operating crew during the annual requalification examination in mitigating the consequences of events in Scenario 5, “Reactor Power Oscillation/High Pressure Cooling Injection Steam Leak,” 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
- Ability to identify and implement appropriate Technical Specification actions and Emergency Plan actions and notifications.

The crew’s performance in these areas was compared to pre-established operator actions expectations and successful critical tasks completions requirements.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12Q)

a. Inspection Scope

The inspectors reviewed the system health reports, associated CARDs, white papers for probabilistic risk assessments on conditional probabilities, and the control room unit logs for the following systems to evaluate the maintenance rule program characterization of failed structures, systems, and components in the maintenance rule program. The inspectors also evaluated the performance goals and performance monitoring.

- Station Air (P5001)
- Non-Interruptable Air (P5002)

b. Findings

No findings of significance were identified.

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

.1 Routine Control of Work and Plant Risk Evaluation

a. Inspection Scope

The inspectors reviewed the documents listed in the “List of Documents Reviewed” section of this report to determine if the risk associated with the activities listed below agreed with the results provided by the licensee’s risk assessment tool. In each case, the inspectors conducted walkdowns to ensure that redundant mitigating systems and/or barrier integrity equipment credited by the licensee’s risk assessment remained available. When compensatory actions were required, the inspectors conducted plant inspections to validate that the compensatory actions were appropriately implemented. The inspectors also discussed emergent work activities with the shift manager and work week manager to ensure that these additional activities did not change the risk assessment results.

Maintenance Activity Assessed	**Week Inspected**
Corroded fastener and welds on the service pump suction columns	4/6/03
Shutdown safety: postings, defense-in-depth, and procedure	4/27/03
Emergency diesel generator (EDG) 12 outage and Restricted/Controlled access to Div 2 EDGs	6/1/03
EDG 12 blower clearance found out of specifications during safety system outage	6/8/03
Control room emergency filtration differential pressure transmitter NO22A failure	6/8/03
Insulation removed on Div. 1 & 2 RHR piping	6/22/03
Issues with electrical protection assembly (EPA) reactor protection system (RPS) breakers (undervoltage light on but breaker not tripped & breaker not latching after reset)	6/22/03

b. Findings

No finding of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors assessed the following operability evaluations or issues associated with equipment operability:

- Corroded bolts on the Service Water System (CARD 03-14451)
- Water in the Torus Downcomer Vent Tee (CARD 03-14450)

b. Findings

No finding of significance were identified.

1R16 Operator Work-Arounds (71111.16)

a. Inspection Scope

The inspectors performed a review of all operator workarounds and challenges identified as of May 5, 2003, prior to plant restart following the refueling outage. The inspectors reviewed the following:

- Risk Assessment of Revised Operator Workarounds - May 2003
- Division 2 RHR Reservoir Make-Up Valve Will Not Close Automatically
- Stator Water Cooling System - Compensatory Action Is Required Whenever the Operating Pump Is Shifted

The inspectors compared workaround information to the normal, abnormal, and emergency operating procedures to ensure that operations personnel maintained the ability to correctly respond to plant transients in a timely manner. The inspectors utilized system knowledge, reviewed plant procedures, and interviewed operations personnel to ensure that the workarounds and challenges previously identified did not adversely impact system reliability and availability, create the potential for system misoperation, or result in a workaround that impacted multiple mitigating equipment.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

Engineering Design Package 32499, “Install Spring Wedge Assembly for Set Screw on Jet Pump No. 15,” was reviewed for technical adequacy. The purpose of the modification was to provide a permanent repair for the cracked tack welds on the shroud side restrainer bracket adjusting set screw of jet pump B114C015 (jet pump No. 15). The modification involved the installation of an auxiliary spring wedge assembly between the jet pump restrainer bracket and the jet pump inlet-mixer assembly.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

.1 Routine Testing After Maintenance

a. Inspection Scope

The inspectors reviewed and observed the following post-maintenance testing activities involving risk significant equipment in the Mitigating Systems cornerstones:

- Division 2 Core Spray System testing following Work Request 000Z013739, "Repack Division 2 Core Spray Pump Minimum Flow Valve"
- Division 1 Control Air Center Auto Start Test following its safety system outage
- Drywell Sump Outboard Isolation Valve (CARD 03-11287)
- Leak on West Air Receiver Drain Valve for EDG 13
- Various Post Maintenance Tests performed during the Reactor Pressure Vessel Hydrostatic Test

The inspectors verified that the post-maintenance test was adequate for the scope of the maintenance work performed, acceptance criteria were clear, and operational readiness consistent with design and licensing basis documents was demonstrated. The inspectors also verified that the impact of the testing had been properly characterized in the risk assessment, the test was performed as written, the testing prerequisites were satisfied, and that the test data was complete. Following the completion of the test, the inspectors verified that the system was returned to its normal standby configuration.

b. Findings

No findings of significance were identified.

.2 Post Maintenance Testing of the West Station Air Compressor

a. Inspection Scope

The inspectors reviewed control room logs, work requests, and CARDs, and interviewed system engineering and operations personnel to understand the circumstances surrounding the failure of the west station air compressor that occurred during post maintenance testing.

b. Findings

Introduction: A Green finding was identified for conducting inappropriate maintenance on the breaker for the west station air compressor that led to excessive cycling of the breaker and failure of the compressor coupling during compressor shutdown, an issue considered greater than minor because it affected the Initiating Events Cornerstone.

Description: Breaker 72A-3D refurbishment on the west station air compressor was performed on December 20, 1999, under Work Request V298960311. During that activity, maintenance personnel installed an incorrect breaker primary closing spring

(Asea Brown Boveri Part Number No. 50402B vice Part No. 650216A11). The correct spring was stronger and had fewer coils. The incorrect spring size would prevent the breaker from remaining closed to provide power to the compressor motor. Instead, the breaker would cycle open and close, thereby interrupting power to the motor.

On October 22, 2002, during shutdown of west station air compressor P5001D003, the breaker cycled on and off several times, the coordinated manual control switch light flickered and the ammeter oscillated between 0 and 200 amps. Condition Assessment Resolution Document 02-16323 was initiated for tracking purposes only, meaning that no work requests were generated to repair the problem. Again on November 9, 2002, the breaker began cycling and CARD 02-16378 was written to address this condition. Following this event, the licensee decided to place the west station air compressor on “non-preferred use,” meaning that operations personnel would select either the center or the east station air compressor to operate the station air system. Planners initiated Work Request 0000Z023462 to correct the condition. Troubleshooting proceeded under that work request and, on February 26, 2003, breaker cycling occurred again. Attempts to resolve the problem were unsuccessful because on March 13, 2003, during a post maintenance test run, the west station air compressor motor began sparking, smoking and glowing red on the portion of the shaft that enters the motor housing. The compressor was immediately shut down and investigation into the failure was started under CARD 03-00203.

A visual inspection was conducted on the coupling where the licensee discovered a crack. Upon disassembly, the licensee noticed severe torque induced damage on the compressor shaft. The licensee contacted and discussed the results of their investigation with the vendor, Centac. The licensee determined that the most probable cause of the coupling failure and shaft damage was the breaker cycling.

An induction motor is used to operate the compressor and stored internal voltage is characteristic of this type of motor. If the motor is restarted before the internal voltage has dissipated, the resultant large voltage can cause a corresponding coupling torque 20 times the rated design value. Centac Operation and Maintenance Notes 11-96 provided examples of sheared couplings, shafts or gearing as worst case examples while twisted pinions and detached impellers occurred during momentary power loss, such as during a breaker cycling. The licensee determined that the incorrect spring size, installed in December of 1999, was the cause of the breaker cycling condition.

The correct spring was installed and the compressor was tested satisfactorily. Corrective actions to the CARD included addressing this issue in electrical maintenance training and inspecting breakers removed in RF09 for correct closing spring sizes.

<u>Analysis</u>: The inspectors determined that the finding was a performance deficiency because maintenance personnel installed the incorrect primary closing spring during breaker 72A-3D refurbishment for the west station air compressor. The finding was greater than minor because it affected an attribute and objective of the Initiating Events Cornerstone in that the error resulted in the ultimate failure of the west station air compressor. Updated Final Safety Analysis Report Section 9.3.1.3, ”Safety Evaluation,” states that maximum plant availability and control air system reliability are ensured by

providing three station air compressors. In addition, if left uncorrected, this finding could result in a more significant safety concern (i.e., a plant transient).

During SDP Phase 1 screening of the issue, the inspectors determined that one less compressor incrementally increased the likelihood of an initiating event and that the failed compressor contributed to the likelihood of unavailable mitigating equipment. Using the current risk-informed inspection notebook for Fermi 2 (Revision 0) for the Phase 2 SDP analysis, the inspectors determined that this finding was potentially greater than very low safety significance. Specifically, the inspectors determined that this issue caused the likelihood for transients involving a loss of the compressed air system to be increased by an order of magnitude using Usage Rule 1.3 of IMC 0609, Appendix A, Attachment 2.

The initiating event likelihood was evaluated for a greater than 30-day period because the condition had existed since December 1999. However, after a review of additional information, the inspectors determined that a Phase 3 analysis was required. As a result, the inspectors requested assistance in determining risk from the NRC Regional Senior Reactor Analyst (SRA). The SRA informed the inspectors of a refinement to the Fermi 2 SDP notebooks based on a recent bench marking trip the SRA conducted at Fermi in April 2003. During that trip, the SRA recognized that the loss of one of three compressors for an indefinite period represented a low initiating event frequency of a Loss of Instrument Air and, therefore, this issue was considered of low safety significance and a Green finding (**FIN 50-341/2003-06-01**).

Enforcement: No violation of regulatory requirements occurred. The inspectors determined that the finding did not represent a non-compliance because it occurred on nonsafety-related, plant support equipment.

1R20 Refueling and Outage (71111.20)

a. Inspection Scope

The inspectors observed control room activities associated with the shutdown for the scheduled refueling outage including removing equipment from service, inserting control rods, completing mode specific surveillance testing, and monitoring reactor coolant temperature. The inspectors attended daily outage meetings, reviewed outage control center and control room operator logs, and conducted daily control room tours to ensure that shutdown safety was maintained throughout the outage, reactor coolant system instrumentation provided accurate information, the decay heat removal systems were functioning properly, and inventory and reactivity controls were maintained. The inspectors conducted periodic observations of outage related work activities to ensure that work activities were performed in accordance with plant procedures. The inspectors performed tours of the Turbine Building, Reactor Building, and Drywell 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. The inspectors observed refueling activities to verify that the licensee adhered to established procedures and Technical Specification requirements for handling of irradiated fuel. The inspectors verified that the licensee maintained secondary containment in accordance with Technical Specification requirements.

The following major activities were observed or performed:

- Fuel bundle oriented incorrectly in Spent Fuel Pool, CARD 03-12015
- The inspectors reviewed the licensee's restart restraint process and verified closure of selected issues. Issues reviewed included the RHR reservoir service water pumps' bolt and column degradation and the gasket material (foreign material intrusion) identified in the reactor pressure vessel
- Routine drywell tours
- Attended Outage Planning Meetings
- Verified Defense-in-Depth for shutdown cooling
- Refueling Operations
- Drywell closeout Inspection
- Verified Shutdown Cooling Tagouts
- Verified Completion of Restart Restraint Items

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

.1 Routine Surveillance Testing

a. Inspection Scope

The inspectors observed surveillance testing or reviewed test data for risk-significant systems or components to assess compliance with Technical Specifications, 10 CFR 50, Appendix B, and licensee procedure requirements. The testing was also evaluated for consistency with the UFSAR. The inspectors verified that the testing demonstrated that the systems were ready to perform their intended safety functions. The inspectors reviewed whether test control was properly coordinated with the control room and performed in the sequence specified in the surveillance instruction, and if test equipment was properly calibrated and installed to support the surveillance tests. The procedures reviewed are listed in the attached List of Documents Reviewed. The specific surveillance activities assessed were:

- Reactor pressure vessel hydrostatic and system leakage test
- EDG 11 slow start surveillance test

b. Findings

No findings of significance were identified.

.2 Scram Timing Tests Conducted During Reactor Pressure Vessel Leakage Test

a. Inspection Scope

The inspectors questioned the acceptability of conducting control rod scram time tests during system leakage test of the reactor pressure vessel with average reactor coolant temperatures above 200 degrees Fahrenheit. The inspectors interviewed operations and licensing personnel and the staff from the office of Nuclear Reactor Regulation to understand whether moving control rods with reactor pressure vessel temperatures elevated above 200 degrees Fahrenheit was acceptable. To aid in the issue disposition, the inspectors reviewed the Code of Federal Regulations, Improved Technical Specifications and associated amendment requests and Bases and Amendment 114 to Special Operations Technical Specification 3/4.10.7, “Hydrostatic Leak Test.”

b. Findings

Introduction

A Green NCV was identified for the failure to comply with 10 CFR 50.71(e) related to not updating the Updated Safety Analysis Report (USAR) with the effects of a safety analysis and evaluation performed by the licensee in support of an approved license amendment (License Amendment 114), which allowed average reactor coolant system temperature to be between 200 and 212 degrees Fahrenheit for conducting inservice leak tests and hydrostatic tests while in Mode 4. Specifically the licensee’s safety analysis and evaluation credited all control rods as fully inserted during these testing evolutions; however, the licensee did not keep all control rods fully inserted, but performed control rod scram time testing during the period when reactor coolant temperature was above 200 degrees.

Description: On September 5, 1997, the licensee requested (as noted in Letter NRC-97-0107) a special test exception to allow in-service leak testing and hydrostatic testing to be performed at temperatures greater than 200 degrees Fahrenheit but less than or equal to 212 degrees Fahrenheit in Mode 4. Mode 4 is defined in Technical Specifications, in part, as average reactor coolant temperature less than or equal to 200 degrees Fahrenheit. The next higher mode, Mode 3, was defined, in part, as average reactor coolant temperature greater than 200 degrees Fahrenheit. This request was considered an exigent Technical Specification change to support an October 3, 1997, mid-cycle outage to replace a leaking fuel assembly.

Enclosure 2, “No Significant Hazards Consideration,” of Letter NRC-97-0107 stated that hydrostatic testing at temperatures between 200 and 212 degrees Fahrenheit will not significantly increase the probability of an accident previously evaluated due, in part, to all control rods fully inserted and nearly water-solid conditions. Therefore, the stored energy in the reactor core coolant would be very low and the potential for causing fuel failures with a subsequent increase in coolant activity was minimal. On September 30, 1997, the NRC issued Amendment 114, to change Technical Specification 3/4.10.7. In the Safety Evaluation, the staff agreed that permitting the average reactor coolant temperature to be between 200 and 212 degrees Fahrenheit while performing the leak test would not substantially affect the results of potential accidents that may occur with the average reactor coolant temperature since the hydrostatic leak tests are performed near water solid and with all control rods inserted. The change was implemented but the associated Technical Specification Bases were not changed to state the rods would be inserted during the leakage tests.

Procedure 43.000.009, “Reactor Pressure Vessel System Leakage Test,” was used by the licensee to satisfy the pressure test and it was revised on October 13, 1997, allowing the test to be performed at temperatures between 200 and 212 degrees Fahrenheit. Nevertheless, step 5.1.2 of Procedure 43.000.009 provided the option of conducting Procedure 54.000.003, “Control Rod Scram Insert Time Test,” during the leakage test. Allowing scram timing tests to be performed contradicted the analysis statement that the test was performed with all control rods fully inserted as described in Enclosure 2 of Letter NRC-97-0107. However, during the October 1997 outage, operators conducted the leakage test under Infrequently Performed Test or Evolution 97-05 where operators raised reactor coolant system temperature between 200 and 212 degrees Fahrenheit and scram time tests were <u>not</u> performed.

The licensee did not revise the USAR with the effect of the safety analysis/evaluation that supported license amendment 114. Consequently, since this information was not in the USAR, the information was not available while preparing new Procedure 24.137.21,”Reactor Pressure Vessel System Leakage Test,” which replaced Procedure 43.000.009. Step 5.1.2 of Procedure 24.137.21 also permitted scram time testing of the control rods even though the licensing basis for TS 3/4.10.7 specified that all control rods would be fully inserted with reactor pressure vessel temperature between 200 and 212 degrees Fahrenheit during the hydrostatic test.

Between 1998 and 1999, the licensee began to convert current Technical Specifications to Improved Technical Specifications. While evaluating the less restrictive changes to Technical Specifications 3/4.10.7, the special operations specification in improved Technical Specifications 3/4.10.1, “Inservice Leak and Hydrostatic Testing Operation,” did not restrict the hydrostatic test temperature to a maximum of 212 degrees Fahrenheit, and this restriction was removed The condition that the control rods be fully inserted during hydrostatic testing as specified in Amendment 114 was not specifically evaluated while processing amendment 134, for the technical specification conversion.

Since all subsequent revisions to Procedure 24.137.21 allowed scram timing tests to be performed during the reactor pressure vessel leakage test, and the improved Technical Specification bases did not specify a restriction that all control rods were inserted during hydrostatic testing, operators potentially continued to perform scram time testing at average reactor coolant temperatures greater than 200 degrees Fahrenheit since 1998. Recently, operators conducted scram time testing above 200 degrees Fahrenheit during Refueling Outage 9. Given the existence of amendment 114, the inspectors challenged the acceptability of performing the scram timing test during the hydro test at temperatures greater than 200 degrees Fahrenheit. On July 9, 2003, the licensee initiated CARD 03-11742 to document the inspectors’ questions.

<u>Analysis</u>: This finding was greater than minor because this condition allowed a procedure change that permitted the conduct of control rod scram testing (i.e., withdrawal of selected control rods) concurrent with operation under the provisions

of Technical Specification 3/4.10.7, that would have otherwise required prior NRC approval in accordance with 10 CFR 50.59. The conduct of control rod scram testing concurrent with operation under the provisions of Technical Specification 3/4.10.7 impacts the equipment lineup (inserted reactivity) assumed as the initial condition for a safety analysis/evaluation for an approved license amendment. This initial condition (all rods fully inserted) was credited in the licensee's application for amendment 114 and the NRC staff safety evaluation documenting approval of the license amendment 114.

The failure to comply with 10 CFR 50.71(e) and the resultant inappropriate procedure change created a situation where less shutdown margin than originally evaluated and accepted by the NRC staff was present when these activities (control rod testing and hydrostatic testing) were conducted concurrently. Thus this finding affects the Initiating Events Cornerstone of the Reactor Safety Strategic Performance Area. Using Phase 1 of Manual Chapter 0609, "Significance Determination Process," Appendix A, Attachment 1, the inspectors determined this issue was of very low significance (Green) because the finding did not contribute to the likelihood of a primary or secondary system loss of coolant accident, does not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available nor increase the likelihood of a fire or internal/external flood.

Enforcement: 10 CFR 50.71(e) states, in part, that the licensee shall revise the USAR to include the effects of all safety evaluations performed by the licensee either in support of requested license amendments or in support of conclusions that changes did not involve an unreviewed safety question. Contrary to the above, the licensee failed to revise the USAR with the effects of a safety analysis and evaluation in support of approved license amendment 114, which allowed average reactor coolant system temperature to be between 200 and 212 degrees Fahrenheit for conducting in-service leakage tests and hydrostatic tests while in Mode 4, Technical Specification 3/4.10.7. Specifically the licensee's safety analysis and evaluation credited all control rods as fully inserted during these testing evolutions. Because the failure to revise the USAR with this information is of very low safety significance and has been entered into the licensee corrective action program (CARD 03-11742), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (**NCV 50-341/2003-006-02**).

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Plant Walkdowns, Radiological Boundary Verification, and Radiation Work Permit (RWP) Reviews

a. Inspection Scope

The inspectors reviewed the station's implementation of physical and administrative controls over access to radiologically restricted areas, including worker adherence to these controls, by reviewing station procedures, RWPs, electronic dosimetry alarm set points, and walking down radiologically significant areas (radiation areas, high radiation areas (HRAs), and locked HRAs) of the station. Specifically, areas in the Reactor Building (including the Drywell and Steam Tunnel), Turbine Building, and On-Site Storage Facility were observed to verify these areas were posted and controlled in accordance with 10 CFR Part 20, licensee procedures, and Technical Specifications.

The inspectors also reviewed RWPs with the potential for internal dose assignments of 50 millirem committed effective dose equivalent or more, to assess the appropriateness of engineering controls employed to minimize dose. Through day 14 of RF09, the licensee had not assigned any internal dose greater than 50 millirem committed effective dose equivalent. Therefore, the inspectors reviewed licensee tracking reports and assessed the adequacy of internal dose assessments for two personal contamination events resulting from under vessel control rod drive replacement activities.

b. Findings

No findings of significance were identified.

.2 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed CARDs completed in conjunction with the refueling outage which focused on access control to radiologically significant areas, radiation worker practices, and radiation protection (RP) technician practices. The inspectors reviewed these documents to assess the licensee’s ability to identify repetitive problems, contributing causes, the extent of conditions, and implement corrective actions intended to achieve lasting results.

b. Findings

No findings of significance were identified.

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

.1 Radiological Work/ALARA Planning

a. Inspection Scope

The inspectors reviewed the station’s procedures for radiological work/ALARA planning and scheduling, and evaluated the dose projection methodologies and practices implemented for the RF09 refueling outage, to verify that sound technical bases for outage dose estimates existed. Specifically, the inspectors reviewed radiologically significant (i.e., collective dose estimates of 5.0 person-rem or greater) RWP/ALARA planning packages, including:

- Control Rod Drive Exchange (RWP No. 03-1117)
- In-Service Inspections (RWP No. 03-1112)

- Insulation Removal, Repair, and Replacement (RWP No. 03-1110)
- Install and Remove Scaffold, Power, and Lights in Drywell and Steam Tunnel (RWP No. 03-1105)
- Install and Remove Temporary Shielding and Install Permanent Shielding in Drywell (RWP No. 03-1103)
- Perform Refuel Activities on Reactor Building Level 5 (RWP No. 03-1251)
- E11FO50A and E11FO50B - Perform Softseat Replacement in lieu of Hardseat Modifications (RWP No. 03-1127)

The inspectors reviewed these seven radiologically significant RWP/ALARA planning packages to verify that adequate person-hour estimates, job history files, lessons learned, industry experiences, and the use of mockups (where applicable) were utilized in the ALARA planning process and to confirm that these elements were integrated into the associated RWPs.

b. Findings

No findings of significance were identified.

.2 Job Site Inspections and ALARA Controls

a. Inspection Scope

The inspectors observed work activities in the radiologically restricted areas that were performed in radiation areas, HRAs, and locked HRAs to evaluate the use of ALARA controls. Specifically, the inspectors reviewed radiological surveys, attended pre-job radiological briefings, and assessed job site ALARA controls, at least in part, for the following work activities:

- Control Rod Drive replacement (RWP No. 03-1117)
- Inspection of the East Main Steam Reheater (RWP No. 03-1207)
- Removal of the Main Steam Bypass Valve and transfer to the rebuild area (RWP No. 03-1207)
- In-vessel visual inspection of the Control Rod Drive blades on Reactor Building Level 5 (RWP No. 03-1251)
- Traversing In-Core Probe room entry for explosive valve detonation and replacement, fire detector operability/sensitivity test, and visual inspection of penetrations (RWP No. 03-1152)

Worker instruction requirements, including protective clothing, engineering controls to minimize dose exposures, the use of predetermined low dose waiting areas, as well as the on-the-job supervision by the work crew leaders and RP technicians, were observed to determine if the licensee had maintained the radiological exposure for these work activities ALARA. Enhanced job controls including RP technician use of electronic teledosimetry and cameras was also evaluated to assess the licensee's ability to maintain real-time doses ALARA in the field. Additionally, the inspectors observed the implementation of dosimetry placement changes necessitated by significant dose rate gradients under the reactor vessel during control rod drive replacement activities (per the requirements of RWP No. 03-1117).

b. Findings

No findings of significance were identified.

.3 Radiation Worker/RP Technician Performance

a. Inspection Scope

The inspectors observed radiation workers performing the activities described in Section 2OS2.2 and evaluated their awareness of radiological conditions, personal electronic dosimetry alarm set points, and their implementation of applicable radiological controls. Additionally, the inspectors observed RP technician control over these work activities to evaluate if the technicians were aware of the radiological conditions in their workplace, the RWP controls/limits in place for the work activities, and that they appropriately communicated any significant changes in radiological conditions to the workers whose activities they were overseeing.

b. Findings

No findings of significance were identified.

.4 Verification of Dose Estimates, Dose Trending, and Dose Tracking Systems

a. Inspection Scope

The inspectors reviewed the licensee's total outage dose estimates, selected individual job dose estimates and the related dose trending for the refueling outage. As of April 9, 2003, (day 12 of an estimated 29 day outage), the licensee had recorded a collective dose of 57.965 person-rem compared to the estimate of 66.500 person-rem for that point in RF09 (and a total estimate of 110.000 person-rem for the outage). The inspectors reviewed and discussed with RP staff the scope change for RWP 03-1127 (50A/50B valve seat modification), and associated re-evaluation of the ALARA plan, to evaluate the licensee's ability to assess the effectiveness of an ALARA plan in a timely manner and institute changes in the plan or its execution, as warranted. The licensee's dose tracking system was also reviewed to determine if the level of dose tracking detail, dose report timeliness, and report distribution were sufficient to support the control of collective and individual dose for the outage.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed licensee CARDs completed in conjunction with the RF09 outage which focused on ALARA planning and controls. The inspectors additionally reviewed a recent Nuclear Quality Assurance Audit of the RP program (focusing on

Nuclear Quality Assurance's review of ALARA planning and controls). The inspectors reviewed these documents to assess the licensee's ability to identify repetitive problems, contributing causes, the extent of conditions, and develop corrective actions intended to achieve lasting results.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS3 Radioactive Material Control Program (71122.03)

Review of an Unrestricted Release of Radioactive Material Occurrence During Contractor Whole Body Counting In-processing

a. Inspection Scope

The inspectors reviewed the circumstances associated with the unrestricted release of low-level radioactive material (i.e., external contamination on a lanyard) that occurred on March 20, 2003, following whole body counting of contractors as part of their station in-processing. Specifically, the inspectors reviewed the licensee's initial CARD, investigative documents (including worker statements and a time line of the event), whole body count and survey data, and the incident was discussed with the radiation protection manager and several members of the RP staff.

b. Findings

Introduction: A self-revealing Green finding and an associated Non-Cited Violation (NCV) were identified for the failure to maintain control of a measurable amount of radioactive material (i.e., external radioactive contamination on a lanyard) identified during whole body counting of contractors during station in-processing.

Description: On March 20, 2003, a whole body count vendor operator was conducting whole body counting of contract personnel as part of their station in-processing in the General Training and Orientation Center (GTOC). The GTOC is located outside of the station's Protected Area but is within the Owner Controlled Area (OCA) of the station. At approximately 11:39 a.m., the first contract individual was identified as having a positive whole body count (WBC) with Mn-54, Co-58, and Cs-137 (6.3 nanocuries (nCi), 6.4 nCi, and 2.4 nCi, respectively). The vendor operator instructed the contractor to remove his lanyard and conducted a second WBC which was negative for the presence of measurable radioactivity. Subsequently, the vendor operator returned the lanyard to the contractor with the instructions that "...he discard the old lanyard, since it may be mildly contaminated." At approximately 12:12 p.m., a second contractor's WBC was positive for the presence of radioactivity (approximately 9.8 nCi Co-60). The vendor operator had the contractor remove his fleece vest and repeated the WBC. The follow-up WBC was negative for the presence of measurable radioactivity, and, similarly, the vendor operator returned the fleece vest to the contractor with a suggestion that he launder the vest. At approximately 1:59 p.m., a third contractor's WBC indicated possible positive activity with a 143 keV peak. The contractor indicated that his boots may have been previously contaminated, thus the vendor operator performed another WBC of the contractor with his boots removed; the follow-up WBC was negative for the presence of measurable radioactivity.

This event was self-revealing when, at approximately 2:15 p.m., the RP Instrument Supervisor and a principle radiological engineer went out to the GTOC to check on the vendor operator and they were subsequently informed about the recent positive WBCs and apparent external contaminations. The RP Instrument Supervisor initiated an investigation and was able to take control of the contaminated materials/clothing from two of the individuals who were still within the OCA by approximately 5:30 p.m. However, the first contractor had apparently left the OCA with the externally contaminated lanyard. Radiation Protection management was able to contact the first contractor later that evening; the contractor indicated that he was still in possession of the lanyard, and RP management requested that he place the lanyard in a bag and bring it into the station the following morning (March 21, 2003). However, when the contractor returned the lanyard to the station, follow-up surveys and gamma spectroscopic analyses could not identify the radioactive material on the lanyard, in the contractor's automobile, or in the bag and briefcase the contractor said he had placed the lanyard in overnight. Extensive licensee surveys of the WBC room and the GTOC did not identify any additional detectable radioactive material.

The licensee's investigation revealed that the vendor operator was apparently not cognizant of the procedural and regulatory requirements to take control of any measurable radioactive material outside of the radiologically restricted areas. The licensee additionally identified that there was less than adequate vendor oversight by the RP department and procedure deficiencies which contributed to the occurrence.

Analysis: The inspectors determined that the issue was associated with the "Program and Process" and "Human Performance" attributes of the Public Radiation Safety Cornerstone and affected the cornerstone objective in ensuring adequate protection of public health and safety from exposure to radioactive materials released into the public domain. Also, the issue involved an occurrence in the licensee's radioactive material control program that is contrary to both NRC regulations and licensee procedures. Therefore, the issue was more than minor and represents a finding which was evaluated using the significance determination process (SDP) for the Public Radiation Safety Cornerstone.

The inspectors determined that the vendor operator's apparent lack of knowledge as to the requirements to control measurable amounts of radioactive material, which was exacerbated by less than adequate RP oversight of the vendor and procedure deficiencies, led to the unrestricted release of measurable radioactive material into the public domain (outside the OCA). As such, the inspectors determined utilizing Manual Chapter 0609, Appendix D, "Public Radiation Safety SDP," that the finding involved radioactive material control, but transportation was not involved. Additionally, public radiation exposure was not greater than 0.005 rem (5 millirem) and the licensee did not have more than five radioactive material control occurrences (in the previous

8 quarters). Consequently, the inspectors concluded that the SDP assessment for this finding was of very low safety significance (Green).

Enforcement: Title 10 CFR 20.1802 requires that the licensee shall control and maintain constant surveillance of licensed material that is in a controlled or unrestricted area and that is not in storage. On March 20, 2003, the licensee failed to maintain control of a measurable amount of licensed radioactive material (i.e., external radioactive contamination on a lanyard) identified during whole body counting of a contractor. This failure constitutes a violation of 10 CFR 20.1802. However, because the licensee documented this issue in its corrective action program (CARD 03-00690) and because the violation is of very low safety significance, it is being treated as an NCV (**NCV 50-341/03-006-03**).

3. SAFEGUARDS

Cornerstone: Physical Protection

3PP2 Access Control (Identification, Authorization and Search of Personnel and Packages (IP 71130.02)

a. Inspection Scope

The inspectors reviewed the licensee's protected area access control equipment testing and maintenance procedures to determine if testing was performance-based, challenged the detection capabilities of the equipment, and was in accordance with security plan requirements. The inspectors observed licensee testing of all access control equipment to determine if testing and maintenance practices were performance based. On two occasions, during peak ingress periods, the inspectors observed in-processing search of personnel and packages to determine if search practices were conducted in accordance with regulatory requirements, and that sufficient security force staffing was available for the search functions.

The inspectors reviewed a sample of licensee security logged events and other security documents for identification and resolution of problems. In addition, the inspectors interviewed security managers to evaluate their knowledge and use of the licensee's corrective action system.

b. Findings

No findings of significance were identified.

3PP3 Response to Contingency Events (71130.03)

a. Inspection Scope

The inspectors reviewed the current Plant Protective Strategy. The inspectors also conducted a walkdown of the protected area boundary and alarm system and observed testing of selected protected area alarm zones. The inspectors reviewed licensee drill and exercise critiques pertaining to response to security contingency events.

The inspectors reviewed a sample of licensee security logged events for identification and resolution of problems. In addition, the inspectors interviewed security managers to evaluate their knowledge and use of the licensee's corrective action system.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification (71151)

Security Performance Indicator Verification

a. Inspection Scope

The inspectors verified the data for the Physical Protection Performance Indicators pertaining to Fitness-For-Duty, Personnel Reliability and Personnel Screening Program, and Protected Area Security Equipment. Specifically, a sample of plant reports related to security events and other applicable security records were reviewed for the period between October 1, 2002 and March 31, 2003.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Review of Identification and Resolution of Problems for Inservice Inspection Related Issues

a. Inspection Scope

From April 7, 2003 through April 11, 2003, the inspectors performed a review of a sample of inservice inspection related problems that were identified by the licensee and entered into the corrective action program. The inspectors conducted this in an on-site office on the second floor of the Availability Improvement Building within the site protected area. The inspectors reviewed these corrective action program documents to confirm that the licensee had appropriately described the scope of the problems. Additionally, the inspectors' review included confirmation that the licensee had an appropriate threshold for identifying issues and had implemented effective corrective actions. The inspectors performed these reviews to ensure compliance with 10 CFR 50, Appendix B, Criterion XVI "Corrective Action" requirements. The specific corrective action documents that were reviewed by the inspectors are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

.2 Routine Review of Identification and Resolution of Problems

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 system at an appropriate level, that corrective actions were performed in a timely manner and that adverse trends were identified and addressed. The inspectors selected the following documents to determine if problem characterization was accurate and to verify that extent of condition reviews were adequately completed or were in the process of being performed.

- West Station Air Compressor
- Division 1 Thermal Recombiner Repeat Failures

4OA3 Event Followup (71153)

.1 Review of Licensee Event Reports

a. Inspection Scope

The inspectors performed an onsite review of records to evaluate the root cause and corrective actions for the licencee event reports (LERs) discussed in the "Findings" section below. The inspectors evaluated the timeliness, completeness, and adequacy of the root cause and corrective actions in accordance with the requirements of 10 CFR Part 50, Appendix B, as appropriate.

b. Findings

(Closed) LER 50-341/02-005-00: Non-Conservative Setpoint for Stability Option III (OPRM) Period Based Algorithm.

On November 21, 2002, the General Electric Company informed the licensee that the stability Option III period based detection algorithm range for minimum period detection (Tmin) values may be non-conservative. General Electric recommended that the average power range monitor, oscillation power range monitor (OPRM) upscale trip (Technical Specification Limiting Condition for Operation 3.3.1.1, Function 2.f) be considered inoperable for plants with Tmin set at greater than 1.2 seconds. The licensee had Tmin set at 1.4 seconds so their OPRMs were declared inoperable.

General Electric's 10 CFR Part 21 Notification concerning this issue states, "With Tmin set to 1.2 seconds or lower, the OPRM is considered to be operable." Additionally, the report also provided guidance on the setting of the maximum period detection (Tmax) value, stating, "The OPRM is considered operable for values of Tmax set to 3.0 seconds or higher." At the time, the licensee had Tmax set at 3.0 seconds.

Technical Service Request 32279 was initiated to change the OPRM Tmin and Tmax values to 1.2 and 1.4, respectively for all four OPRMs. Work requests 000Z023654, 000Z023657, 000Z023658, and 000Z023659 were performed to make the necessary changes. The inspectors verified that the setpoints were changed, as required, on all OPRMs. The LER was reviewed by the inspectors and no findings of significance were identified. The licensee documented this issue in CARD 02-18019. This LER is closed.

.2 Momentary Loss of Shutdown Cooling

a. Inspection Scope

On April 28, 2003, the inspectors evaluated the control room staff response to a loss of shutdown cooling due to operator error. The inspectors observed plant parameters and operator activities associated with restoration of decay heat removal. The inspectors reviewed operations logs and interviewed operators to understand the circumstances surrounding the unexpected momentary loss of shutdown cooling.

b. Findings

Introduction: A self-revealed Green NCV was identified for failure to comply with 10 CFR 50, Appendix B, Criterion V, related to operators failing to depress the open push button for E1150-F008, the RHR Outboard Suction Isolation Valve, when restoring Reactor Protection System (RPS) B power. As a result of this failure, the valve closed when RPS B power was restored and the plant lost shutdown cooling.

Description: On April 28, 2003, the licensee was restoring RPS B in accordance with Procedure 23.316. During the power restoration process, the outboard RHR insolation valve would close unless the valve's open button was depressed prior to power restoration. Contrary to the procedural instruction, the operator failed to depress the open button and the valve closed as designed. The valve isolated the suction flow path for RHR thus causing a loss of the shutdown cooling. The licensee recognized the loss of shutdown cooling and restored decay heat removal within 1 hour.

Analysis: The performance deficiency associated with this event was the failure to follow procedures for shifting RPS busses. The inspectors compared this finding to those listed in Appendix E of Manual Chapter 0612 and concluded it most closely matched Item 4.b and was more than minor because it caused a plant transient - loss of shutdown cooling. The inspectors compared the finding to Appendix G of NRC Inspection Manual 0609 and concluded that finding screened as Green. This finding identified a loss of shutdown cooling. However, Appendix G does not require a quantitative analysis for loss of shutdown cooling for the plant conditions of greater than 2 hours to boil. The finding is of very low safety significance (Green) based on the guidance of Appendix G.

Enforcement: 10 CFR 50, Appendix B, requires, that activities affecting quality shall be prescribed by documented instructions, procedures or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures or drawings. The licensee developed Procedure 23.316 to shift RPS busses and it required depression of the open pushbutton for E1150-F008. Contrary to these requirements, the operator failed to depress the pushbutton and caused a loss of shutdown cooling. The licensee entered this finding into the corrective action program (CARD 03-17206). Because of the very low safety significance and because the issue has been entered into the licensee's corrective action program, it is being treated as a NCV, consistent with Section VI.A.1 of the NRC enforcement Policy **(NCV 50-341/03-006-04)**.

4OA5 Other

The inspectors reviewed the final report for the March 18, 2003, Plant Evaluation performed by an inspection team from the Institute of Nuclear Power Operations. No further inspection was deemed necessary by NRC inspectors, and no assessment was made of the results of the inspection.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. O'Connor and other members of licensee management at the conclusion of the inspection on July 11, 2003. 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 exits were conducted for:

- Radiation Protection inspection with Mr. Cobb on April 11, 2003.
- Inservice Inspection with Mr. O'Connor on April 11, 2003.
- Safeguards inspection with Mr. S. Peterman on May 22, 2003.

.3 Re-Exit Meeting

The inspectors reviewed the inspection findings with S. Stasek and others to communicate final characterization of the issues. The meeting was held on July 22, 2203.

4OA7 Licensee-Identified Violations

The following violations of very low significance were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Manual, NUREG-1600, for being dispositioned as NCVs.

- Technical Specification 3.6.1.3 requires that a primary containment penetration be isolated within 4 hours, if the associated primary containment isolation valve is not operable. Contrary to this, on March 19, 2003, a drywell floor drain sump outboard containment isolation valve was inoperable, and the penetration was not isolated within 4 hours. This was identified in the licensee's Corrective Action

Program as CARD 03-11287. This finding is of very low safety significance because it does not represent an open pathway in the physical integrity of the reactor containment.

- 10 CFR 50, Appendix B, Criterion V, requires procedures for activities affecting quality. Contrary to this, on March 5, 2003, no procedures were established to prevent insulators from removing a small amount of insulation from Divisions 1 and 2 RHR heat exchanger piping while the plant was at power, thereby affecting the environmental qualification of pot-accident sampling system containment isolation valves. This condition existed for 14 days. This was identified in the licensee's Corrective Action Program as Condition Assessment Resolution Document 03-14848. This finding is of very low safety significance because of the small amount of insulation removed, the low probability of design basis accident occurring while the insulation was removed and no impact on valve operability.

- 10 CFR 50, Appendix B, Criterion XVI, requires that measures be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to this requirement, licensee inspections of the residual heat removal (RHR) reservoir on April 5, 2003, as a part of corrective actions to the failure of bolts on the number 2 circulating water pump, identified that adequate inspections had not been established in the RHR reservoir to identify and prevent micro biologically induced corrosion that degraded bolts, fasteners and welds on both divisions of service water pump suction columns at the ultimate heat sink. The conditions causing slow degradation of components had been developing for the life of the plant. This issue was identified in the licensee's Corrective Action Program as CARD 03-14451. This finding is of very low safety significance because, although some degradation had begun on all ten service water pumps in the reservoir, no actual loss of safety function occurred for any of the pumps.

ATTACHMENT: Supplemental Information

KEY POINTS OF CONTACT

Licensee
A. Brooks, ISI Level III
D. Cobb, Plant Manager
D. Craine, General Supervisor - Radiological Engineering
L. Craine, General Supervisor - Radiation Protection
J. Davis, Manager, Outage Management
T. Dong, Manager, Performance Engineering
T. Duffy, General Superintendent, Nuclear Security Operation
G. Heitzenrater, Assistant Manager, System Engineering
H. Higgins, Manager - Radiation Protection
R. Johnson, Supervisor, Licensing
J. Korte, Manager, Nuclear Security
W. O'Connor, Jr., Vice President Nuclear Generation
N. Peterson, Manager, Nuclear Licensing
T. Stack, Security Planning Analyst
S. Stasek, Director, Nuclear Assessment

Nuclear Regulatory Commission
M. Ring, Chief, Division of Reactor Projects, Branch 1

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-341/03-06-01	FIN	Wrong Spring Installed in West Station Air Compressor Breaker Causes Compressor Failure
50-341/03-06-02	NCV	Failure to Include Technical Specification Amendment 114 into UFSAR
50-341/03-06-03	NCV	Failure to Maintain Control of Licensed Radioactive Material
50-341/03-06-04	NCV	Operator Error Causes Loss of Shutdown Cooling

Closed

50-341/03-06-01	FIN	Wrong Spring Installed in West Station Air Compressor Breaker Causes Compressor Failure
50-341/03-06-02	NCV	Failure to Include Technical Specification Amendment 114 into UFSAR
50-341/03-06-03	NCV	Failure to Maintain Control of Licensed Radioactive Material
50-341/03-06-04	NCV	Operator Error Causes Loss of Shutdown Cooling
50-341/02-005-00	LER	Non-Conservative Setpoint for Stability Option 3 (OPRM) Period Based Algorithm

LIST OF ACRONYMS USED

ALARA	As-Low-As-Is-Reasonably-Acheivable
CARD	Condition Assessment Resolution Document
CFR	Code of Federal Regulations
EDG	Emergency Diesel Generator
F	Fahrenheit
GTOC	General Training and Orientation Center
HRA	High Radiation Area
LER	Licensee Event Report
nCi	nanocurie (1×10^{-9} curies)
NCV	Non-Cited Violation
OCA	Owner Controlled Area
OPRM	Oscillation Power Range Monitor
RF09	Fermi 2's Ninth Refueling Outage
RHR	Residual Heat Removal
RP	Radiation Protection
RPS	Reactor Protection System
RWP	Radiation Work Permit
SDP	Significance Determination Process
Tmin	Option III Period Based Detection Algorithm Range ror Minimum Period Detection
Tmax	Option III Period Based Detection Algorithm Range for Maximum Period Detection
UFSAR	Updated Final Safety Analysis Report
WBC	Whole Body Count

LIST OF DOCUMENTS REVIEWED

1R04 Equipment Alignment

CARD 03-17130, “Component Labels do not Match Procedure Line-ups,” dated 4/21/03

CARD 03-17138, “Valves not Restored to Correct Line-up Position during Valve Line-up, dated 4/20/03

Dwg. No. 6M721-2081, Rev. AO, “Control Rod Drive Hydraulic System Reactor Building Part 1 of 2,” dated 5/17/02

Dwg. No. 6M721-5703-1, Rev. W, “Control Rod Drive System Functional Operating Sketch,” dated 5/17/02

Dwg. No. 6M721-5703-2, Rev. Q, “Control Rod Drive SCRAM Discharge System Functional Operating Sketch,” dated 5/18/01

ST-OP-315-0010-001, Rev. 13, “Operations Training - Control Rod Hydraulics”

23.106, Attachment 1, “Control Rod Drive Hydraulic System Valve Lineup”

1R04 Equipment Alignment - Semi Annual Inspection

System Health Fermi 2, E1100, Residual Heat Removal, 4th Quarter 2002

System Health Fermi 2, E1100, Residual Heat Removal, 3rd Quarter 2002

System Health Fermi 2, E1100, Residual Heat Removal, 2nd Quarter 2002

System Health Fermi 2, E1100, Residual Heat Removal, 1st Quarter 2002

CARD No. 01-19330, “Nuclear Operator Finds and Stops Leak from Division 2 RHR (Mispositioned Valve), dated 11/24/01

Plant Technical Procedure - Fermi 2, System Operating Procedure 23.205, “Residual Heat Removal System,” Revision 80

Safety Tagging Record 2003-000141, Residual Heat Removal System, Div 1 RHR Outage

Drawing No. 6M721-5706-2, “Residual Heat Removal Division I Functional Operating Sketch,” Revision W

Drawing No. 6M721-5706-1, “Residual Heat Removal Division II Functional Operating Sketch,” Revision Y

CARD 03-17380, “Significant Errors Identified on Plant Drawing I-2334-57,” dated May 7, 2003

CARD 03-10525, “Procedure 24.413, Attachment 3, Control Center HVAC System Instrument Lineup Sheet has Numerous Errors,” dated May 6, 2003

<u>1R05</u> <u>Fire Protection</u>

Plant Technical Procedure 28.507.02, Rev. 7, “Fire Door Surveillance Test”

CARD 03-10893, “NRC personnel question condition of fire seal,” dated 6/27/03

CARD 03-01445, “Emergency light not working,” dated 6/25/03

CARD 03-01263, “Lights do not illuminate,” dated 6/26/03

Hourly fire watch log on 6/25/03

CARD 00-15864, “Lack of analysis supporting adequacy of Div. 1 and Div. 2 BOP batteries to support actions required for 20.000.18,” dated 6/16/03

CARD 00-15866, “Lack of analysis supporting stripping of DC supply to RHR complex during implementation of 20.000.18,” dated 6/16/00

CARD 00-15865, “Procedural discrepancies that may prevent or negate actions required for 20.000.18,” dated 6/16/00

Plant Technical Procedure 28.507.01, Rev. 6, “Fire Barrier Inspection”

MES17, Rev. 13, “Engineering Support Conduct Manual Chapter 17 - Conduct of Design Verification”

<u>1R06</u> <u>Flood Protection</u>

UFSAR Figure 9A-4, “Fire Protection Evaluation, Reactor and Auxiliary Buildings, First Floor Plan, Elevation 583.5 ft.”

UFSAR Figure 9A-6, “Fire Protection Evaluation, Reactor and Auxiliary Buildings, Second Floor Plan, Elevation 613.5 ft.”

<u>1R07</u> <u>Heat Sink Performance</u>

Plant Technical Procedure 47.205.02, “Residual Heat Removal Division 2 (South) Heat Exchanger Performance Test,” Revision 8

1R08 Inservice Inspection

I. DOCUMENTS REVIEWED

CARD 01-20653; Cracked Stiffener Plate Weld; dated October 30, 2001

CARD 01-20682; Debris Dropped Into Reactor Vessel; dated November 13, 2001

CARD 02-10515; Support E113154-603 Has Rust; dated August 6, 2002

02-10913; Support E113154-002 Has Rust; dated February 1, 2002

CARD 02-10968; GE RICSIL 087 Shroud Cracking Mitigation Effectiveness; dated March 15, 2002

CARD 02-14797; BWR VIP Evaluation Of Core Shroud Crack; dated October 3, 2002

CARD 02-19486; ISI NDE Evaluation 99-056 Requires Revision; dated September 24, 2002

Condition Assessment Resolution Documents Issued As a Result of Inspection Activities

03-16118; Apparent Differences Between Fermi WPS A11-2.5 And Original Tension Test data; dated April 9, 2003

03-16367; Evaluate Of Crevice Corrosion Cracking Degradation Mechanism For Recirculation Inlet Nozzles; dated April 9, 2003

03-16369; Prepare 39 NDE Series Procedure To Address RI ISI Expanded Weld Coverage; dated April 9, 2003

03-14501; Documentation Deficiency Regarding The Use Of a 1-Mil Wire As The Resolution Standard For Performance Of IVVI Inspections; dated April 10, 2003

Code Replacement/Repair Activities

WO CM-Z013499; Division 2 RHR HX Cracked Stiffener Plate Weld; dated November 6, 2001

WO CM-Z984589; Hydrogen Recombiner Control Valve Replacement; dated August 11, 2000

WO CM-Z991909; RHR HX Outlet Relief Valve Replacement; dated November 27, 2001

Drawings

E232-901-11; Lower Vessel Shell Assembly Machining and Welding; Revision 11

E232-902-16; Upper Vessel Shell Assembly Machining and Welding; Revision 16

6M721-5360-5; Inservice Inspection Detail Drawing Reactor Vessel Category B-A, 8, B-H, Components Reactor Building Unit #2; Revision 5

6M721-5363-5; Inservice Inspection Detail Drawing Reactor Vessel CRD and Flux Monitor Nozzles-Reactor BLDG Unit 2; Revision 5

Nondestructive Examination Reports

RF 09-26; Dye Penetrant Examination Of SLC Pipe-To-Elbow 2 Inch Weld FW-C41-2979-63564; dated April 4, 2003

RF 09-27; Dye Penetrant Examination Of SLC Pipe-To-Elbow 2 Inch Weld FW-C41-2979-63565; dated April 4, 2003

RF 09-28; Dye Penetrant Examination Of SLC Pipe-To-Reducer Weld FW-C41-5058-54355; dated April 4, 2003

RF-09-07; Ultrasonic Examination Of Recirculation Riser Inlet Nozzle-To-Safe End Weld 2-303G; dated April 6, 2003

96-038;Ultrasonic Examination Of Nozzle-To-Shell Weld 4-316C; dated October 17, 1996

13-314E; Ultrasonic Examination Of Recirculation Inlet Nozzle-To-Shell Weld; dated October 12, 1998

15-308B; Ultrasonic Examination Of Vertical Shell Weld 15-308B; dated October 14, 1998

Procedures

GE-UT-209; Procedure For Automated Ultrasonic Examination of Dissimilar Metal Welds, And Nozzle To Safe End Welds; Revision 12

GE-UT-705; Procedure For The Examination of Reactor Pressure Nozzle Inner Radius And Nozzle To Vessel Welds With The Geris 2000 OD In Accordance With Appendix VIII; Revision 0

GE-UT-704; Procedure For The Examination of Reactor Pressure Vessel Welds With Geris 2000 OD In Accordance With Appendix VIII; Revision 0
PDI-UT-1; PDI Generic Procedure For The Ultrasonic Examination Of Ferritic Piping Welds; Revision 3

43.000.017; Reactor Pressure Vessel- Invessel Internals Inspection; Revision 13

39.NDE.001; Liquid Penetrant Examination, Solvent Removable; Revision 21

39.NDE.002; Magnetic Particle Examination; Revision 22

GE-ADM-1002; Procedure For Nondestructive Examination Data Review And Analysis Of Recorded Indications; Revision 4

01-049; RI-ISI Degradation-Specific Inspection Requirements And Examination Methods; Revision 0

Miscellaneous Documents

Weld procedure specification A11-2.1; Manual Gas Tungsten Arc Welding, Revision 1

Weld procedure specification A11-1.2; Manual Shield Metal Arc Welding, Revision 1

Procedure Qualification Record WA11-2.5, dated October 9, 1987

Procedure Qualification Record WA11-2.1, dated October 21, 1987

Procedure Qualification Record WA11-3.1

Procedure Qualification Record WA11-3.3, dated June 15, 1990

Radiographic Records for Reactor Vessel Longitudinal Weld 15-308B; dated September 1969 and June 1970

DER 91-0262; ISI-NDE Linear Indications On RHR Piping; dated April 25, 1991

DER 91-0234; ISI-NDE Linear Indications On Recirculation Piping; dated May 2, 1991

ROC-001; Minutes Of The Element Selection Meeting For The Risk-Informed ISI Project At The Fermi 2 Nuclear Power Plant; dated March 2, 2001

II. INFORMATION REQUESTED ON JANUARY 2, 2003, BY E-MAIL (To A. Brooks)

A. Please provide the following information to Melvin S. Holmberg at the Region III NRC office located at 801 Warrenville Rd, Lisle IL 60532, no later than March 14, 2003, to support the NRC Inservice Inspection (IP 71111.08) scheduled to begin at the Fermi site April 07, 2003.

1) A detailed schedule of nondestructive examinations planned for Class 1 & 2 systems and containment, performed as part of your ASME Code ISI Program during the scheduled inspection week. This should also include any special nondestructive examinations of core internal components such as the core shroud welds.

2) A copy of the procedures used to perform the examinations identified in A.1. For ultrasonic examination procedures qualified in accordance with Appendix VIII, of Section XI of the ASME Code, provide documentation supporting the procedure

qualification. This documentation should include the test data identifying the types of defects used in the procedure qualification, the equipment used (cables, probes, transducers including serial numbers) and the Code Edition used for qualification. Additionally, the data supporting the detection and sizing capability of the procedure is to be provided.

3) A copy of any ASME Section XI, Code Relief Requests applicable to the examinations identified in A.1.

4) A copy of the 90 day ISI summary report from the previous outage.

5) A list identifying nondestructive examination reports (ultrasonic, radiography, magnetic particle, dye penetrant, visual (VT-1, VT-2, VT-3)) which have identified relevant indications on Code Class 1 & 2 systems in the past two refueling outages.

6) List of welds in Code Class 1, 2 and 3 systems which have been completed since the beginning of the last refueling outage (identify system, weld number and reference applicable documentation).

7) For any reactor vessel weld examinations scheduled during the inspection, provide a detailed description of the welds to be examined, extent of the planned examination and a copy of your responses to the NRC, associated with Generic Letter 83-15.

8) Identify any non-code repairs (if any) performed on Code Class 1,2, or 3 systems within the last two refueling outages.

9) Provide a list with description of ISI related issues entered into your corrective action system beginning with the date of the last refueling outage.

10) Provide a copy of any part 21 reports submitted beginning with the date of the last refueling outage.

11) Copy of responses to NRC Generic Letter 94-03: INTERGRANULAR STRESS CORROSION CRACKING OF CORE SHROUDS IN BOILING WATER REACTORS and core shroud weld examination schedule.

B. Information to be provided on-site to the inspector at the entrance meeting:

1) Updated schedule for item A.1.

2) For welds selected by the inspector from A.6 above, provide copies of the following documents:

a) Document of the weld number and location (e.g., system, train, branch).
b) Document with a detail of the weld construction.
c) Applicable Code Edition and Addenda for weldment.

d) Applicable Code Edition and Addenda for welding procedures.
e) Applicable weld procedures (WPS) used to fabricate the welds.
f) Copies of procedure qualification records (PQRs) supporting the WPS on selected welds.
g) Copies of mechanical test reports identified in the PQRs above.
h) Copies of the nonconformance reports for the selected welds.
i) Radiographs of the selected welds and access to equipment to allow viewing radiographs.

3) For the repair/replacement activities selected by the inspector provide a copy of the records of the repair or replacement required by the ASME Code Section XI Articles IWA -4000 or IWA 7000.

4) Copy of the most recent quality assurance department audit, which included the ISI program and activities. Copies of documents resolving findings in this audit.

5) For core shroud welds examined within the previous two refueling outages, provide the non-destructive examination records for the core shroud welds inspected.

6) Ready access to the Editions of the ASME Code (Sections V, IX and XI) applicable to the inservice inspection program and the repair/replacement program.

1R11 Licensed Operator Requal

SS-OP-202-0321, Rev. 0, Scenario 5

Plant Technical Procedure 20.300.72C, Rev. 0, “Loss of Bus 72C”

ARP 1D66, Rev. 12, “Steam Leak Detection Abient Temp High”

ARP 1D70, Rev. 7, “Steam Leak Detection Diff Temp High”

ARP 3D34, Rev. 7, “SEC Contm Temp High-High EOP Entry”

Technical Specification 3.5.1, “ECCS - Operating”

Technical Specification 3.6.1.3, “Primary Containment Isolation Valves (PCIVs)”

Technical Specification 3.6.2.3, “Residual Heat Removal (RHR) Suppression Pool Cooling”

Technical Specification 3.6.2.4, “Residual Heat Removal (RHR) Suppression Pool Spray”

Technical Specification 3.7.1, “Residual Heat Removal Service Water (RHRSW) System”

Technical Specification 3.7.2, “Emergency Equipment Cooling Water (EECW)/Emergency Equipment Service Water (EESW) System and Ultimate Heat Sink (UHS)”

Technical Specification 3.8.4, “DC Sources - Operating”

Technical Specification 3.3.3.1, “Post Accident Monitoring (PAM) Instrumentation”

Technical Requirement 3.7.7, “Appendix R Alternative Shutdown Auxiliary Systems”

ODE-5, Rev. 0, “Operations Department Expectation Roles and Responsibilities”

CARD 01-15056, “Simulator Foxboro indicator repairs are difficult/expensive due to obsolescence,” dated 5/9/01

CARD 02-12904, “Equivalent replacement process does not appear to adequately consider simulator hardware procurement/notification,” dated 2/6/02

CARD 03-15164, “Transformer obsolete,” dated 4/30/03

Detroit Edison Memorandum from Don Cobb, Director, Nuclear production to Site Supervisors, “Approved 2003 Projects,” dated 6/5/03

Modification No. 29446, “HPCI Keep Fill System”

Modification No. 30251/30252, “IPCS”

Modification No. 30844, “EECW Make Up System project”

Modification No. 31341, “Modify P44F603B”

Modification No. 32065/29460, “Site-Wide Communication System”

Modification No. 32148, “Replace Gland Sealing Steam Controllers”

Modification No. 32453, “Replacement of GEMAC Controller P43K817 for TBCCW TCV”

Modification No. 32455, “Replacement of GEMAC Controller for SWC System”

Modification No. 32458, “Replacement of GEMAC Controller for RBCCW”

Modification No. 32496, “Replacement of GEMAC Controller for Turbine Oil Outlet”

Modification No. 32497, “Replacement of GEMAC Controller for Hydrogen Cooler Outlet Temperature”

Modification No. 32507, “Replacement of GEMAC Controller for Generator Seal Oil Cooler”

EP 101, “Classification of Emergencies,” Revision 28

1R12 Maintenance Rule Implementation

CARD 03-17673; P5000F441, Division 2 Control Air Isolation Leaks by Closed Seat, dated 5/26/03

CARD 02-16102; Failure to Meet Acceptance Criteria on 24.129.01, Step 5.1.33, dated 8/31/02

CARD 03-17687; Loose Wire in H11 P914 Causes P5000 F440 to Close and Start Division 1 CAC, dated 5/18/03

CARD 02-01539; NIAS Division 2 Dryer Malfunction, dated 8/13/02

Maintenance Rule Functional Failure Evaluation, System P5001; A valve on the East IAS Dryer Failed Open Causing Continuous Dryer Blowdown, Event Date 3/5/03

Maintenance Rule Functional Failure Evaluation, System P5001; While Starting the West Station Air Compressor for PMT of Several Jobs, The Inboard Motor Bearing Starting Sparking, Glowing, and Smoking, Event Date 3/13/03

Maintenance Rule Functional Failure Evaluation, System P5001; During PMT for P142010100, the CSAC Would not Maintain Header Pressure Above 95 psig, Event Date 2/21/02

Maintenance Rule Functional Failure Evaluation, System P5001; During PMT for P377970404, Four Attempts Were Made to Open P5000F403 and Were Unsuccessful; Event Date 1/15/02

Maintenance Rule Functional Failure Evaluation, System P5001; During Performance of 24.129.01, Valve P5000F403 Would not Stroke Open, Event Date 12/1/01

Maintenance Rule Functional Failure Evaluation, System P5001; Valve P500F018B Leaks by the Seat, Event Date 2/26/02

Maintenance Rule Functional Failure Evaluation, System P5001; During PMT for P330010100, the ESAC Would not Maintain Header Pressure, Event Date 2/20/02

1R13 Maintenance Risk Assessment and Emergent Work

PBF-19, Appendix I; Integrated Work Management Guideline, Outage Nuclear Safety; Rev. 2

RF-09 Defense in Depth Summary; dated May 1 through May 3, 2002

CARD 03-16396, “Blower peek hole clearance on EDG #12 shows an adverse trend,” dated 6/3/03

CARD 02-19674, “Evaluation of elevated corrosion rates,” dated 9/24/02

Letter from Dr. Joram Lichtenstein, P.E. of Washington Group International to George Piccard of DTE Energy, “Detroit Edison, Enrico Fermi 2, Pump Corrosion,” dated 5/5/03

Letter from Mr. Girija Shukla of Detroit Edison to NRC, “Implementation of Commitments Made in Response to Generic Letter 89-13,” dated 7/29/91

Letter from Mr. Girija Shukla of Detroit Edison to NRC, “Response to NRC Generic Letter 89-13,” dated 1/26/90

GE Betz Inorganic Analysis Report, Laboratory ID No. 81679.1, dated 4/29/03

GE Betz Metallurgical Lab Report, Reference No. 2003-0332, dated 5/12/03

Structural Integrity Associates Report No. SIR-03-058, Rev. 0, “Metallurgical Evaluation of RHRSW Pump Bolts,” dated 5/20/03

CARD 03-14451, “Corroded Bolts on RHRSW Pump A Column Flanges,” dated 4/5/03

CARD 03-01098, “EPA Circuit Breaker Malfunction,” dated 5/21/03

Dwg No. 6I721-2151-1, Rev. X, “Schematic Diagram Rx Prot Sys. Motor-Generator Set A,” dated 11/13/92

Dwg. No. 6I721-2151-2, Rev. U, “Schematic Diagram Rx Prot Sys. Motor-Generator Set B,” dated 11/13/92

CARD 03-01095, “Undervoltage light is on - breaker not tripped,” dated 5/20/03

CARD 03-14848, “Insulation removed online from RHR heat exchanger piping contrary to approved disposition of CARD 02-14782,” dated 3/19/03

Work Request No. A560030100, “Erect Scaffolds, reove/reinstall insulation and perform cleaning of ISI welds in various locations in the drywell,“ dated 5/6/03

Selected operator logs March 5, 2003 - March 19, 2003

Risk Evaluation, “Insulation removed with HELB or LOCA Probability,” dated 9/5/01

Dwg. No. 6M721-5706-1, Rev. Y, “Residual Heat Removal (RHR) Division II Functional Operating System,” dated 11/27/02

1R15 Operbility Evaluations

CARD 03-14450, “Water accumulation in torus downcomer to vent header tee connections,” dated 4/6/03

1R16 Operator Workarounds

Nuclear Generation Memorandum NPOP-03-0024, “Aggregate Assessment of Operator Work Arounds,” dated May 5, 2003

Nuclear Generation Memorandum TMSA-03-0032, “Risk Assessment of Revised Operator Work Arounds - May 2003,” dated May 5, 2003

1R17 Permanent Plant Modifications

CARD 03-11200, “Narrow and Wide Range Reactor Pressure on C32-R609, not Reading as Expected,” dated 2/21/03

1R19 Post Maintenance Testing

CARD 03-01130, “During Core Spray Run with Flow through E2150-F031B, 140 DPM Leak was Noted Coming from Packing Gland,” dated 04/22/03

Plant Technical Procedure 47.306.03, “MOV Motor Testing with Motor Power Monitoring System,” Revision 4

Plant Technical Procedure 35.000.216, “Valve Repacking,” Revision 29

Plant Technical Procedure 43.203.005, “Div 2 CSS Leakage Monitoring Test,” Revision 26

Plant Technical Procedure 24.203.03, “Division 2 CSS Pump and Valve Operability, and Automatic Actuation,” Revision 42

Work Request 000Z013739, “Repack Division 2 Core Spray Pump Minimum Flow Valve”

Technical Specification 5.5.2, “Primary Coolant Sources Outside Containment

UFSAR 9.3.1, “Compressed Air System”

Memorandum from Mark Adams, Risk Assessment, to Bob Slottke, Work Week Manager, “Risk Assessment for the Week of May 26, 2003,” dated 5/23/03.

Work Request No. P021020100, “Replace Div. 1 NIAS supply isolation valve solenoid valve,” dated 2/24/03

CARD 03-17874, “Inadequate PMT for PM P021020100 - NRC Concern,” dated 6/2/03

Job ID No. AB01030731, “Perform 27.129.01 Div 1 (North) Control Air Compressor Auto Start Test,” dated 5/21/03

CARD 03-17887, “Request logic functional test be created for P5000F402,“ dated 6/9/03

Dwg. No. 6M721-5730-1, Rev. AF, “Station Air System (Turbine Bldg.) Functional Operating Sketch,” dated 4/26/03

Dwg. No. 6M721-2015, Rev. BN, “Diagram Station and Control Air,” dated 4/26/03

Dwg. No. 6M721-5730-1, Rev. AF, “Non-Interruptible Control Air Sys Division I & II Functional Operating Sketch,” dated 12/27/01

Work Request No. 000Z013986, “Revise the N CAC S/G Green Band per RID-71406,” dated 10/14/02

Work Request No. Q333030100, “Sample Crankcase Oil, Inspect Belts, Lube Motor Bearings, and Clean Exterior,” dated 3/6/03

CARD 02-12806, “Install better EDG air receiver drain valves,” dated 7/10/02

CARD 03-17330, “R3000F041B, East starting air receiver drain valve, failed PMT,” dated 6/12/03

RID-71880, “EDG Receiver Drain Valve R3000F040B and R3000F041B,” dated 6/5/03

Design Calc. No. 3226, Rev. 0, “Enrico Fermi Atomic Power Plant Unit 2, 1E Equipment Qualification Review, System: G11,” dated 11/12/85

Work Request No. G142950926, “Replace DW Floor Drain Sump Solenoid Valve Per NE-6.6-EQMS.031,” dated 3/6/03

Specification No. 3071-198, Rev. E, “Wall and Floor Penetration Seals and Fire Stops, Enrico Fermi Power Plant Unit 2”

Dwg. No. 6M721-5710-2, Rev. Y, “Sump Pumps System Functional Operating Sketch,” dated 10/9/98

Dwg. No. 6M721-5710-4, Rev. M, “Floor Drain Collector Diagram Radwaste System Functional Operating Sketch,” dated 7/3/01

Raychem Heat Shrink Tubing Insulation Design and Installation Information for Work Request No. G142950926 and PIS No. G11F408, dated 3/19/03

Plant Technical Procedure 35.CON.022, Rev. 29, “New Cable Terminations”

Plant Technical Procedure 24.702.01, Rev. 32, “Miscellaneous Systems Valve Operability Test”

CARD 03-11287, “Required action of Tech Spec 3.6.1.3 condition A not taken within allowable completion time,” dated 3/20/03

Selected operator logs March 19, 2003 - March 20, 2003

Revision of Maintenance and Surveillance letter NE-6.6-EQMS.031, Rev. 10, “Environmental Qualification maintenance & Surveillance Requirement for EQ1-EF2-052A for Asco Solenoid NP Series 8320-Suffix E Solenoid Valves,” dated 1/8/03

UFSAR 3.11.2.2, “Qualification Tests”

UFSAR 7.1.1.3.4, “Specific Regulatory Requirements”

Selected CECO information on PIS No. G11F408

Plant Technical Procedure 35.000.214, Rev. 23, “Installation of Electrical Equipment Moisture Seals”

Safety Tagging Record Archive Report for STR No. 2003-001154

Work Request No. V298960311, “Refurbish 480V Breaker 72A-3D,” dated 11/15/99

CARD 02-16323, “Abnormal indication on shutdown,” dated 10/22/02

CARD 03-00203, “Bearing Failure,” dated 3/13/03

Work Request No. 0007023462, “Ammeter Oscillated Upon Air Compressor Shutdown (WSAC),” dated 2/21/03

Selected operator logs February 26, 2003 - February 27, 2003

CARD 98-11915, “Inadequate component preventive maintenance,” dated 3/20/98

CARD 98-11484, “PM’s not performed on West and Center Station Air Compressor,” dated 1/30/98

Work Request No. P147020100, “perform Annual maintenance on West Station Air Comrpessor,” dated 9/28/01

1R20 Refueling and Outages

TMRF-03-0047, Ltr Smith to Davis Confirmation of Compliance with Footnote 11 of RG 1.183, dated March 28, 2003

82.000.04, Refueling and Core Post-Alteration Verification, Rev. 35

MES32, Conduct of Fuel and Special Nuclear Material Management, Rev. 9

MOP13, Conduct of Refueling and Core Alterations, Rev. 8

EOOS Safety Function Assessment 4/16/2003-4/19/2003

CARD 03-16929, Cracked Tack Welds on Jet Pump No. 16 Restrainer Screw (Shroud Side), dated 4/20/03

Procedure 22.000.02, “Plant Startup to 25 Percent Power,” Revision 56

Procedure 22.000.01, “Plant Startup Master Checklist,” Revision 50

Gasket FME Emergent Issues Team Report; dated May 2, 2003

GE-NE-0000-0016-1045-R0; Lost Parts Analysis; Gasket Material in the Enrice Fermi 2 Reactor Pressure Vessel (RPV); Rev. 0

Root Cause Report, CARD 03-14451; RHR Reservoir Service Water Pumps’ Bolt and Column Degradation; dated April 30, 2003

Detroit Edison Memorandum from Peter Smith, Manager Nuclear Fuels to Jon Davis, Manager, Nuclear Outage management, “Confirmation of Compliance with Footnote 11 of RG 1.183,” dated 3/28/02

Fermi 2 - RF09 - EOOS Safety Function Assessment from 4/16/03 to 4/19/03

Plant Technical Procedure 82.000.04, Rev. 35, “Refueling and Core Post-Alteration Verification”

Fermi 2 Operations Conduct Manual MOP13, Rev. 8, “Conduct of Refueling and Core Alterations”

Fermi 2 Engineering Support Conduct manual MES32, Rev. 9, “Conduct of Fuel and Special Nuclear Material Management”

RF-09 Daily Status Reports March 29, 2003 - May 10, 2003

CARD 03-12859, “Investigate and evaluate LLRT results for G1100-F019,” dated 4/9/03

CARD 03-14497, “Valve measured thrust does not meet design calculation required thrust,” dated 4/10/03

CRD 03-16121, “Evaluate LLRT results for the primary containment pneumatics Div. 2 supply outboard isolation valve,” dated 4/10/03

CARD 03-16731, “Metal shaving observed on video of cell 26-35 on core plate S.W. of FSP & in cell 42-35,” dated 4/10/03

CARD 03-14985, “Pealing paint on nitrogen supply lines to drywell-to-torus vacuum breakers,” dated 4/10/03

CARD 03-16014, “Charred field wires in MOV,” dated 4/8/03

CARD 03-12917, “VOTES and MPM tests indicate limit switch settings need adjusted,” dated 4/11/03

CARD 03-14871, “HP3 stop vlv pilot poppet seat indications,” dated 4/11/03

CARD 03-00372, “Water trap float valve sticking,” dated 4/11/03

CARD 03-11390, “Receiving Div. 1 CAC trouble (Alarm 7D51) for dryer malfunction when dryer shifts alarm resets and then comes back,” dated 4/11/03

CARD 03-16122, “Investigate and evaluate LLRT test results for the stem test on T4800F404, F405, and F409,” dated 4/10/03

Outage Shift Manager Logs March 28, 2003 - May 4, 2003

RF09 Refuel Outage Overview, dated 5/15/03

1R22 Surveillance Testing

24.137.21; Reactor Pressure Vessel System Leakage Test; Rev. 8

TS 3.10.1; Inservice Leak and Hydrostatic Testing Operation, Amendment No. 134

24.307.14; Emergency Diesel Generator 11 - Start and Load Test, Rev. 46

Detroit Edison Memorandum from Patrick Fallon, Critique Leader IPTE 97-05 to DISTRIBUTION, “Critique of IPTE 97-05,” dated 10/28/97

CARD 03-11742, “NRC Questions regarding License Amendment 114, Addition of a Special Test Exception to Allow Inservice Leak and Hydrostatic Testing to be Performed at Temepratures Greater than 200 deg. F,” dated 7/9/03

Detroit Edison Licensing Change Request 97-1311-OPL

Job ID No. 0984980815, “Perform 24.137.021 Reactor Pressure Vessel System Leakage Test,” dated 9/21/98

Selected operator logs April 29, 2003 - May 4, 2003

Document Change Request 98-1825, “Reactor Pressure Vessel System Leakage Test,” dated 8/19/98

Plant Technical Procedure 43.000.009, Rev. 33, “Reactor Pressure Vessel System Leakage Test”

Selected operator logs October 20, 1998

Hydrostatic/System Leakage Test Temperature/pressure Data Sheet, Procedure No. 22.000.05, Attachment 6, dated 4/29/03

UFSAR 14.1.3.2.69, “Reactor System Hydrostatic Preoperational Test”

UFSAR 5.6.2.1, “Reactor Pressure Vessel Temperature”

UFSAR 6.2.1.2.1.9, “Access for Refueling Operations”

UFSAR 6.2.1.2.1.10, “Venting and Vacuum Relief System”

UFSAR 6.2.1.3.5, “Small Breaks”

UFSAR 7.6.1.2.3.1, “Reactor Pressure Vessel Temperature”

UFSAR 4.5.2.2.4.5, “Surveillance Tests”

UFSAR 5.2.4.2.4, “Temperature Limits for Inservice Inspection Hydrostatic or Leak Pressure Tests”

Plant Technical Procedure 24.137.21, Rev. 9, “Reactor pressure Vessel System Leakage Test”

Technical Specification No. 3.10.1, “Inservice Leak and Hydrostatic Testing Operation”

CARD 03-12803, “Discrepancy between Tech Spec LCO 3.10.1 and RPV leakage test procedure 24.137.21,” dated 5/1/03

ASME Code Case No. N-416-1, “Alternative Pressure Test Requirement for Welded Repairs or Installation of Replacement Items by Welding, Class 1, 2, and 3 Section XI, Division 1,” dated 2/15/94

ASME Code Case No. N-498-1, “Alternative Rules for 10-Year System Hydrostatic Testing for Class 1, 2, and 3 Systems Section XI, Division 1,” dated 5/11/94

Selected operator logs April 29, 2003 - May 5, 2003

3PP2 Access Control

Security Loggable Event Reports for January 1 - May 15, 2003

Security Related Condition Reports for January 1 - May 9, 2003

Security Equipment Maintenance Log Book, January 1 - May 20, 2003

SEP-SE-1-01; Testing and Maintenance, Revision 20; April 3, 2003

Sentex Technology Inc. Scanex 1 Explosive Detector Vendor Manual

3PP3 Response to Contingency Events

Nuclear Security Drill Worksheets (Table Top Drills) January 3,7,10,11, 2003 and March 7-14, 2003

Security-Related Condition Report Listing for January 1 - May 9, 2003

Force-on-Force Drill Evaluations for February 5-6 and May 15, 2003

4OA1 Performance Indicator Verification

Camera and Microwave Compensatory Measure Tracking; October 2002 and March 2003

SDI-20; Security Performance Indicator Tracking and Reporting; Revision 4

Compensatory Measure Summary Reports, October 2002 and March 2003

Security Equipment Verification Report; October 2002 and March 2003

Security Loggable Events for January 1 - May 15, 2003

4OA2 Identification and Resolution of Problems

2OS1 Access Control to Radiologically Significant Areas

CARD 03-12618; West MSR HEPA Found Running with No Filters Installed; dated April 4, 2003

CARD 03-14199; Three Workers Externally and Internally Contaminated Working on E1100F050A valve; dated April 1, 2003

CARD 03-14780; Worker Entered Drywell on Wrong Task for Work. The Right RWP was Used; dated April 2, 2003

CARD 03-14792; Improper Rad Worker Practice - Failed to Meet Dress Requirements of RWP; dated April 8, 2003

CARD 03-16243; Personnel Entered Drywell Under Wrong RWP; dated March 31, 2003

CARD 03-16843; GE Personnel Contaminations Associated with Undervessel Work; dated April 10, 2003

Plant Technical Procedure 67.000.400; Personnel Decontamination and Assessment; Revision 13

RWP 03-1009; Fermi 2 Personnel - Perform Pre-job Walkdowns, Inspections, and Supervisory Tours; Revision 0

RWP 03-1106; Walkdowns and Inspections in Drywell and RB-1 Steam Tunnel; Revision 0

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

RF09 Dose By Day Spreadsheets; dated April 7 through April 10, 2003

Audit Report 02-0112; Nuclear Quality Assurance Audit Report - Radiation Protection, Radioactive Effluent Monitoring, Radiological Material, Transfer and Disposal, and Non-Radiological Environmental Protection Programs; dated October 7 through November 25, 2002

CARD 03-14200; Obstacles Encountered Implementing the E11-F050A Hydrolazing Work Plan Increased Station Dose; dated April 3, 2003

Plant Technical Procedure 43.606.001; Traversing In-Core Probe Shear Valve Explosive Charge Test and Replacement; Revision 28

Plant Technical Procedure 63.000.200; ALARA Reviews; Revision 14

RWP/ALARA Plan No. 03-1103; Drywell - Install and Remove Temporary Shielding, Install Permanent Shielding (EDP 32051); Revision 0

RWP/ALARA Plan No. 03-1105; Install and Remove Scaffold, Power, and Lights in Drywell and Steam Tunnel; Revision 0

RWP/ALARA Plan No. 03-1110; Insulation Removal, Repair, and Replacement; Revision 0

RWP/ALARA Plan No. 03-1112; ISI Inspections. Work to Include Welds and Weld Preps, Snubbers and Spring Cans, Bioshield Doors, and All Other Associated Tasks; Revision 0

RWP/ALARA Plan No. 03-1117; CRD Exchange - To Include All CRD Drywell and Bullpen Work, Excluding Pre-outage, Filter Removal and Rebuild Associated Evolutions; Revision 0

RWP/ALARA Plan No. 03-1127; E11FO50A and E11FO50B - Perform Hardseat Modification Install New Discs; Revision 0

RWP/ALARA Plan No. 03-1127; E11FO50A and E11FO50B - Perform Softseat Replacement in Lieu of Hardseat Modifications; Revision 1

RWP/ALARA Plan No. 03-1251; Perform Refuel Activities on RB-5. Includes Vessel Assembly and Dissassembly, Core Alterations, ISI Work, Bridge Repair, LPRM Replacement, RP and Radwaste Support of All Activities; Revision 0

RWP 03-1152; TIP Room - Initial Entry, Explosive Valve Testing and Rework, Containment Verification, Drywell Inspection, ARM Calibration, Fire Detector Test; Revision 0

RWP 03-1207; Main Unit Turbine Components Including HP/LP Stop, Throttle and Bypass Valve Maintenance and East/West MSR Inspections; Revision 1

2PS3 Radioactive Material Control Program

CARD 03-00690; Individuals Inprocessing Were Identified with Externally Contaminated Personal Items; dated March 21, 2003

NPRC-03-0081; Notifications for Positive Whole Body Count Results; dated March 26, 2003

NPRC-03-0088; Whole Body Counting Protocol; dated April 1, 2003

Plant Technical Procedure 65.000.265; Maintenance and Operation of the Fermi 2 Whole Body Counters Using Renaissance Software; Revision 2

4OA3 Event Followup

CARD 03-17206; Loss of SDC Cooling due to Operator Error; dated April 28, 2003

Selected operator logs April 28, 2003

23.316, “RPS 120V AC and RPS MG Sets, Revision 42, dated September 30, 2002

Detroit Edison LER No. 02-005, “Non-Conservative Setpoint for Stability Option III (OPRM) Period based Algorithm - Tmin,” dated 1/16/03

CARD 02-18019, “Potential Part 21 reportable condition: Non-conservative Tmin setpoint in OPRM system,” dated 11/21/02

Dwg. No. 6M721-5739-2, Rev. O, “Hydrogen Recombiner System Functional Operating Sketch,” dated 4/23/03

Dwg. No. 6M721-2087, “Diagram (Post LOCA) Combustible Gas Control System,” dated 4/23/03

CARD 00-18919, “Temperature limit not reached in required time,” dated 8/2/00.

CARD 98-22312, “Div. 1 thermal recombiner fails to come to temperature in the required time,” dated 11/4/98.

DER 96-1849, “Div. 1 Thermal Recombiner Surveillance Failure,” dated 12/18/96

CARD 01-15519, “Over 300 ml water off of the recombiner during surveillance 24.409.01,” dated 5/13/01

CARD 02-16354, “Div. TRS failed surveillance,” dated 11/2/02

CARD 02-19137, “Inadequate system restoration,” dated 11/11/02

CARD 01-14825, “PST event to drain & check for leakage from T4804F001 cause increased wear on piping components,” dated 5/16/01

DER 96-0188, “Div. 2 thermal recombiner surveillance failure,” dated 2/23/96

CARD 97-13364, “Div. I thermal recombiner thermocouples not indicating,” dated 11/5/97

CARD 98-15179, “Potential improper testing of thermal recombiner heaters,” dated 7/20/98.

CARD 02-14275, “Acceptance criteria was not met,” dated 5/2/02

Selected operator logs May 30, 2003 - June 2, 2003