May 12, 2005

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

SUBJECT: FERMI POWER PLANT, UNIT 2, NRC INTEGRATED INSPECTION REPORT 05000341/2005004 and 05000341/2005005

Dear Mr. O'Connor:

On March 31, 2005, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Fermi Power Plant, Unit 2. The enclosed report documents the inspection findings which were discussed on April 7, 2005, with you and other members of your staff.

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

Based on the results of this inspection, five findings of very low safety significance (Green) were identified, three of which involved violations of NRC requirements. However, because these findings were of very low safety significance and because the issues were entered into your corrective action program, the NRC is treating these violations as Non-Cited Violations in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

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

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

Sincerely,

/RA/

Robert M. Lerch, Acting Chief
Branch 6
Division of Reactor Projects

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

Enclosure: Inspection Report Nos. 05000341/2005004 and 05000341/2005005
w/Attachment: Supplemental Information

cc w/encl: N. Peterson, Manager, Nuclear Licensing
D. Pettinari, Legal Department
Compliance Supervisor
G. White, Michigan Public Service Commission
Michigan Department of Environmental Quality
Monroe County, Emergency Management Division
Planning Manager, Emergency Management Division
MI Department of State Police

DOCUMENT NAME: E:\Filenet\ML051320379.wpd
To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	RIII	RIII		
NAME	DJones:dtp	RLerch		
DATE	05/12/05	05/12/05		

OFFICIAL RECORD COPY

ADAMS Distribution:
LXR1
NKR
RidsNrrDipmIipb
GEG
KGO
SJC4
CAA1
C. Pederson, DRS (hard copy - IR's only)
DRPIII
DRSIII
PLB1
JRK1
ROPreports@nrc.gov (inspection reports, final SDP letters, any letter with an IR number)

U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-341
License No:	DPR-43
Report Nos:	05000341/2005004 and 05000341/2005005
Licensee:	Detroit Edison Company
Facility:	Fermi Power Plant, Unit 2
Location:	6400 N. Dixie Hwy. Newport, MI 48166
Dates:	January 1 through March 31, 2005
Inspectors:	S. Campbell, Senior Resident Inspector T. Steadham, Resident Inspector W. Slawinski, Senior Radiation Specialist
Observers:	M. Franke, Resident Inspector, Perry M. Salter-Williams, Resident Inspector, Davis-Besse
Approved by:	R. Lerch, Acting Chief Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IRs 05000341/2005004 and 05000341/2005005; 01/01/2005-03/31/2005; Fermi Power Plant, Unit 2; Maintenance Risk Assessments and Emergent Work Evaluation, Operability Evaluations, Post Maintenance Testing, Radioactive Material Processing and Transportation, Event Followup.

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

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

Cornerstone: Initiating Events

- Green. A finding of very low safety significance was self-revealed on February 4, 2005, when the "D" residual heat removal (RHR) pump tripped on high suction pressure while attempting to initiate shutdown cooling. The primary cause of this finding was related to the cross-cutting area of Human Performance because operators failed to follow procedures for placing shutdown cooling in service. The issue was a Non-Cited Violation of Technical Specification 5.4.1.a for the failure to follow procedures.

  This finding was more than minor because it can be reasonably viewed as a precursor to a more significant event and it was associated with the Human Performance attribute of ensuring the availability, reliability, and capability of RHR to respond to initiating events. The finding was of very low safety significance because the inspectors answered "no" to all five screening questions for the Mitigating Systems Cornerstone. Initial corrective actions consisted of an RHR system walkdown to verify the system integrity followed by an operability evaluation. (Section 1R15.2).

- Green. The inspectors identified a finding of very low safety significance for the failure to perform an adequate post maintenance test after the licensee replaced the three main generator excitation power converter modules during refueling outage 10. The failure to perform an adequate test contributed to an automatic reactor shutdown when the automatic voltage regulator tripped.

  The finding was determined to be more than minor because it affected the Initiating Events Cornerstone in that it ultimately contributed to a reactor scram. The finding was of very low safety significance because it did not contribute to the likelihood of a primary or secondary system loss of cooling accident initiator, that mitigation equipment or functions would be unavailable, or increase the likelihood of a fire or internal/external flood. No violations of regulatory requirements were identified. Immediate corrective actions consisted of thoroughly troubleshooting the automatic voltage regulator and related modules as well as initiating a root cause analysis. (Section 1R19.2)

- Green. A finding of very low safety significance was self-revealed when design control measures were not implemented during plant construction while changing the drywell cooler gasket design from a wide to a narrow gasket. Consequently, wide instead of narrow gaskets were employed, and sufficient torque values used to secure the cooler end bells to the cooler tube sheet were not translated to maintenance procedures to ensure reassembly of a leak tight joint. As a result, the gasket on drywell cooler 4 failed and caused unidentified leakage to increase above 5 gallons per minute resulting in a plant scram and an alert declaration.

  The finding was determined to be more than minor because it affected the Initiating Events Cornerstone in that operators manually shutdown the reactor in response to an apparent increase in unidentified leakage caused by the leak from the failed gasket. The finding was of very low safety significance because it did not contribute to the likelihood of a primary or secondary system loss of cooling accident initiator, that mitigation equipment or functions would be unavailable, or increase the likelihood of a fire or internal/external flood. A Non-Cited Violation of 10 CFR 50, Appendix B, Criterion III, “Design Control,” was identified. Immediate corrective actions included replacing the wide gasket with the appropriate narrow gasket and torque the end bell bolts sufficiently for a leak tight joint. (Section 4OA3.1).

Cornerstone: Mitigating Systems

- Green. A finding of very low safety significance for the failure to properly connect a test lead was self-revealed during an event. An electrician inadvertently caused a short in the essential trip logic of emergency diesel generator (EDG)-14 which rendered it unavailable an inoperable. No violation of regulatory requirements was identified. The primary cause of the finding was related to the cross-cutting area of Human Performance.

  The finding was determined to be more than minor because it was associated with the Human Performance attribute of ensuring the availability, reliability, and capability of EDG-14 to respond to initiating events. The finding was of very low safety significance because all other EDGs remained operable and the actual loss of safety function of EDG-14 was shorter than its technical specification allowed outage time of 7 days. No violations of regulatory requirements were identified. Immediate corrective actions consisted of troubleshooting the effects of the short and replacing a damaged fuse. (Section 1R13.2)

Cornerstone: Public Radiation Safety

- Green. The inspectors identified a finding of very low safety significance and an associated violation of NRC requirements for the failure to perform complete radiation surveys of an outbound shipment of radioactive material to demonstrate compliance with Department of Transportation limits for the applicable mode of transport.

This issue 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 that ensures adequate protection of public health and safety from exposure to radioactive materials transported into the public domain. The issue represented a finding of very low safety significance because shipment package and vehicle radiation limits were not exceeded, no package breach occurred during transit, nor were certificate of compliance or low level burial ground issues applicable, and shipment notification and emergency information requirements were met. A Non-Cited Violation of 10 CFR 71.5(a) was identified for the failure to complete transport vehicle surveys applicable to the mode of transport as required by

49 CFR 173.475. Immediate corrective actions included postponement of further radioactive material shipments until tailgate training was provided to those staff that perform shipment radiation surveys. The primary cause of the finding was related to the cross-cutting area of Human Performance. (Section 2PS2.4)

B. Licensee-Identified Violations

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

REPORT DETAILS

Summary of Plant Status

Unit 2 began the inspection period at full power where it remained until January 24, 2005, when the reactor was manually shut down due to an increase in the unidentified drywell leakage rate. Repairs were completed and reactor startup commenced on February 2, 2005, at 5:10 p.m. The reactor was declared critical that night at 10:42 p.m. The reactor was manually shut down from approximately 12 percent power at 5:34 p.m. February 3, 2005, to fix a valve packing leak. On February 5, 2005, after repairs were completed, startup began at 4:55 p.m. with criticality being achieved at 10:54 p.m. The reactor reached full power on February 7, 2005, where it remained at or near for the duration of the inspection period.

1. REACTOR SAFETY

Cornerstone: Mitigating Systems

1R01 Adverse Weather (71111.01)

a. Inspection Scope

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

These activities represented one inspection sample.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments (71111.04)

.1 Partial System Walkdown (71111.04Q)

a. Inspection Scope

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

- standby liquid control performed on January 19, 2005;
- standby feedwater performed on January 21, 2005;

C residual heat removal (RHR), Division 2, performed on February 11 through February 22, 2005; and

C emergency diesel generator (EDG)-14 performed on February 22, 2005.

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

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

These activities represented four inspection samples.

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown (71111.04S)

a. Inspection Scope

The inspectors performed one complete system walkdown of the following risk significant system:

- condensate system performed January 31 through February 11, 2005.

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

These activities represented one inspection sample.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Fire Protection Walkdowns (71111.05Q)

a. Inspection Scope

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

- reactor building fourth floor;
- reactor building third floor;
- reactor building second floor;
- halon systems for the relay room, cable spreading room, computer room, and other areas;
- reactor building first floor; and
- turbine building third floor.

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

These activities represented six inspection samples.

b. Findings

No findings of significance were identified

1R06 Flood Protection (71111.06)

a. Inspection Scope

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

The inspectors also reviewed the associated work activities to verify identified problems were being entered into the corrective action program with the appropriate characterization and significance.

These activities represented one inspection sample.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

On February 15, 2005, the inspectors observed operations crew number 3 during the annual requalification examination in mitigating the consequences of events in scenario SS-OP-904-0209, “Reactor pressure vessel level instrument fails, reactor recirculation pump fails, small loss of coolant accident, partial failure of emergency core cooling system,” on the simulator. The inspectors evaluated the following areas:

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

The crew’s performance in these areas was compared to pre-established operator action expectations and successful critical task completion requirements.

These activities represented one quarterly testing/training inspection sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12Q)

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following risk-significant systems:

- RHR service water system; and
- RHR mechanical draft cooling towers.

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

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

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

These activities represented two inspection samples.

b. <u>Findings</u>

No findings of significance were identified.

1R13 <u>Maintenance Risk Assessments and Emergent Work Evaluation</u> (71111.13Q)

.1 <u>Routine Review of Maintenance Activities</u>

a. <u>Inspection Scope</u>

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

- EDG-14 safety system outage;
- reactor core isolation cooling system safety system outage;
- division 1 control center heating ventilation and air conditioning safety system outage;

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

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

These activities represented three inspection samples.

b. <u>Findings</u>

No findings of significance were identified.

.2 Loss of Essential Trip Function for Emergency Diesel Generator 14

a. Inspection Scope

The inspectors reviewed the events and circumstances surrounding the troubleshooting activities on February 26, 2005, when an electrician inadvertently caused a short and lost power to the essential trip logic strings for EDG-14. The inspectors reviewed documents and interviewed personnel.

These activities represented one inspection sample.

b. Findings

Introduction

A finding of very low safety significance (Green) was self-revealed when an electrician improperly connected test equipment that caused a short and rendered EDG-14 inoperable and unavailable. No violations of regulatory requirements was identified.

Description

On February 26, 2005, electricians were performing troubleshooting activities under WR 000Z050680 to locate the source of a DC ground on the 2PB neutral line. The ground was traced to panel R30-P341, "EDG Relay Box." The electricians reviewed the circuit and correctly identified the point where the test equipment, called a DC Scout, was to be connected at fuse FU-10. An independent verifier confirmed the connection point.

The licensee’s expectation was to connect the test lead to the fuse clip lug not the fuse clip itself; however, because troubleshooting electrical circuits was considered to be skill of the craft, this expectation was not proceduralized. The electrician who attempted to connect the test lead to FU-10 believed that the most secure location would be to connect the lead to the fuse clip. When he opened the lead clip to connect it to the fuse clip, one end of the lead clip contacted a nearby fuse and caused a short to FU-10.

The short caused a loss of power to panel R30-P341 when another fuse in the circuit feeding the panel opened. As a result, all DC power to the essential trip strings was lost thereby rendering EDG-14 unavailable. The electricians stopped work and consulted with operators. The licensee investigated the extent of condition and discovered the open fuse. The fuse was replaced and the EDG logic was reset approximately 2 hours after the event. Troubleshooting on the ground continued without any additional problems. The licensee entered this event into their corrective action program as CARD 05-21300.

Analysis

The inspectors determined that the failure to properly connect the test equipment was a performance deficiency warranting a significance determination. The inspectors concluded the finding was more than minor in accordance with IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Disposition Screening,” because it was associated with the Human Performance attribute of ensuring the availability, reliability, and capability of EDG-14 to respond to initiating events.

The inspectors completed a significance determination of this issue using IMC 0609, “Significance Determination Process, (SDP)” Appendix A, Attachment 1, “SDP Phase 1 Screening Worksheet for IE [Initiating Events], MS [Mitigating Systems], and B [Barriers] Cornerstones.” The inspectors concluded this finding affected the MS Cornerstone and answered, “NO,” to all relevant questions. Specifically, all other EDGs remained operable and the actual loss of safety function for EDG-14 was shorter than its TS allowed outage time of 7 days. Therefore, this finding was considered to be of very low safety significance (Green).

Enforcement

Because this activity was considered to be skill of the craft, the inspectors concluded that detailed instructions on lead placement were unnecessary. As such, this finding was not indicative of a specific licensee weakness in either procedure adherence or adequacy; therefore, no violation of regulatory requirements occurred. The licensee’s immediate corrective actions consisted of troubleshooting the effects of the electrical short and replacing a damaged fuse. This issue was considered a finding of very low safety significance (FIN 05000341/2005004-01.)

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

a. Inspection Scope

The inspectors reviewed operator performance in coping with the events and circumstances surrounding the February 4, 2005, trip of Division 2 RHR pump D due to a pressure spike in the pump discharge and a corresponding closure of the suction valves. The inspectors reviewed operator logs and plant computer data and interviewed operators and engineers to determine what occurred, how the operators responded, and if operator response was in accordance with both the relevant procedures and training.

These activities represented one inspection sample.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

.1 Routine Operability Evaluation Review

a. Inspection Scope

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

D. CARD 05-21282 to evaluate the use of calcium magnesium acetate on the roof of the RHR complex; and

E. CARD 05-21700 to evaluate the RHR system-to-torus leakage due to degraded RHR pump discharge check valves.

These activities represented two inspection activities.

b. Findings

No findings of significance were identified.

.2 Residual Heat Removal Pressure Transient Due To Valve Misalignment

a. Inspection Scope

The inspectors reviewed the events and circumstances related to the residual heat removal (RHR) pump “D” trip on February 4, 2005. The inspectors performed their review to determine if the condition rendered the involved equipment inoperable or resulted in an unrecognized increase in plant risk. The inspectors also ensured the licensee appropriately applied TS limitations and appropriately returned the affected equipment to an operable status. The inspectors interviewed engineering and operations personnel, reviewed various documents and drawings, and received assistance from both regional and headquarters specialists.

These activities represented one inspection sample.

b. Findings

Introduction

A Green finding was self-revealed on February 4, 2005, when the RHR “D” pump tripped during a pressure transient when operators attempted to start the pump in shutdown cooling (SDC). A non-cited violation (NCV) of TSs for failing to follow approved station procedures for placing the RHR system in the SDC mode of operation was identified.

Description

As a result of a drywell cooler leak on January 24, 2005, operators manually shutdown the plant. Several hours later on January 25, 2005, operators began preparations for placing Division 2 RHR into the SDC mode of operation in accordance with Section 5.3 of Procedure 23.205, “Residual Heat Removal.” Operators selected the Division 2 “D” RHR pump to be placed in service. Step 5.3.2.1 directed operators to close the torus suction isolation valves and open the SDC suction isolation valves for both the “B” and “D” pumps. When performing this step, operators did not perform the “B” pump re-alignment because they mistakenly believed doing so was not applicable to placing the “D” pump in service. As a result, the “B” pump suction remained lined-up to the torus when operators performed Section 7.2 to fill and vent the Division 2 RHR system.

When Steps 7.2.2.3.a and 7.2.2.3.b were performed to open the bypass valves around the "B" pump discharge check valve, operators received the RHR Divisions 1 and 2 keep-fill low pressure alarms, the condensate low pressure alarm, and indications that condensate flow had increased by approximately 700 gallons per minute (gpm). These alarms were received because the mis-alignment caused a flow path to drain the RHR and keep-fill systems to the torus. As the water level dropped in the condensate surge tank, which supplied the keep-fill system, the air pocket expanded and eventually made its way to the RHR piping via the keep fill lines. Because the RHR Division 1 and Division 2 cross-tie valve was open at the time, these conditions affected both RHR divisions. The following step, Step 7.2.2.3.c, directed the operators to vent the high-point RHR SDC suction line. Water draining to the torus created a slight vacuum in the system called an eductor effect. After operators noticed that air was being sucked into the vent line, they isolated the vent but did not notify the control room.

The keep-fill and condensate pressure alarms were locked-in for approximately 20 minutes when the torus high level alarm came in. Operators then closed the "B" pump bypass valves and the SDC suction vent valves and took actions to lower the torus level to within normal limits. After reviewing drawings, operators discovered the misalignment and re-performed Sections 5.3 and 7.2 of Procedure 23.307 but did not re-perform the venting for either the low pressure coolant injection (LPCI) injection line or the heat exchanger as required by Steps 7.2.2.1 and 7.2.2.2. When the SDC suction line was vented, operators noted a large amount of air.

Step 5.3.2.16 required operators to perform post-warming venting on RHR Divisions 1 and 2; however, only Division 2 was vented. During this vent, a large amount of air was vented from the head spray line. When the "D" RHR pump was started, reactor water level dropped seven inches which operators mistakenly believed to be a normal fluctuation. On January 30, 2005, during a routine vent check, air was found air in the head spray line. Division 2 RHR was later shutdown on February 2, 2005, to support plant start-up.

During start-up activities, the licensee discovered a steam leak from the "A" reactor recirculation pump discharge valve and decided to shut the plant down to facilitate repairs. On February 4, 2005, the "D" pump was again selected for SDC. While performing Step 5.3.2.16, operators again failed to vent Division 1; however, air was vented from the Division 2 drywell spray line.

The "D" RHR pump tripped approximately 3 seconds after it was started due to a pressure spike from the collapsing air bubble and the SDC suction valves closed on a high pressure isolation. Subsequent venting of Division 1 RHR yielded approximately 8 minutes of air from the LPCI injection line and 5 minutes of air from the drywell spray header. No additional air was noted from a re-vent of Division 2 including the pump casings for RHR pumps "B" and "D." Residual heat removal pump "B" was later successfully started in SDC mode on February 4, 2005, with no additional problems noted.

The licensee documented the loss of SDC in Condition Assessment Resolution Document (CARD) 05-20760 and formed an emergent issues team to investigate the event. Following the event, the inspectors performed walk-downs of the RHR piping and noted no damage to the piping or supports. The licensee completed a past operability evaluation which concluded that the LPCI mode of RHR remained operable between January 25, 2005, and February 4, 2005, with the air-voids present in the RHR system.

The inspectors reviewed the evaluation with the assistance of both regional and headquarters specialists and determined that although LPCI operability was justified, the licensee did not address the containment heat removal function of RHR which was more risk-significant than LPCI. The licensee re-evaluated this function and determined that it, too, remained operable.

The licensee discussed this event with site personnel to re-emphasize the importance of good operational decision making, human performance, and procedure adherence. Further, expectations to immediately inform the control room of any anomalies while venting were also emphasized.

Analysis

The inspectors determined that the multiple examples of operators failing to follow procedures was a performance deficiency warranting a significance determination. The inspectors concluded the finding was more than minor in accordance with Inspection Manual Chapter 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," because the finding can be reasonably viewed as a precursor to a more significant event. Additionally, the finding was associated with the Human Performance attribute of ensuring the availability, reliability, and capability of RHR to respond to initiating events.

The inspectors completed a significance determination of this issue using Inspection Manual Chapter 0609, "Significance Determination Process (SDP)," Appendix A, Attachment 1, "SDP Phase 1 Screening Worksheet for IE [Initiating Events], MS [Mitigating Systems], and B [Barriers] Cornerstones." The inspectors concluded that this finding affected the MS Cornerstone. Because the inspectors answered "no" to all five screening questions for the MS Cornerstone, this finding was considered to be of very low safety significance (Green).

Enforcement

Technical Specification 5.4.1.a required that written procedures be established, implemented and maintained in accordance with the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, dated February 1978. Regulatory Guide 1.33, Appendix A, Section 4, "Procedures for Startup, Operation, and Shutdown of Safety-Related BWR Systems," required, in part, instructions for energizing, filling, venting, startup, and changing modes of operation for shutdown cooling. Procedure 23.205 required, in part, the following in preparation of placing Division 2 RHR in SDC:

- Step 5.3.2.1 required operators to close the torus suction isolation valves and open the SDC suction isolation valves for both the “B” and the “D” RHR pumps. Contrary to this, on January 24, 2005, during initial preparations to place Division 2 RHR in SDC, operators failed to close the torus suction isolation valves and open the SDC suction isolation valves for the “B” RHR pump.

- Step 7.2.2.1 required operators to vent the Division 2 LPCI discharge piping. Contrary to this, on January 24, 2005, during the second attempt to fill and vent Division 2 RHR, operators failed to vent the Division 2 LPCI discharge piping.

- Step 7.2.2.2 required operators to vent the Division 2 RHR heat exchanger. Contrary to this, on January 24, 2005, during the second attempt to fill and vent Division 2 RHR, operators failed to vent the Division 2 RHR heat exchanger.

- Step 5.3.2.16 required operators to perform post-warming venting on both RHR divisions. Contrary to this, on both January 24, 2005, and February 4, 2005, operators failed to vent Division 1 RHR during post-warming venting.

However, because this violation was of very low safety significance and because it was entered into the licensee’s corrective action program, this violation is being treated as a Non-Cited Violation (NCV 05000341/2005004-02), consistent with Section VI.A of the NRC Enforcement Policy. The licensee entered this issue into their corrective action program as CARD 05-20760. Initial corrective actions consisted of an RHR system walkdown to verify the system integrity followed by an operability evaluation.

1R16 Operator Work-Arounds (71111.16)

a. Inspection Scope

The inspectors evaluated the risk assessment of revised operator work-arounds, letter TMSA-05-0007 dated January 21, 2005, to identify any potential effect on the functionality of mitigating systems or on the operators' response to initiating events.

The inspectors selected operator work-around for a leaking gland seal system bypass valve, N3013F603, to review as a potential operator work-around in order to understand how this task was accomplished and the potential effect on plant operations. The inspectors reviewed selected procedures and documents.

These activities represented one inspection sample.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

Engineering Design Package (EDP) 29238, for modification to the reactor recirculation pump "A" thermal barrier heat exchanger, was reviewed and selected aspects were discussed with engineering personnel. This document and related documentation were reviewed for adequacy of the safety evaluation, consideration of design parameters, implementation of the modification, post-modification testing, and that relevant procedures, design, and licensing documents were properly updated. The modification was for equipment upgrades of existing equipment.

These activities represented one inspection sample.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

.1 Routine Review of Post Maintenance Testing

a. Inspection Scope

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

- Procedure 24.415, Drywell Cooling Fan Test;
- Work Request (WR) 000A412060100, Post Maintenance Test Following Packing Replacement on B3105F031A;
- Procedure 24.307.48, EDG-14 Fast Start Followed by Load Reject; and
- Procedure 24.206.01, Revision 63, Reactor Core Isolation Cooling Pump and Valve Operability Test.

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

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

These activities represented three inspection activities.

b. Findings

No findings of significance were identified.

.2 Power Converter Module Post Maintenance Testing

a. Inspection Scope

The inspectors reviewed PMT activities associated with the replacement of three power converter modules for the main turbine generator excitation system and related auxiliaries. The inspectors reviewed the scope of the work performed and evaluated the adequacy of the specified PMT. The inspectors interviewed operations and engineering department personnel and reviewed the completed PMT documentation.

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

These activities represented one inspection activity.

b. Findings

Introduction

The inspectors identified a finding of very low safety significance (Green) for the failure to perform an adequate PMT after the licensee replaced the three main generator excitation power converter modules during refueling outage (RF)-10. Failure to perform an adequate PMT contributed to an automatic reactor shutdown when the automatic voltage regulator (AVR) tripped. This issue was considered NRC identified because the licensee failed to identify the inadequate PMT for over 5 months and did not identify it without the inspectors' questions. No violation of regulatory requirements was identified.

Description

On December 4, 2004, at 4:17 a.m. during start-up from RF-10, a turbine trip and reactor scram occurred from 57 percent reactor power. The turbine trip was due to an AVR failure because incompatible circuit boards were installed on the system during RF-10.

The licensee had experienced one other AVR failure on September 4, 2004. As a result of that failure, the licensee replaced the three main turbine generator exciter power converter modules, EG1, EG2, and EG3, with new modules as recommended by the AVR vendor. This work was performed under WR 000Z042819 and completed on November 21, 2004. Each module contained the following four boards:

- UNS 0670, power supply board;
- UNS 0671, thyristor pulse board;
- UNS 0672, ARCNet coupler board; and
- UNS 0673, pulse conduction monitoring board.

Module EG1 was supplied with all new components whereas EG2 and EG3 included the licensee's old UNS 0673 boards. Otherwise, both EG2 and EG3 included all new components. The vendor described all of the new components as being an upgrade from the licensee's existing hardware that would enhance the reliability of the overall system.

Because the modules were replaced during RF-10, the system was not operating as it would be during power operations. Consequently, the licensee's PMT scope consisted of re-energizing the modules, ensuring power was available to the module components, and the error indications on the modules could be reset. The inspectors reviewed the PMT and determined that at best, the PMT tested only the UNS 0670 and UNS 0672 boards. Because the generator was off-line at the time, neither the UNS 0671 nor the UNS 0673 boards were functionally tested during the PMT for any of the three modules.

The licensee then started the plant and synchronized the generator to the grid at 9:47 a.m. on December 3, 2004. At 11:00 a.m., operators began receiving control room annunciator 4D53, "AVR General Alarm," and initiated CARD 04-26422 to document the AVR error codes related to the alarm. Operators noted alarms indicative of a failure of the UNS 0671 and/or the UNS 0673 boards on several occasions along with other alarms indicative of more widespread degradation. At 4:17 a.m. the next day, the AVR tripped and caused a reactor scram. The licensee documented the scram in CARD 04-26443. Immediate corrective actions consisted of thoroughly troubleshooting the AVR and related modules as well as initiating a root cause analysis.

Subsequent investigation into the cause of the AVR trip determined that modules EG1 and EG2 experienced concurrent failures. The system was designed to trip if at least two of the three modules fail concurrently. Further investigation determined that the EG1 fault occurred due to either an intermittent fault or incompatibility with its UNS 0673 board whereas the EG2 fault was caused by either an intermittent fault or incompatibility with its UNS 0672 board.

The licensee later replaced UNS 0673 in the EG1 module with the old UNS 0673 board removed during RF-10 and all three UNS 0672 boards with those removed during RF-10. No further AVR alarms were received.

The inspectors determined that although the PMT performed on EG1, EG2, and EG3 for WR 000Z042819 did not adequately test the functionality and compatibility of the three modules, the licensee could have reasonably performed such a test after the generator was synchronized. The licensee would have then discovered a problem with two of the three modules prior to operators raising power above the reactor protection system turbine stop and control valve closure scram setpoint which could have avoided a reactor scram. The licensee entered this issue into their corrective action program as CARD 05-22341. The problem identification and resolution aspect of this event is discussed in Section 4OA2.

<u>Analysis</u>

The inspectors determined that the failure to perform a proper PMT on the replacement power converter modules was a performance deficiency warranting a significance determination. The inspectors concluded the finding was more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," because it was associated with the design control attribute of limiting the likelihood of events that upset plant stability. Specifically, the performance deficiency, a lack of PMT, did not identify an inappropriate modification that ultimately contributed to a reactor scram.

The inspectors completed a significance determination of this issue using IMC 0609, "Significance Determination Process, (SDP)" Appendix A, Attachment 1, "SDP Phase 1 Screening Worksheet for IE [Initiating Events], MS [Mitigating Systems], and B [Barriers] Cornerstones." The inspectors concluded this finding affected the IE Cornerstone. However, since the finding did not contribute to the likelihood of a primary or secondary system loss of coolant accident initiator, that mitigation equipment or functions would be unavailable or increase the likelihood of a fire or internal/external flood, this finding was considered to be of very low safety significance (Green).

Enforcement

Because the AVR is not safety-related, neither 10 CFR 50, Appendix B, nor Regulatory Guide 1.33, Appendix A, Section 9, apply. Therefore, no violation of regulatory requirements was identified and this was a Green finding (FIN 05000341/2005004-03). Immediate corrective actions consisted of thoroughly troubleshooting the automatic voltage regulator and related modules as well as initiating a root cause analysis.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors observed the licensee's performance during the January 24, 2005, forced outage due to a gasket failure and subsequent reactor building closed cooling water (RBCCW) leak from drywell cooler 4 followed by a manual reactor scram due to increased unidentified leakage.

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

The inspectors frequently performed the following activities during the forced outage:

- attended control room operator and outage management turnover meetings to verify the current shutdown risk status was well understood and communicated;
- performed walkdowns of the main control room to observe the alignment of systems important to shutdown risk;

- observed the operability of reactor coolant system instrumentation and compared channels and trains against one another; and

- performed walkdowns of the turbine, auxiliary, and reactor buildings to observe ongoing work activities to ensure work activities were performed in accordance with plant procedures, and to verify procedural requirements regarding fire protection, foreign material exclusion, and the storage of equipment near safety-related structures, systems, and components were maintained.

These activities represented one inspection sample.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

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

- reviewed video tapes of the as-found condition of the torus suction strainer;
- observed standby feedwater pump "A" operability test Procedure 24.107.03;
- reviewed source range monitor/intermediate range monitor dry tube inspection;
- observed sequence of events test 04-11, Emergency Equipment Cooling Water Backup Heat Exchanger D Test; and
- verified TS requirements satisfied for power increase.

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

These activities represented five inspection samples.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed one temporary modification and verified the installation was consistent with design modification documents and the modifications did not adversely impact system operability or availability.

- Temporary Modification 04-0033; “Modify West Station Air Compressor Lube Oil System.”

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

These activities represented one inspection sample.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed the licensee perform an emergency preparedness drill on March 23, 2005. The inspectors observed activities in the control room simulator and technical support center. The inspectors also attended the post-drill facility critiques in the technical support center and simulator immediately following the drill and the overall drill critique. The focus of the inspectors’ activities was to note any weaknesses and deficiencies in the drill performance and ensure the licensee evaluators noted the same weaknesses and deficiencies and entered them into the corrective action program. The inspectors placed emphasis on observations regarding event classification, notifications, protective action recommendations, and site evacuation and accountability activities. As part of the inspection, the inspectors reviewed:

C Scenario 37; Intrusion, Reactor Feed Pump Trip, Anticipated Transient Without Scram.

These activities represented one inspection sample.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Public Radiation Safety

2PS2 Radioactive Material Processing and Transportation (71122.02)

.1 Radioactive Waste System Description and Waste Generation

a. Inspection Scope

The inspectors reviewed the liquid and solid radioactive waste system descriptions in the UFSAR and the 2002 and 2003 Annual Radioactive Effluent Release Reports for information on the types and amounts of radioactive waste (radwaste) generated and disposed. The inspectors reviewed the scope of the licensee's audit/self-assessment activities with regard to radioactive material processing and transportation programs to determine if those activities satisfied the requirements of 10 CFR 20.1101(c).

These activities represented one inspection sample.

b. Findings

No findings of significance were identified.

.2 Radioactive Waste System Walkdowns

a. Inspection Scope

The inspectors walked down portions of the solid radwaste processing systems to verify these systems were consistent with the descriptions in the UFSAR and the Process Control Program and to assess the material condition and operability of the systems. The status of radioactive waste process equipment that was not operational or was abandoned in-place was reviewed, along with the licensee's administrative and/or physical controls, in order to ensure that the equipment would not contribute to an unmonitored release, adversely affect operating systems, or be a source of unnecessary personnel exposure.

The inspectors reviewed the licensee's processes for transferring waste resin into shipping containers (and for dewatering) to determine if appropriate waste stream mixing and sampling was performed so as to obtain representative waste stream samples for analysis. The inspectors reviewed the licensee's practices for the collection of area smear surveys to represent the dry-active waste (DAW) stream and the method used for determining the radionuclide mix of filter media to ensure they were representative of the intended radwaste stream. Additionally, the inspectors reviewed the methodologies for quantifying gamma emitting radionuclide waste stream content, for determining waste stream tritium concentrations, and for waste concentration averaging to ensure that representative samples of the waste products were provided for the purposes of waste classification pursuant to 10 CFR 61.55.

These activities represented one inspection sample.

b. Findings

No findings of significance were identified.

.3 Waste Characterization and Classification

a. Inspection Scope

The inspectors reviewed the licensee's methods and procedures for determining the classification of radioactive waste shipments including the use of scaling factors to quantify difficult-to-measure radionuclides (e.g., pure alpha or beta emitting radionuclides and those that decay by electron capture). The inspectors reviewed the licensee's radiochemical sample analysis results for each of the licensee's waste streams which consisted of radwaste bead resin/charcoal, condensate resin, various filter media, reactor water cleanup, and fuel pool cooling/cleanup resins and DAW. The licensee had not made any shipments of activated metals since the last inspection in this area, so this waste stream was not reviewed. The inspectors also reviewed the lower limit of detection values for each waste stream and corresponding radionuclide groupings in 10 CFR 61.55 to verify consistency with the NRC Branch Technical Position on Radioactive Waste Classification. These reviews were conducted to verify that the licensee's program assured compliance with 10 CFR 61.55 and 10 CFR 61.56, as required by Appendix G of 10 CFR Part 20. Additionally, the inspectors reviewed the licensee's waste characterization and classification program to ensure that reactor coolant chemistry data was periodically evaluated to account for changing operational parameters that could potentially affect waste stream classification and thus validate the continued use of existing scaling factors between annual sample analysis updates.

These activities represented one inspection sample.

b. Findings

No findings of significance were identified.

.4 Shipment Preparation and Shipment Manifests

a. Inspection Scope

The inspectors reviewed the documentation of shipment packaging, surveying, package labeling and marking, vehicle checks and placarding, emergency instructions, and licensee verification of shipment readiness for six non-excepted radioactive material and radwaste shipments made between November 2003 and March 2005. The inspectors verified that the requirements specified in the Certificate of Compliance were met for selected shipments made in NRC approved Type B casks. These shipments included:

- dewatered spent resin in a Type B cask;
- low specific activity material (contaminated metal) in a sea-land container;
- low specific activity material (DAW) in a sea-land container;

- low specific activity material (contaminated metal) in a sea-land container;
- surface contaminated objects (contaminated tools) in metal boxes; and
- dewatered spent resin in a Type B cask.

The inspectors selectively verified that the requirements of 10 CFR Parts 20 and 61 and those of the Department of Transportation (DOT) in 49 CFR Parts 170-189 were met for each shipment. Specifically, records were reviewed and some of the staff involved in shipment activities were interviewed to verify that packages were blocked/braced and labeled and marked properly, that package and transport vehicle surveys were performed with appropriate instrumentation and survey results satisfied DOT requirements, and that the quantity and type of radionuclides in each shipment were determined accurately. The inspectors also verified that shipment manifests were completed in accordance with DOT and NRC requirements, that shipping documents included the required emergency response information, that the recipient was authorized to receive the shipment, and that shipments were tracked as required by 10 CFR Part 20. The inspectors also reviewed the licensee's transportation security plan required by 49 CFR 172.800/802 and discussed its implementation with the licensee's principal shipping specialist.

Selected staff involved in shipping activities were questioned by the inspectors to verify that they had adequate skills to accomplish shipment related tasks, to determine if the shippers were knowledgeable of the applicable regulations, and to verify that personnel demonstrated adequate skills to satisfy package preparation requirements for public transport with respect to NRC Bulletin 79-19, "Packaging of Low-Level Radioactive Waste for Transport and Burial," and 49 CFR Part 172 Subpart H. Additionally, the lesson plan used for training Hazardous Material (Level 1) staff involved in radioactive material shipping was reviewed for compliance with the hazardous material training requirements of 49 CFR 172.704.

These activities represented two inspection samples.

b. <u>Findings</u>

<u>Introduction</u>

The inspectors identified a Green finding of very low safety significance and an associated violation of NRC requirements for the failure to perform complete radiation surveys of an outbound shipment of radioactive material to demonstrate compliance with DOT radiation level limits for the applicable mode of transport.

<u>Description</u>

During a review of radioactive material shipment records, the inspectors identified radiation survey practices that were not consistent with the licensee's procedure so as to demonstrate compliance with DOT requirements. Specifically, the radiation survey for a shipment which departed the licensee's facility on December 3, 2004, failed to include complete vehicle surveys as required by DOT regulation and the licensee's procedure, as described below.

Between mid-November 2004 and December 2, 2004, the licensee's radiation protection and radwaste staffs surveyed and decontaminated tooling and equipment used during the licensee's refueling outage and packaged the material into seven containers (metal boxes) for return shipment to a contractor located in Memphis, Tennessee. Each container was then individually surveyed, with no box exceeding a contact dose rate of 200 millirem/hour (the DOT radiation level limit for packages consigned as exclusive use in an open transport vehicle). These surveys identified a maximum contact dose rate on the side of an individual box at 140 millirem/hour with a maximum of 50 millirem/hour measured on the top surface of another box. The packaging and survey activities were accomplished on the fifth floor of the reactor building. Following these activities, the boxes were moved by crane down from the refuel floor and subsequently loaded onto an open flat-bed style transport vehicle. The boxes were positioned side-by-side on the flat-bed trailer, strapped-down, and a tarp covered the load.

On December 3, 2004, radiation protection staff performed surveys of the loaded vehicle at various locations, which included surveys at the vertical planes and under the vehicle, in the cab of the vehicle and at two meters from the vertical planes of the vehicle, as required. However, a survey was not performed at the upper surfaces of the load to ensure the 200 millirem/hour DOT limit was met at that location. A survey on the upper surfaces of the load was required by the licensee's procedure Plant Technical Procedure 67.000.103, "Survey of Radioactive Material Shipments," and was required by DOT regulation in 49 CFR 173.475 to demonstrate compliance with DOT limits. The individual that performed the survey reasoned that since no individual box exceeded the DOT limit when surveyed on the refuel floor, a top-of-load survey was not required following vehicle loading. However, given handling of the boxes during their transfer from the refuel floor to the loading area and further handling during the loading process, a potential for changes of the radiation profile of an individual box existed. Consequently, a top-of-load survey was warranted as well as required.

The shipment was consigned as Exclusive Use and categorized as Class 7 (radioactive) material, surface contaminated object (SCO II), containing a total activity of about 342 millicuries. The shipment departed the Fermi facility on December 3, 2004, and arrived at the contractor's facility on December 6, 2004. Receipt surveys performed by the contractor upon shipment arrival at its facility in Memphis, Tennessee, revealed no radiation levels above DOT limits.

Analysis

The failure to conduct a survey over the top surfaces of the load as required by the licensee's procedure and DOT regulation so as to demonstrate compliance with DOT radiation level limits is a performance deficiency. 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 to ensure adequate protection of the public from the exposure to radioactive materials transported into the public domain. Also, the issue involved an occurrence in the licensee's radioactive material transportation program that was contrary to the licensee's procedure and NRC/DOT regulations. Therefore, the issue was more than minor and represented a finding which was evaluated using the SDP for the Public Radiation Safety Cornerstone.

Since the finding was a transportation problem that involved a radiation survey issue, the inspectors utilized IMC 0609, Appendix D, "Public Radiation Safety SDP," to assess its significance. That assessment determined the issue represents a finding of very low safety significance (Green) because shipment package and vehicle radiation limits were not exceeded, no package breach occurred during transit, nor were certificate of compliance or low level burial ground issues applicable, and shipment notification and emergency information requirements were met.

Enforcement

Title 10 CFR 71.5(a) required each licensee who transports licensed material outside of the site of usage, as defined in the NRC license, or where transport is on public highways, or who delivers licensed material to a carrier for transport, shall comply with the applicable requirements of the DOT regulations in 49 CFR Parts 170 through 189 appropriate to the mode of transport.

Title 49 CFR 173.475 required, in part, that before each shipment of any Class 7 (radioactive) materials package, the offeror must insure by examination or appropriate tests that the external radiation levels are within allowable limits in 49 CFR Parts 171-178. Title 49 CFR 173.441(b)(2) specifies that radiation levels for shipments consigned as exclusive use in an open flat-bed style vehicle not exceed specified values at any point on the vertical planes projected from the outer edges of the vehicle, on the upper surface of the load or enclosure if used, and on the lower external surface of the vehicle. Contrary to these requirements, on December 3, 2004, the licensee delivered to a carrier for transport on a flat-bed style vehicle packages which contained Class 7 (radioactive) material and failed to determine by appropriate test (i.e., radiation survey) that the upper surfaces of the load met radiation level limits.

Immediate corrective actions taken by the licensee included suspension of further radioactive material shipments pending tailgate training with those staff involved in shipment surveys. An extent of condition evaluation was also initiated. Additional corrective actions were being formulated. Since the licensee documented this issue in their corrective action program as CARD 05-21433, and because the violation is of very low safety significance, it is being treated as a Non-Cited Violation (NCV 05000341/200504-04).

.5 Identification and Resolution of Problems for Radwaste Processing and Transportation

a. Inspection Scope

The inspectors reviewed selected CARDs and any audit reports that related to the radioactive waste and radioactive materials shipping program since the last inspection to verify that the licensee had effectively implemented the corrective action program and that problems were identified, characterized, prioritized, and corrected. The inspectors also verified that the licensee's oversight mechanisms were capable of identifying repetitive deficiencies or significant individual deficiencies in problem identification and resolution.

The inspectors also selectively reviewed other corrective action reports generated since the previous inspection and a 2002 self-assessment report that dealt with the radioactive material shipping and radwaste processing program, interviewed staff, and reviewed documents to determine if the following activities were being conducted in an effective and timely manner commensurate with their importance to safety and risk:

- initial problem identification, characterization, and tracking;
- disposition of operability/reportability issues;
- evaluation of safety significance/risk and priority for resolution;
- identification of repetitive problems;
- identification of contributing causes;
- identification and implementation of effective corrective actions; and
- implementation/consideration of risk significant operational experience feedback.

These activities represented one inspection sample.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Identification and Resolution of Problems

Introduction

As discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify all issues were being entered into the licensee's corrective action system at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Some minor issues entered into the licensee's corrective action program as a result of the inspectors’ observations are included in the list of documents reviewed which is attached to this report.

a. Effectiveness of Problem Identification

(1) Inspection Scope

The inspectors reviewed CARDs 05-20581 and 05-20760 to verify the licensee’s identification of the problems was complete, accurate, and timely, and consideration of extent of condition review, generic implications, common cause, and previous occurrences was adequate.

(2) Issues

Failure to Promptly Identify Degraded AVR Condition

On December 3, 2004, operators received control room annunciator 4D53, “AVR General Alarm,” and responded in accordance with the alarm response procedure (ARP). Operators responding to the AVR panel noticed several alarms that were acknowledged and reset. Annunciator 4D53 alarmed again along with numerous alarms on the AVR panel including alarm faults 28 and 29, ARCNet Channel 1 Failure and ARCNet Channel 2 Failure, respectively. The alarms were again reset with the only one clearing being alarm fault number 28. A short time later, alarm fault 28 alarmed again.

Due to the previous AVR trip and reactor scram on September 3, 2004, the inspectors concluded the licensee had sufficient understanding that concurrent failures on both ARCNet channels 1 and 2 would cause an AVR trip. Although the conditions causing both AVR channel failures to alarm were not occurring concurrently, the fact that both channels alarmed sometime between the first and second AVR alarm reset was a significant indicator AVR stability was being challenged.

Given the number and nature of AVR alarms prior to the scram coupled with lessons learned from the previous AVR trip, the inspectors concluded that the licensee had enough information and sufficient time to take appropriate actions to prevent a reactor scram. The licensee entered this issue into their corrective action program as CARD 05-20581.

Failure to Promptly Identify Air in Residual Heat Removal System

The inspectors determined the licensee had seven indications reasonably available to them that could have been used to discover the air in the RHR system prior to the mode change on February 2, 2005. As of January 30, 2005, licensee personnel were aware of the following abnormalities suggestive of a larger issue than was originally assumed on January 25, 2005:

1. control room annunciator 2D86, “RHR Div. I/II Fill Line Pressure Low” was locked in for approximately 20 minutes;
2. control room instruments indicated the condensate flow had increased approximately 700 gpm;
3. while attempting to vent the SDC suction line, air had been sucked into the pipe;
4. the torus water level increased;
5. during re-venting, air was vented from the SDC suction line;
6. when the pump was started, reactor pressure vessel level dropped 7 inches; and
7. on January 30, 2005, air was vented from the head spray line.

During the initial attempt to place Division 2 RHR in SDC on January 25, 2005, operators in the control room received annunciator 2D86, “RHR Div. I/II Fill Line Pressure Low” shortly after opening the bypass valves around the RHR “B” pump discharge check valve. In addition, control room instruments indicated the condensate flow had increased approximately 700 gpm. While intermittent receipt of 2D86 would be considered normal during the filling and venting of RHR, having the alarm locked-in for about 20 minutes was not considered normal. The increase in condensate flow was also an abnormality.

While attempting to vent the RHR SDC suction line, the field operators noted that air was actually being sucked into the pipe. When the torus water level alarm was received, operators attributed it to the RHR manipulations. After investigating the source of the leakage, the operators determined the source of the water to be the RHR keep-fill line but did not know the air in the condensate surge tank had been sucked into the piping. During re-venting, a large amount of air was noted coming from the SDC suction line, the same line that previously was sucking in air. The control room was not notified of either the air initially being sucked in or the air vented out.

When operators started the pump, they noticed an approximate seven inch reactor pressure vessel level drop. The level drop was an indication that the system was not water solid because no drop should have occurred with the water level above the bottom of the dryer skirt. Five days later during a routine vent check, operators found air in the head spray line.

On February 4, 2005, during venting for preparations to again place Division 2 RHR in SDC, operators vented air from the Division 2 drywell spray line. The inspectors concluded that the information gained from the January 25, 2005, event and the newly acquired information that RHR was not water solid as expected should have prevented operators from attempting to start the pump before a full RHR system fill and vent was performed.

The inspectors concluded the licensee had sufficient information that could have been used to promptly identify the air in the RHR system after the January 25, 2005, mis-alignment. For example, through interviews, the inspectors learned control room personnel would have viewed the air being sucked into the suction piping as a significant issue warranting a greater level of scrutiny of the torus high level alarm than had been performed. Operators further stated it was reasonable to believe they would have performed a full system fill and vent which would have purged all the air and prevented the February 4, 2005, trip.

Failure to Thoroughly Evaluate the Effects of Air in the Residual Heat Removal System

The inspectors determined the licensee narrowly focused their analysis on the LPCI mode of RHR because they did not evaluate the effects of air on the containment heat removal (CHR) mode of operation. The inspectors concluded this was deficient because CHR was more risk-significant than LPCI.

Upon review of available documents, the inspectors determined air could have been trapped in an approximately five-foot section of piping supplying the CHR line. This line represented a high point branching off the same line that supplied the LPCI injection line which was one place where operators found air during one of their venting evolutions. As such, the inspectors determined the air could have accumulated in the CHR line thereby creating the possibility of a pressure transient the licensee had not previously evaluated.

The inspectors presented this information to the licensee who reviewed the inspectors' concerns. The licensee determined that although undiscovered air could have accumulated in the CHR line, any possible air entrapment would not have affected system operability. Supporting their evaluation was the fact that both divisions successfully operated in CHR during surveillance testing after February 4, 2005, for approximately 3 hours with no adverse effects noted. Additionally, the inspectors performed focused inspections on the relevant portions of RHR piping and noted no deficiencies indicative of a water hammer event.

b. Effectiveness of Corrective Actions

(1) Inspection Scope

The inspectors reviewed CARD 04-25446 to verify the licensee's corrective actions for CARD 04-24490 documenting an inadequate quarantining of parts were adequate to prevent recurrence.

(2) Issues

As described in inspection Section 4OA2.2 of Inspection Report 05000341/2004007, the inspectors identified a Green finding for the licensee's failure to quarantine parts for the high pressure coolant injection outboard steam isolation valve, E4150F003. As described in Section 4OA7 of this report, the licensee's immediate corrective actions were ineffective in preventing the failure to quarantine parts for the same valve in RF-10. The inspectors considered this to be an example of ineffective corrective action to prevent re-occurrence of a similar issue.

.2 Selected Sample for Annual Review: Repetitive Packing Leakage on Reactor Recirculation Pump Discharge Valve B3105F031A

a. Inspection Scope

The inspectors reviewed CARDS, DERs, WRs, and EDPs and interviewed station personnel regarding the repetitive packing leaks on reactor recirculation pump "A" discharge isolation valve B3105F031A. The inspectors opened one unresolved item to conduct a further review of the adequacy of the maintenance history of this valve.

b. Findings

Introduction

Discharge valve B3105F031A provides isolation for maintenance and the safety function of the valve is to close automatically to allow low pressure core cooling to the unaffected loop during a loss of coolant accident.

Reactor recirculation discharge valve B3105F031A has been leaking from its packing since RF-4 and was repaired by repacking the stuffing box, applying a torque to the gland packing nuts, and/or back seating the valve. The as-found and as-left packing leak averaged 32 drops per minute since this outage.

Over many outages, several maintenance activities were performed on the valve. Engineering Design Package 28886 replaced the 60 ft-lb torque motor with an 80 ft-lb torque motor, changed the actuator gear ratio, and revised the closure logic for the motor operated valve to use long torque switch bypass. Maintenance personnel implemented EDP 29258 to change the packing from a three to a single-stage packing, thereby requiring a greater torque application to the packing gland nuts. Also, EDP 30011 was issued to change the control wiring to bypass the open limit switch so the valve could be back seated electrically to stop packing leakage. The EDP was canceled due to a potential of exceeding the back seat force of 75,547 pounds; however, installed sensors indicated the force was not excessive. Further, during the performance of this EDP, the mechanics identified a 20.5-inch long score mark on the stem that damaged the packing whenever the valve was stroked. Mechanics discovered the Belleville washer guides, tubing that goes over the washer guide studs, were crushed due to insufficient packing found in the stuffing box. The mechanics discovered five packing rings inside the stuffing box instead of seven rings. Mechanics also discovered a slightly bent valve stem. No CARDs were written for the crushed washer guide, the insufficient quantity of packing or the bent stem. Two rings of packing were added and the washer guides were replaced.

On September 9, 2003, during power operations, drywell unidentified leakage increased above the administrative limit of 0.50 gpm to 0.54 gpm. A drywell walk down in RF-10 revealed the packing leak from B3105F031A accounted for a majority of the unidentified leakage.

During RF-10 on November 28, 2003, B3105F031A had a small stream leaking from the packing. The valve could not be placed on the back seat to perform the repack because the back seat failed to seal. Consequently, the valve was repacked three times. During these activities, mechanics discovered that a Belleville washer had been compressed. The compressed Belleville washer was not corrected and could lead to a higher potential for packing leak due to consolidation. To compensate for this, mechanics applied a higher torque of 183 ft-lb torque to the packing gland nut.

Following the December 4, 2004, plant shut down, the valve leaked a stream of water from the packing. The valve was repacked and the as-left leakage was 80 drops per minute.

On December 7, 2004, the licensee initiated CARD 04-26479 to document concerns with the repair history of B3105F031A, in particular, that the valve had been repacked three times during the refueling outage without meeting zero leakage criteria. A request was documented in the CARD to form a site team to review and to understand the entire repair history. This effort was in progress.

Additionally, on December 22, 2004, CARD 04-15842 was initiated documenting concerns regarding the bent stem and the score mark on the stem and unsuccessful repair attempts on valve B3105F031A. The CARD documented the potential for increased packing leakage and that the degradations on the valve stem could prevent the valve from closing. Several recommendations were listed in this CARD to improve monitoring of this valve and to prepare the plant in case of sudden valve failure. The CARD resolution concluded that the valve could meet its design function to maintain reactor coolant system integrity and to close based on successful testing with the existing degraded conditions. This CARD was closed to CARD 04-26479 on January 20, 2005. On February 22, 2005, CARD 05-13229 was written to document that the issues identified in CARD 04-15842 remained unresolved.

On January 24, 2005, operators manually shutdown the unit when the gasket on drywell cooler failed. Throughout forced outage 05-01, the licensee did not repack the valve. On February 3, 2005, operators commenced starting up the unit after completing forced outage 05-01. At 12 percent power, personnel entered the drywell and discovered a three-foot plume of steam from the valve stem area. The licensee manually shutdown the plant and repacked the valve.

The inspectors determined the valve packing has leaked since 1994 and the licensee decided to repack, back seat, and/or apply a torque to the packing gland nuts to slow leakage. Torque values have increased since RF-4 from 57 ft-lbs to 183 ft-lbs, attributable, in part to the change in packing design. Although the valve has passed every post maintenance test following these repairs and drywell unidentified leakage has remained well below the TS value of 5 gpm, the valve has experienced some degradations that have not been understood fully. These included the slightly bent stem, the score mark on the stem, the leaking back seat seal, and the cause for the two missing packing rings. These issues, which could eventually impact valve closure, may be indicative of improper maintenance conducted on this valve. This is an Unresolved Item (URI 05000341/2005004-05) pending the inspectors' review and the licensee's evaluation of the work history of this valve. Because of the leaking back seat seal, the licensee had planned to replace this valve during RF-11.

.3 Bypass Locking Mechanism for Locked High Radiation Door

a. Inspection Scope

The inspectors discovered that a locked high radiation door could be opened without a key. The inspectors discussed this issue with the Region III health physics inspector and Fermi 2 radiation protection personnel.

These activities represented one inspection sample.

b Findings

Introduction

On March 16, 2005, the resident inspectors noticed the door to the reactor building first floor steam tunnel, a high radiation area door that was required to be locked per TSs, could be opened without a key. A door handle was connected to a vertical rod and the rod engaged into the upper door jamb when the handle was turned upward. A tab welded on the rod and resting on the deadbolt locking mechanism prevented rotating the handle downward and rod disengagement from the jamb. The inspectors determined the rod could be rotated such that the tab could slip behind the deadbolt and withdraw the rod from the jamb without using a key. The inspectors reported this condition to the shift manager.

The licensee initiated CARD 05-21882 documenting the degraded locking mechanism and placed flashing lights in the area per Radiation Protection Conduct Manual MRP 06, “Accessing and Control of High Radiation, Locked High Radiation, and Very High Radiation Areas,” as an administrative control to prevent unauthorized entry. The door was modified to prevent rotating the rod locking mechanism. Further, the licensee conducted an extent-of-condition review of other high radiation and locked high radiation doors and initiated CARD 05-21903 when they discovered two additional doors where the locking mechanism could be bypassed.

This is an Unresolved Item (URI 05000341/2004005-06) pending NRC review and evaluation of the issue.

4OA3 Event Followup (71153)

.1 Drywell Cooler Failure

a. Inspection Scope

The inspectors responded to an unexpected increase in drywell unidentified leakage due to a failed waterbox gasket on a drywell cooler causing operators to declare a plant alert and a manual plant shutdown. The inspectors interviewed engineers and operators and reviewed documents associated with the event and previous repairs of the cooler that contributed to the gasket failure.

These activities represented one inspection sample.

b. Findings

Introduction

A self-revealed finding of very low safety significance (Green) and an associated Non-Cited Violation of 10 CFR 50, Appendix B, Criterion III, “Design Control,” was identified when the licensee failed to maintain appropriate design control for translating

the correct gasket configuration and torque values into maintenance procedures for proper reassembly of the drywell coolers.

Description

On January 24, 2005, the joint gasket on the northwest waterbox for drywell cooler T4700B004 failed. The drywell cooling system was neither required for safe shutdown of the plant nor required to mitigate the consequences of any accident. The system was designed to maintain the pressure integrity of the emergency equipment cooling water (EECW) system, which was considered its safety-related function. The drywell coolers were isolated at high drywell pressure to prevent transferring of heat loads to safety related equipment supported by EECW and located outside the drywell.

Non-radioactive water from the reactor building closed cooling water (RBCCW) system, used as a cooling medium for the drywell coolers, leaked from the failed gasket and collected in the drywell floor drain. Control room alarm 2D75, “Drywell Floor Sump Level Rate of Change High,” actuated when the rate of change exceeded two gpm in a 4-hour period. At that time, operations personnel did not know the source was RBCCW.

At 4:10 p.m., the shift manager declared an unusual event when drywell unidentified leakage increased above 10 gpm Operators placed the mode switch in shutdown at 4:19 p.m. At 4:40 p.m., the licensee declared an alert when unidentified leakage increased greater than 50 gpm

Chemists sampled the water from the drywell floor drain sump and found rust inhibitor used in the RBCCW system that confirmed the source of the leak was from the RBCCW. Operators aligned the Division 2 EECW system to the drywell coolers and confirmed that the EECW makeup tank level dropped, indicating that a leak existed in the system. After isolating the string of affected coolers, the unidentified leak rate dropped to less than 1 gpm and the licensee terminated the Alert at 10:28 p.m.

Condition Assessment Resolution Document 05-20426 was written to investigate the cause of the failure. During a walkdown inside the drywell, the licensee identified that the northwest outlet water box end cover gasket was extruded on cooler T4700B004. The extrusion was located at the upper right hand corner between the 7.5-inch spaced bolts.

The licensee examined the remaining thirteen coolers. Of these coolers, 6 of them and cooler T4700B004, had maintenance performed by licensee personnel previously. The remaining seven were originally installed by the vendor. The licensee identified the gaskets on the originally installed coolers were narrower than the full-faced gaskets installed on the coolers that had maintenance.

After 1975, to compensate for a thin, flexible tube sheet with stiffeners, the bolt spacing, and a full-faced gasket, the vendor changed the design from a full-faced to a narrow gasket to achieve a leak tight assembly. However, this design change was not provided to the licensee and, therefore, was not translated into WR instructions for repair of the coolers. During cooler repairs, maintenance craft fabricated replacement gaskets based on the old, full-faced, design as depicted on Cryenco Drawing 28127, Revision A. Neither the drawing depicting the narrow gasket nor the documentation of the design change could be located.

After a review of WRs, the licensee discovered torque values for the cover bolts were removed after 1992 because the mechanics were concerned about warping the tube sheet thereby creating tube leaks. Unfortunately, torque values were not included in procedures to properly tighten the bolts to achieve adequate gasket compression.

Typically, maintenance personnel used a pipe sealant, RTV, to hold the gasket in place and skill-of-the-craft, based on adequate gasket crush, to reassemble the coolers. The gasket material tended to creep and if insufficient torque was applied to the joint, a leak was possible.

Lack of adequate joint loading was a contributing cause to the event. The last maintenance conducted on cooler T4700B004 was in 1996 under WR 000Z966610 to repair a leak from the end bell. Instructions in the WR lacked bolt torque values and required a light coat of RTV be applied on the center rib to hold the gasket in place. The as-found torque values for the bolts were 5 ft-lbs to 52 ft-lbs around the end cover and 2 ft-lbs at the bolt where the gasket extruded. Further, RTV was applied liberally around the mating surfaces of the end bell and cooler, which was inconsistent with the instructions in the WR. The RTV had, in effect, become another gasket material in the joint. The sealant was not recommended for use as a gasket in applications where pressure exceeded 100 pounds per square inch due to the low tensile shear strength of the sealant.

Five (1, 2, 3, 4 and 10) of the 7 coolers with full-faced gaskets were replaced with the narrower design and a torque applied to the bolts sufficient to stop leakage. The post maintenance test included running EECW, which was at a higher pressure than RBCCW. No leaks were identified. The remaining 2 coolers (6 and 7) had wide-faced gaskets and were not repaired because they had been tested and repaired during plant construction and no maintenance had been performed since then. Recent examinations of these gaskets demonstrated no degradation.

The licensee determined the root cause of the gasket failure was inadequate compression as a result of the original equipment manufacturer design employing a thin, flexible tube sheet and a wide gasket. In this configuration, significant cooler end cover bolt torque was required to provide the required level of sealing performance for EECW/RBCCW system service requirements. Other contributing causes included: 1) insufficient flatness of tube sheet surfaces; 2) improper gasket adhesive; 3) bolt spacing pattern too wide; 4) inadequate testing; and 5) tube plug weld interfered with end cover fit.

<u>Analysis</u>

The inspectors determined that failing to maintain proper design control during initial installation of the coolers was a performance deficiency. The inspectors concluded the finding was more than minor in accordance with IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Disposition Screening,” because the finding was associated with the design control attribute of the IE Cornerstone and increased the likelihood of an initiating event. Additionally, since the design change occurred during plant construction with subsequent improper installation of the gasket, the inspectors considered whether this was considered an old design issue. Per IMC 0305, “Operating Reactor Assessment Program,” Section 06.06, “Treatment of Old Design Issues,” this finding was not considered an old design issue because it was identified through a self-revealed event.

The inspectors completed a significance evaluation of this issue using IMC 0609, “Significance Determination Process,” (SDP) Appendix A, “SDP Phase 1 Screening Worksheet for IE, MS and B Cornerstones.” In Appendix A of the IE Cornerstone, the inspectors answered “No” under transient initiators since the finding did not contribute to both the likelihood of an initiating event and the unavailability of mitigating equipment or functions would be unavailable, or increase the likelihood of a fire or internal/external flood. Therefore, the finding was determined to be of very low safety significance (Green).

Enforcement

10 CFR 50, Appendix B, Criterion III, "Design Control," required, in part, that measures be established to assure applicable regulatory requirements and the design basis for structures, systems, and components are correctly translated into specifications, drawings, procedures and instructions. Further, measures shall be established and implemented for the selection and review for suitability of materials, parts, and equipment that are important to the safety-related functions of structures, systems, and components.

Contrary to the above, sometime after January 24, 2005, while investigating the failure of a gasket joint on the northwest waterbox of drywell cooler T4700B004, the licensee discovered the vendor revised the design of the gasket from a wide-faced to a narrow gasket after 1975 to compensate for a thin tube sheet and achieve adequate torque for a leak tight assembly. However, the modification was not adequately implemented and in 1996, a wide-faced gasket was installed instead of a narrow gasket, as required. Further, the licensee failed to include in work instructions torque values for a wide gasket to ensure proper gasket compression for a leak tight joint. The licensee entered this issue into their corrective action program as CARD 05-20426. However, because this violation was of very low safety significance and because it was entered into the licensee's corrective action program, this violation is being treated as a Non-Cited Violation (NCV 05000341/2005004-07), consistent with Section VI.A of the NRC Enforcement Policy.

As part of the licensee's immediate corrective actions, maintenance personnel replaced the wide-faced gasket with a narrow gasket, applied adequate torque to the bolts to ensure a leak tight joint, and conducted an extent-of-condition review.

.2 Missed Declaration of an Unusual Event

a. Inspection Scope

The inspectors responded to an event where an unexpected increase in radiation levels in the reactor building occurred. Operators did not identify an entry into the emergency action level for an unusual event until 2 hours after the event. The inspectors reviewed procedures and drawings, interviewed operations personnel and inspected plant areas where the event occurred.

These activities represented one inspection sample.

b. Findings

Introduction

On February 7, 2005, at 2:29 a.m., operators discovered main steam line C drain valve B2100F080C remained at 50 percent open while trying to fully close the valve. Operators initiated CARD 05-20818 to document the failure. This valve provided the capability to drain the steam line downstream of the outboard main steam isolation valve on main steam line C. This valve has a minimum stop feature in the closed direction such that it is slightly open to maintain a minimum drain path. The valve remained open throughout the night shift and its failure status was turned over to the day shift.

At 5:18 p.m., operators started the hydrogen water chemistry system. The hydrogen water chemistry system was injected into reactor feedwater and was used to reduce the free oxygen concentration by combining it with hydrogen to form water. With a surplus of hydrogen, ammonia was formed and was carried away with the steam. Nitrogen was a component of ammonia. An increase in radiation occurred as ammonia was transported to the condenser because of gammas given off by the decay of nitrogen 16 to oxygen 16. During the pre-job brief for placing hydrogen water chemistry in service, operators did not discuss the impact of an increase in area radiation level with valve B2100F080C open.

At 5:34 p.m., control room operators received alarm 16D5, “Reac/Aux Building BSMT/Subbsmnt High Radn,” in the reactor building basement airlock area reading 100 millirems/hour. Operators entered Emergency Operating Procedure 29.100.01, Sheet 5, for secondary containment high radiation on area radiation monitor channel 30. The flowpath instructions in the emergency operating procedure referenced entry in the Emergency Plan procedure EP-101, AU2 for an unusual event for unexpected increase in plant radiation levels and was missed by the operators responding to the situation.

At 5:45 p.m., the operators determined the increased radiation level was due to B2100F080C being partially open. Main steam line outboard drains header isolation valve B2103F600, a master valve for isolating the downstream drains, was closed and hydrogen water chemistry flow was reduced. These actions caused radiation levels to reduce to 4.5 millirems/hour. Condition Assessment Resolution Document 05-20844 was written to document the event.

At 7:00 p.m., the operators recognized they had missed declaring the unusual event as required by EP101, AU2. Since the event was of short duration and had already terminated upon discovery of the missed classification, NUREG 1022, Revision 2, “Reporting Guidelines,” was reviewed. The NUREG required reporting the condition under 10 CFR 50.72 (a)(1)(I) which required notifying the NRC operations center for the declaration of any of the emergency classes specified in the licensee’s approved emergency plan for either a rapidly concluded event or an oversight in the emergency classification determined a post event review. Condition Assessment Resolution Document 05-20846 was initiated to document that operations personnel missed identifying the emergency classification in the emergency operating procedures.

This is an Unresolved Item (URI 05000341/2005004-08) pending NRC review of the circumstances surrounding the event.

.3 <u>(Closed) URI 05000341/200400501</u>: Extent of Condition Review Associated with EDG Fittings.

On June 2, 2003, licensee personnel performed maintenance on a lube oil pressure sensing line fitting to address a small oil leak associated with EDG-12. The fitting penetrated the engine bulkhead and was configured such that when the fitting was unscrewed from the bulkhead, an internal threaded fitting unscrewed as well. The internal threaded fitting was inaccessible. When the mechanic reconnected the fitting, full thread engagement on the internal fitting was not obtained. The loose connection resulted in indicated lube oil pressures lower than what was observed prior to the work, but above the 26 pound per square inch gauge procedural limit and the low pressure trip setpoint of 23 pound per square inch gauge.

This URI involved a review of whether other bulkhead fittings on EDGs 11, 13, and 14 were affected. The licensee developed a spreadsheet of all fittings penetrating the bulkhead and, using drawings, evaluated whether an internal connection existed. The following WRs were generated and completed to remove access covers near suspected bulkhead fittings for each associated EDG: 1) EDG-11, WR 000Z040951 was completed on June 23, 2004; 2) EDG-13, WR 000Z040987 was completed on January 27, 2005; and 3) EDG-14, WR 000Z040988 was completed on February 15, 2005. The internal fittings were checked for tightness and no loose fittings were found. The inspectors reviewed the work packages, interviewed the system engineer and observed the tightening of the internal connections. This item is closed.

.4 Review of Licensee Event Reports (LER)

(1) (Closed) LER 50-341/2004-003: Standby Liquid Control Pump Inoperable Due to Inadequate Lubrication

On November 2, 2004, the licensee discovered an inadequate level of oil in the standby liquid control pump "B" gearbox impacting pump operability. An operator was inspecting the gearbox because of high vibrations and an abnormal oil analysis from the previous run on October 24, 2004. The licensee determined the purge and drain volumes of previous oil samples were not replenished and the pump would not have been able to perform its intended safety function from July 24, 2004, through November 2, 2004. Corrective actions included inspecting the gearbox for damage, refilling the gearbox, performing an extent of condition review, and issuing night orders to share the lessons learned from the event with Operations staff. This issue resulted in a license-identified finding of very low safety significance as discussed in Section 4OA7 of Inspection Report 05000341/2004007. The LER was reviewed by the inspectors and no additional findings of significance were identified. The licensee entered this issue into their corrective action program as CARD 04-25097. This LER is closed.

(2) (Closed) LER 50-341/2004-004: Automatic Reactor Shutdown Due to Automatic Voltage Regulator Failure

On December 4, 2004, during start-up from RF-10, a turbine trip and reactor scram occurred at 57 percent reactor power. The turbine trip was due to an AVR failure because of improper maintenance on the system during RF-10 as described in Section 1R19.2 of this report. In addition, the inspectors determined this event was related to an issue with problem identification and resolution as described in Section 4OA2.2 of this report. This LER was reviewed by the inspectors and no additional findings of significance were identified. The licensee entered this issue into their corrective action program as CARD 04-26443. This LER is closed.

.5 Review of Licensee Events and Degraded Conditions

a. Inspection Scope

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

- followup on control room AVR alarms during the week of January 10, 2005;
- response to the alert due to increased unidentified leakage on January 24, 2005; and
- response to the RHR pump “D” trip on February 4, 2005;

These activities represented three inspection samples.

b. Findings

No findings of significance were identified.

4OA4 Cross-Cutting Aspects of Findings

.1 A finding in Section 1R13.2 of this report had, as its primary cause, a human performance deficiency, in that, an electrician failed to properly connect a test lead which caused a short and an associated unavailability of EDG-14.

.2 A finding described in Section 1R15.2 of this report had Human Performance as its primary cause when operators failed to follow procedures for placing RHR Division 2 in SDC resulting in a large amount of air pockets in both systems and the subsequent trip of the “D” RHR pump on February 4, 2005.

.3 A finding described in Section 2PS2.4 of this report had as its primary cause a human performance deficiency, in that, radiation protection staff failed to follow procedure and complete vehicle surveys of an outbound shipment to include all required areas.

4OA6 Meetings

.1 Exit Meetings

The inspectors presented the inspection results to Mr. O’Connor and other members of licensee management at the conclusion of the inspection on April 7, 2005. 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

An interim exit was conducted for:

- public radiation safety radioactive waste processing and transportation program inspection with Mr. K. Hlavaty on March 4, 2005, and a followup telephone conversation with Mr. H. Higgins on March 9, 2005.

4OA7 Licensee-Identified Violation

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

Cornerstone: Mitigating Systems

As part of the investigation into deficiencies related to the high pressure coolant injection outboard containment steam isolation valve, E4150F003, the licensee determined it would be necessary to quarantine both the limit and torque switch assemblies from the subject valve operator to determine if they contributed to the failure of the valve to close on August 12, 2004. After troubleshooting during RF-10, maintenance personnel disassembled the limit switch and radiation protection decontaminated both the limit and torque switches so they could be free released. 10 CFR 50, Appendix B, Criterion V, required activities affecting quality be accomplished in accordance with written instructions. Work Request 000Z042368 required that maintenance personnel quarantine the limit and torque switches for E4150F003. Contrary to this, the licensee failed to adequately quarantine the limit and torque switches. The licensee discovered the failure to effectively quarantine the parts during routine troubleshooting activities and entered this issue into their corrective action program as CARD 04-25446 on November 11, 2004. This issue was of very low safety significance because it did not affect the availability or reliability of plant equipment.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

W. O'Connor, Jr., Vice President Nuclear Generation
D. Cobb, Plant Manager
H. Higgins, Radiation Protection Manager
K. Hlavaty, Acting Plant Manager, Production
R. Libra, Director Nuclear Engineering
D. Noetzel, Manager Nuclear System Engineering
N. Peterson, Nuclear Licensing Manager
M. Philippon, Operations Manager
P. Smith, Director Nuclear Assessment
S. Stasek, Director Nuclear Projects
B. Weber, Supervisor Radwaste and Shipping Specialist

NRC

S. Campbell, Senior Resident Inspector, Fermi 2

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000341/2005004-01	FIN	Loss of Essential Trip Function for Emergency Diesel Generator 14
05000341/2005004-02	NCV	Residual Heat Removal Pressure Transient Due To Valve Misalignment
05000341/2005004-03	FIN	Inadequate Power Converter Module Post Maintenance Testing
05000341/2005004-04	NCV	Failure to Complete Outbound Shipment Vehicle Radiation Survey as Required by the Department of Transportation
05000341/2005004-05	URI	Review of Work History to Repair Repetitive Packing Leaks on B3105F031A
05000341/2005004-06	URI	Bypass Locking Mechanism for Locked High Radiation Door to Reactor Building First Floor Steam Tunnel
05000341/2005004-07	NCV	Inadequate Design Control for Assembling Drywell Cooler Gaskets
05000341/2005004-08	URI	Missed Declaration of an Unusual Event

Closed

05000341/2005004-01	FIN	Loss of Essential Trip Function for Emergency Diesel Generator 14
05000341/2005004-02	NCV	Residual Heat Removal Pressure Transient Due To Valve Misalignment
05000341/2005004-03	FIN	Inadequate Power Converter Module Post Maintenance Testing
05000341/2005004-04	NCV	Failure to Complete Outbound Shipment Vehicle Radiation Survey as Required by the Department of Transportation
05000341/2005004-06	NCV	Inadequate Design Control for Assembling Drywell Cooler Gaskets
50-341/2004-004	LER	Automatic Reactor Shutdown Due To Automatic Voltage Regulator Failure
50-341/2004-003	LER	Standby Liquid Control Pump Inoperable Due to Inadequate Lubrication

05000341/2004005-01	URI	Extent of Condition Review Associated with EDG Fittings.

Discussed

None.

LIST OF DOCUMENTS REVIEWED

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

1R01 Adverse Weather

C Procedure 27.000.06; Hot Weather Operations, Revision 0

1R04 Equipment Alignment

- Procedure 23.139; Standby Liquid Control System, Revision 38
- Procedure 24.107.03; Standby Feedwater Pump and Valve Operability and Lineup Verification Test, Revision 34
- Procedure 23.205; Residual Heat Removal System, Revision 87
- Procedure 23.307; Emergency Diesel Generator System, Revision 85
- CARD 05-20835; Base of Center Condenser Pump Leaks
- Procedure 23.107; Reactor Feedwater and Condensate Systems, Revision 101
- WR 000Z031838; Condensate Min Flow Controller Flow Drifting in Auto
- Drawing —5714-1; Condensate System Functional Operating Sketch, Revision AE
- Drawing M-2004-1; Diagram; Div. II Residual Heat Removal System, Revision Z

1R05 Fire Protection

C UFSAR Section 9.A.4.1; Fire Hazards Analysis; Reactor Building, Revision 11

1R06 Flood Protection

C UFSAR 9.3.3; Plant Equipment and Floor Drains
C UFSAR 2.4; Hydrologic Engineering
C UFSAR 3.4; Water Level (Flood) Design
C Regulatory Guide 1.102; Flood Protection for Nuclear Power Plants; Revision 1
C NRC Information Notice 2000-20; Potential Loss of Redundant Safety-Related Equipment Because of the Lack of High-Energy Line Break Barriers; December 11, 2000

1R11 Licensed Operator Requalification

- Scenario SS-OP-904-0209; Reactor Pressure Vessel Level Instrument Fails/Reactor Recirculation Pump Fails/Small Loss of Coolant Accident/Partial Failure of Emergency Core Cooling System, Revision 2

1R12 Maintenance Rule Implementation

- Maintenance Rule Conduct Manual
- STR 2004-003104
- WR E151040100; Calibrate RHRSW Pump “A” Min Flow Valve
- Selected RHRSW maintenance rule functional failure evaluations

1R13 Maintenance Risk Assessment and Emergent Work

- WR 000Z050680; Investigate DC Ground
- WR 000Z050712; PSA RCIC Unavailability

1R14 Non-Routine Events

- Procedure 23.205; Residual Heat Removal, Revision 87

1R15 Operbility Evaluations

- Design Calculation DC-5523; RHR/LPCI Injection Line Water Seal Post DBA-LOCA, Revision A
- CARD 05-21700; Degrading performance on E1100F031B
- CARD 05-21282; Evaluate the Use of Calcium Magnesium Acetate on the Residual Heat Removal Complex Roof
- ARP 5D107, Rev. 10; “Condensate Supply Header Pressure High/Low”
- ARP 2D86, Rev. 7; “RHR Div. I/II Fill Line Pressure Low”
- CARD 05-20760; “RHR Pump ‘D’ Trip Following Overpressure Alarm”
- Selected RHRSW Chemistry Sample Analyses
- DECo File Number TMSA-05-0015; “RHR Pressurization Analysis for CARD 05-20760"
- WR E246970804; “Perform Test on Medium Voltage Breaker and Relay Control, PI Motor”
- Procedure 23.205, Rev. 87; “Residual Heat Removal System”

1R16 Operator Work-Arounds

- TMSA-05-0007 dated January 21, 2005; Risk Assessment of Revised Operator Work-Arounds

1R17 Permanent Modifications

- Engineering Design Package 29328, Revision 0; Modification to the Reactor Recirculation Pump A Thermal Barrier Heat Exchanger

1R19 Post Maintenance Testing

- WR 000Z050680; Ground Fault on Neutral Line
- CARD 05-21300; Blown Fuse While Performing Troubleshooting
- CARD 04-26422; AVR General Alarm
- CARD 04-26443; Reactor Scram on AVR Trip
- WR 000Z042819; Replace MTG Exciter Converter Boards

- WR 000Z042596; Investigate/Troubleshoot Causes of AVR Alarm #47
- Procedure 24.307.48; Emergency Diesel Generator 14 - Fast Start Followed by Load Reject, Revision 9
- WR A412060100; Replace packing in Reactor Recirculation Pump Discharge Valve

1R20 Refueling & Outage Activities

- Operations Conduct Manual MOP 22; Operations Outage Management, Revision 1
- Procedure 22.000.05; Pressure/Temperature Monitoring During Heatup and Cooldown, Revision 39

1R22 Surveillance Testing

- Procedure 24.107.03; Standby Feedwater Pump and Valve Operability and Lineup Verification Test, Revision 34
- Procedure 47.000.02; Mechanical Vibration Measurements For Trending, Revision 31
- Procedure 23.107.01; Standby Feedwater System, Revision 30

1R23 Temporary Plant Modifications

- WR 000Z044320; Install Temp Mod 04-0033
- 10 CFR 50.59 Screen 04-0645; Modify West Station Air Compressor Lube Oil System
- TM 04-0033; Modify West Station Air Compressor Lube Oil System

1EP6 Drill Evaluation

C Scenario 37; Intrusion, Reactor Feed Pump Trip, ATWS, Revision March 2, 2005

2PS2 Radioactive Material Processing and Transportation

- Annual Radioactive Effluent Release Reports for 2002 and 2003; Tables Summarizing Solid Waste and Irradiated Fuel Shipments; dated April 29, 2003 and April 30, 2004
- PTP 65.000.610; Shipping Cask USA/9168/B(U); Revision 5
- MRP21; Radwaste Shipping Operations; Revision 10
- PTP 65.000.515; Receipt, Storage, Inventory, Inspection and Packing of Radioactive Material Shipping Packages; Revision 12
- PTP 65.000.506; Shipping Low Specific Activity Radioactive Material; Revision 18
- PTP 65.000.508; Shipping Less Than or Equal To Type A Quantities of Radioactive Material; Revision 13
- PTP 65.000.509; Shipping Greater Than Type A Quantities of Radioactive Material; Revision 17
- PTP 65.000.522; Shipping Surface Contaminated Object Radioactive Material; Revision 6
- PTP 65.704.001; Setup and Operating Procedure for the RDS-1000 Unit; Revision 1
- MRP26; Process Control Program; Revision 1
- NPRP-03-0226 and NPRP-04-0169; Scaling Factors Report; dated September 3, 2003 and October 7, 2004, respectively
- MRP24; Fermi 2 Title 10 CFR Part 61 Compliance Manual; Revision 3
- NPRP-04-0171; Detection Criteria for 10 CFR Part 61 Sampling; dated October 4, 2004

- CP-GN-528; Hazardous Material Orientation (Level 1 Training); Revision 3
- Hazardous Material Orientation (Level 1) Training Records for Selected Staff; dated June 8 and 15, 2004
- MGA20; Transportation Security Plan; Revision 1
- Shipment Manifest and Associated Shipment Related Documentation for Shipment No. 03-091; Dewatered Spent Resin in a Type B Cask; date shipped November 7, 2003
- Shipment Manifest and Associated Shipment Related Documentation for Shipment No. 04-022; Low Specific Activity Material (contaminated metal) in a Sea-Land Container; date shipped April 20, 2004
- Shipment Manifest and Associated Shipment Related Documentation for Shipment No. 04-063; Low Specific Activity Material (DAW)) in a Sea-Land Container; date shipped October 7, 2004
- Shipment Manifest and Associated Shipment Related Documentation for Shipment No. 04-068; Low Specific Activity Material (contaminated metal) in a Sea-Land Container; date shipped November 16, 2004
- Shipment Manifest and Associated Shipment Related Documentation for Shipment No. 04-092; Surface Contaminated Objects (contaminated tools) in Metal Boxes; date shipped December 3, 2004
- Shipment Manifest and Associated Shipment Related Documentation for Shipment No. 05-006; Dewatered Spent Resin in a Type B Cask; date shipped February 16, 2005
- Nuclear Quality Assurance Audit Report 04-0110; Radiological Material Transportation and Disposal Including Process Control Program; report dated July 16, 2004
- CARD No. 05-21433; Shipping Survey Not Performed in Accordance with 67.000.103; dated March 3, 2005
- NPRC-02-0377; Focused Self-Assessment of Radwaste Processing with the RDS-1000; report dated December 18, 2002
- CARD No. 04-20304; Implement New Radioactive Material Transportation/Shipping Regulations; dated January 28, 2004
- CARD No. 05-21008; Waste Liner Won’t Fit CNS 8-120B-2S Shipping Cask; dated February 14, 2005
- CARD No. 04-23913; Develop Radwaste Just in Time Training to Support DOT Regulatory Changes; dated August 30, 2004
- CARD No. 03-17714; Damaged Sealing Surface on CNS 8-120B-2 Cask Last Used at Fermi 2; dated October 14, 2003

<u>4OA2</u> <u>Identification and Resolution of Problems</u>

<u>4OA3</u> <u>Event Follow-Up</u>

- LER 2004-004; Automatic Reactor Shutdown Due to Automatic Voltage Regulator Failure
- LER 2004-003; Standby Liquid Control Pump Inoperable Due to Inadequate Lubrication

LIST OF ACRONYMS USED

ADAMS	Agency-wide Document and Management System
AVR	Automatic Voltage Regulator
CARD	Condition Assessment Resolution Document
CFR	Code of Federal Regulations
CHR	Containment Heat Removal
DAW	Dry-Active Waste
DOT	Department of Transportation
EDG	Emergency Diesel Generator
EECW	Emergency Equipment Cooling Water
gpm	Gallons Per Minute
LER	Licensee Event Report
LPCI	Low Pressure Coolant Injection
MS	Mitigating Systems
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records
PMT	Preventative Maintenance Testing
Radwaste	Radioactive Waste
RBCCW	Reactor Building Closed Cooling Water System
RF	Refueling Outage
RHR	Residual Heat Removal
SDC	Shutdown Cooling
SDP	Significance Determination Process
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
V	Volts
WR	Work Request