January 16, 2002

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

SUBJECT: FERMI 2 NUCLEAR POWER STATION
NRC INSPECTION REPORT 50-341/01-17(DRP)

Dear Mr. O'Connor:

On December 29, 2001, the NRC completed an inspection at your Fermi 2 Nuclear Power Station. The enclosed report documents inspection findings which were discussed on December 21, 2001, with Mr. Cobb, and other members of your staff.

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

Based upon the results of this inspection, the inspectors identified two issues of very low safety significance (Green) which were determined to involve a violation of NRC requirements. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating these issues as a Non-Cited Violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny this Non-Cited Violation, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-001; and the NRC Resident Inspectors at the Fermi 2 Nuclear Power Station.

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

Sincerely,

/RA/

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

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

Enclosure: Inspection Report 50-341/01-17(DRP)

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

REGION III

Docket No:	50-341
License No:	DPR-43
Report No:	50-341/01-17(DRP)
Licensee:	Detroit Edison Company
Facility:	Enrico Fermi, Unit 2
Location:	6400 N. Dixie Hwy. Newport, MI 48166
Dates:	November 17 through December 29, 2001
Inspectors:	S. Campbell, Senior Resident Inspector J. Larizza, Resident Inspector R. Alexander, Radiation Specialist T. Kim, Project Manager
Approved by:	Mark A. Ring, Chief Branch 1 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000341-01-17(DRP), on 11/17-12/29/01, Detroit Edison Company, Fermi 2 Nuclear Power Station. Maintenance Risk Assessments and Emergent Work Evaluation.

The inspection was conducted by resident and specialist inspectors. The inspection identified two Green findings which were examples of a Non-Cited Violation. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply are indicated by "No Color" or by the severity level of the application violation. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

Cornerstone: Initiating Events

- Green. The inspectors identified an example of a Non-Cited Violation of Technical Specification 5.4.1.a for using the incorrect procedure for restoring the north reactor feedwater pump following emergent work activities that involved inappropriate opening of the north reactor feedwater pump discharge valve. Control room operators used a system operating procedure that required plant conditions of 950 pounds per square inch reactor pressure and both north and south reactor feedwater pump turbines operating. However, actual conditions were about 650 pounds per square inch reactor pressure and only the south reactor feedwater pump turbine was operating.

  The finding had an actual impact of: 1) discharging about 1.8 million pounds mass per hour of cold water moderator to the reactor vessel, 2) an unexpected power excursion from about 4 to 11 percent, causing a one-half scram signal from intermediate range monitor E, 3) an unexpected reactor water level increase to 225 inches, which was above the Level 8 trip setpoint, and 4) sending isolation signals to the high pressure coolant injection pump, reactor core isolation coolant pump and the only operating south reactor feedwater pump (stopping water to the reactor vessel). The finding was of very low safety significance because the event occurred during reactor startup and at low reactor power level and the power level excursion was not significant. Because the finding was of very low safety significance and the finding was captured in the licensee's corrective action program, this finding is being treated as an example of a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy (Section 1R13).

Cornerstone: Mitigating Systems

- Green. The inspectors identified an example of a Non-Cited Violation of Technical Specification 5.4.1.a for not completing the valve lineup while venting and draining the Division 2 residual heat removal system after completing heat exchanger relief valve testing. The operator failed to complete the instructions for venting and draining the Division 2 residual heat removal system before the system was refilled and caused an inadvertent discharge of approximately 400 gallons of contaminated water into the reactor building.

The finding was more than minor for the following reasons: 1) high contamination levels in the reactor building resulted from the spill, 2) the potential loss of residual heat removal cooling water from the system, and 3) the potential challenge to electrical equipment wetted from the spill. The finding was of very low safety significance because neither personnel contamination nor personnel overexposure occurred, electrical equipment was not damaged, and the residual heat removal system was not required at the time of the event. Because the finding was of very low safety significance and the finding was captured in the licensee's corrective action program, this finding is being treated as an example of a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy (Section 1R13).

Report Details

1. REACTOR SAFETY

Cornerstone: Mitigating Systems

Plant Status

At the beginning of the inspection period, the plant had been shutdown to conduct the eighth refueling outage. On November 27, 2001, the refueling outage was completed and operators commenced raising reactor power. On November 28, 2001, power ascension stopped at 4 percent, when an unexpected power increase to 11 percent followed by an expected level increase to the Level 8 trip setpoint occurred while placing the north reactor feedwater pump in service. After resolution of the problems, restart of the unit occurred on November 29, 2001, and operators raised reactor power and synchronized the unit to the grid on November 30, 2001. The plant reached 100 percent on December 2, 2001. Power remained at 100 percent until December 6, 2001, when an operator broke a vent line on a stator cooling water heat exchanger causing the control room operators to scram the plant manually upon losing stator cooling water pressure. Following repairs, operators restarted the unit on December 6, 2001, and synchronized the main turbine to the electrical grid on December 8, 2001. Reactor power remained at 100 percent until operators decreased power to 65 percent to conduct a control rod shuffle on December 15, 2001. After completing the shuffle, reactor power was raised to 100 percent on December 16, 2001. Reactor power remained at 100 percent through the remainder of the inspection period.

1R01 Adverse Weather (71111.01)

a. Inspection Scope

On December 4 and December 20, 2001, the inspectors used procedure 27.000.04, "Freeze Protection Lineup Verification," to a conduct a partial walkdown of the residual heat removal (RHR) service water complex and reactor/auxiliary building to verify freeze protection readiness.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments (71111.04S)

a. Inspection Scope

The inspectors conducted a complete system alignment verification of the condensate and core spray systems. The verification included a review of documents to determine the correct system lineup, including abnormal and emergency operating procedures, drawings, the Updated Final Safety Analysis Report, and the vendors' manuals. Also, the inspectors reviewed outstanding maintenance work requests on the system and any

deficiencies that affect the ability of the system to perform its function. Outstanding design issues including temporary modifications, operator workarounds, and items tracked by engineering department personnel were reviewed. The walkdown identified any discrepancies between the existing system equipment lineup and correct lineup.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q)

a. Inspection Scope

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

- Third Floor Reactor Building (Zone 7)
- Fifth Floor Reactor Building, Refuel Floor (Zone 9)
- Third Floor Auxiliary Building, Control Room (Zone 9)
- Division 2 RHR Building (Zone 2)
- First Floor Reactor Building (Zone 5)
- Reactor Building South East Quadrant (Zone 2)

b. Findings

Following a review of maintenance activities for fire detection equipment (fire indicating lights, fire detection bells, ionization detectors, carbon dioxide shutoff dampers, and smoke detectors), the inspectors noted that about 168 components were not tested per fire detection zone operability procedures. The licensee initiated Condition Assessment Resolution Document (CARD) 01-20330 in response to the inspectors' concerns. This item will be an unresolved item, **(URI 50-341/01-17-01)** pending the inspectors' review of the licensee's evaluation of the CARD and a review of the criteria for testing fire detection equipment in various zones.

1R05 Fire Protection (71111.05A)

a. Inspection Scope

On December 18, 2001, the inspectors observed the licensee's fire brigade respond to an unannounced simulated fire on the first floor of the radwaste building in the chemical storage area. The inspectors observed proper use of protective clothing and self-contained breathing apparatus, the availability of sufficient fire fighting equipment, effective radio communications and effective fire brigade leader directions. The inspectors noted that the pre-planned drill scenario was followed and the drill objectives were met, and observed a drill critique at the termination of the drill.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors reviewed surveillance procedure 47.205.01, "RHR Division 1 (North) Heat Exchanger Performance Test," and reviewed data collected during the test. The inspectors reviewed 1996, 1998 and 2000 data for the previous Division 1 heat exchanger tests and examined the performance trending.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12Q)

a. Inspection Scope

The inspectors reviewed the system health reports, associated CARDs, white papers for probabilistic risk assessment on conditional probabilities, and the control room unit logs for the following systems to determine whether the maintenance rule program had been implemented appropriately by assessing the characterization of failed structures, systems, and components. The inspectors also determined whether goal setting and performance monitoring were adequate for the following systems:

- Condensate Storage Tank (P1100)
- Mechanical Draft Cooling Towers (E1156)
- Turbine Building Closed Cooling Water System (P4300)
- Safety Relief Valves (B2104)
- RHR Service Water (E1151)
- Condensate System (N2000)

b. Findings

No findings of significance were identified.

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

1. Restoration of Division 2 RHR System

a. Inspection Scope

On November 24, 2001, operators found Division 2 RHR keep fill isolation valve E1100F208 mispositioned open while restoring the system. The inspectors reviewed work packages, safety tagging records and Level 3 CARD 01-19330 and interviewed operators who were involved in the event.

b. Findings

The inspectors identified one Green finding involving an example of a Non-Cited Violation of Technical Specification 5.4.1.a for inadequate implementation of the RHR system operating procedure, which resulted in the inadvertent discharge of about 400 gallons of water from the Division 2 RHR keep fill system into the reactor building.

On November 24, 2001, the licensee completed an emergent work item per work request 000Z991909 to test the RHR Division 2 heat exchanger B outlet line relief valve E1100F025B. During system restoration, about 400 gallons of water were discharged through the Division 2 RHR supply to thermal recombiner water spray cooler T4804001B vent valve E1100F255 into the reactor building. The operators were using system operating procedure 23.205, "RHR System," to fill and vent the RHR system with water from the condensate storage tank. A non-licensed operator noticed the leak into the reactor building and notified the control room. A control room operator diverted the water flow from the open valve to the torus by opening Division 2 RHR test line valve E1150F028B and Division 2 RHR torus cooling line isolation valve E1150F027B. Radiological surveys indicated contamination levels above the 100,000 dpm/100 cm^2 high contamination limits at 110,000 dpm/100 cm^2. Water had migrated from the third floor to the basement and dripped onto electrical cable trays and equipment.

The licensee's investigation determined the cause to be inadequate implementation of system operating procedure 23.205 to complete the valve lineup after system draining. The operator, who performed Section 7.6, "Draining Division 1(2) RHR to Torus," completed only the portion of the procedure to drain the system. The operator did not perform the section that instructed closing of valve E1100F255 after the system was drained and vented. An incorrect assumption was made that the associated safety tagging record (2001-006932) would restore all valves to the proper lineup after the work was completed. The safety tagging record did not list a restoration position for valve E1100F0255. Attachment 1B, "Division 2 RHR Initial Valve Lineup," of procedure 23.205 required that valves E1100F208 and E1100F255 be closed. Drawing 6M721-5706-1, "RHR Division 2 Functional Operating Sketch," listed valve E1100F255 as closed.

The performance deficiency associated with this event was inadequate implementation of the valve lineup section in the RHR system operating procedure that led to the unexpected discharge of contaminated water from the RHR system onto the reactor building floor. The finding was more than minor for the following reasons: 1) high contamination levels in the reactor building resulted from the spill, 2) the potential loss of RHR cooling water from the system, and 3) the potential challenge to electrical equipment wetted from the spill. The event was of very low safety significance because neither personnel contamination nor personnel overexposure occurred, electrical equipment was not damaged, and the RHR system was not required at the time of the event.

Technical Specification 5.4.1.a requires written procedures be established, implemented, and maintained covering the activities specified in Regulatory Guide 1.33, Appendix A. Regulatory Guide 1.33, Appendix A, Item 4h requires procedures for

emergency core cooling water systems. On November 24, 2001, operations personnel failed to fully complete the draining and venting steps listed in Section 7.6 of procedure 23.205. Consequently, Division 2 RHR supply to thermal recombiner water spray cooler T4804001B vent valve E1100F255 was left open while filling the Division 2 RHR system, an emergency core cooling water system. Failure to fully implement the procedure is an apparent violation. However, because of the very low safety significance and because the issue is in the licensee's corrective action program, it is being treated as an example of a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy **(NCV 50-341/01-17-02)**. This Non-Cited Violation is addressed in CARD 01-19330.

.2 Restoration of 5 North Feedwater Heater

a. Inspection Scope

On November 28, 2001, operators found that main turbine extraction steam feedwater heater No. 5N drain valve N3016F355 had been mispositioned opened during plant startup. The inspectors reviewed work packages, safety tagging records and Level 2 CARD 01-19702 and interviewed operators who were involved in the event.

b. Findings

No findings of significance were identified.

.3 Restoration of North Reactor Feedwater Pump (N2102D010)

a. Inspection Scope

On November 28, 2001, water was unexpectedly sent to the reactor when operators restored the north reactor feedwater pump (NRFP) to service after repairing a leaking suction strainer. The inspectors reviewed logs and CARDs, and interviewed operators in response to a feedwater transient event that occurred while restoring the NRFP following the repair of a leaking suction strainer.

b. Findings

The inspectors identified one Green finding involving a second example of a Non-Cited Violation of Technical Specification 5.4.1.a for implementing an inadequate procedure while placing the NRFP in service. Operators used the incorrect procedure for conducting this activity and opened the NRFP discharge valve. The event caused water level to rise to the Level 8 trip setpoint, causing isolation signals to be sent to the high pressure coolant injection (HPCI) system, reactor core isolation cooling (RCIC) system, and south reactor feedwater pump turbines. The south reactor feedwater pump (SRFP) tripped and a half scram was generated from intermediate range monitor E. The event occurred as follows:

The licensee had developed a safety tagging record to isolate the NRFP to work on the associated strainer. The record listed the following positions for the valves:

- NRFP Suction Valve N2000F634 - closed
- NRFP Discharge Line Isolation Valve N2100F607 - closed
- NRFP Discharge Hydraulic Stop Valve N2100F045A - was not listed closed or open but remained open
- NRFP to Reactor Pressure Vessel Startup Level Control Isolation Valve N2100F611 - closed

Operators performed this lineup and mechanics tightened nuts on a strainer to stop the leak. At the same time, the SRFP was in operation, providing flow to the reactor through the startup level control valve and associated piping. Reactor power was about four percent, reactor pressure was approximately 650 psig and reactor water level was about 197 inches. Four of the intermediate range monitors (IRMs) were on Range 9 (scale 0-40) and three were on Range 10 (scale 0-125). The trip setpoint for the 0-40 percent range IRMs is 38 percent and the trip setpoint for the 0-125 percent range IRMs is 120 percent.

Operators selected procedure 23.107, "Reactor Feed Pump Operation," Section 7.2, "Restoring an Isolated Reactor Feed Pump Turbine with Condenser Vacuum Established," to restore the NRFP. However, prerequisites for this procedure were reactor pressure at 940 psig and both the north and south reactor feed pump turbines started. The operators failed to recognize that the plant was not in this condition while attempting to restore the system. The plant was at 650 psig with only the SRFP running and on startup level control. No procedure existed to restore the NRFP under the existing plant condition.

The operators decided to begin placing the NRFP in service and performed the valve lineup in procedure 23.107, Section 7.2. Operators opened valve N2100F611, which diverted some flow from the line containing the startup level control valve N2100F403 to the discharge line of the NRFP. When the operators continued performing the lineup in procedure 23.107, they failed to recognize the impact of opening NRFP discharge line isolation valve N2100F607. Once they opened the valve, operators saw that feed flow had increased and that the reactor water level had increased and closed the valve immediately. About 1.8 million pounds mass per hour of cold water went into the reactor.

Because the cold water (moderator) entered the reactor, reactor power increased from 4 to 11 percent. Consequently, because only IRM E exceeded the trip setpoint, a half scram occurred. Reactor water level exceeded the Level 8 trip setpoint at 215 inches and sent isolation signals to steam isolation valves for the RCIC and HPCI systems, the main turbine and the reactor feedwater pumps. The main turbine was not on line at the time. The SRFP tripped as designed. Also, isolation valves for the RCIC and HPCI turbines were already closed. Maximum water level reached about 225 inches, which was well below the main steam line nozzles to the main turbine. Level remained at 225 inches for greater than 5 minutes. Reactor power dropped quickly to the initial level of four percent. Operators subsequently restarted the SRFP to maintain water flow to

the reactor. The plant remained in a steady state condition following the event. Operators initiated CARD 01-22208.

The performance deficiency associated with this event is the use of an inadequate procedure for the existing plant conditions while restoring the NRFP after repairs, which led to the unexpected transient. This finding was greater than minor because it had an actual impact of an unexpected small power excursion and an unexpected level increase that isolated safety equipment and stopped water flow to the vessel during startup. The event was of very low safety significance because the transient occurred during reactor startup at low power level and the power level excursion was not significant.

Technical Specification 5.4.1.a requires that written procedures be established, implemented and maintained covering the activities specified in Regulatory Guide 1.33, Appendix A. Appendix A of Regulatory Guide 1.33, Item 4o requires procedures for operating the reactor feedwater system. Contrary to Technical Specification 5.4.1.a and Regulatory Guide 1.33, Procedure 23.107, "Reactor Feed Pump Operation," Section 7.2, "Restoring an Isolated Reactor Feed Pump Turbine with Condenser Vacuum Established," provided inadequate instructions for placing the NRFP in service with the plant at 650 psig and four percent power and only the SRFP in service. The procedure required that the plant be at 950 psig with both north and south feedwater pump turbines operating. This is an apparent violation. However, because of the very low safety significance and because the issue is in the licensee's corrective action program, it is being treated as another example of Non-Cited Violation **NCV 50-341/01-17-02**, consistent with Section VI.A.1 of the NRC enforcement policy.

1R14 Nonroutine Plant Evolutions (71111.14)

1. Level 8 Trip During Startup

a. Inspection Scope

On November 28, 2001, the inspectors observed how control room personnel responded to an unexpected increase in reactor power (4 to 11 percent), an unexpected increase in level to the Level 8 trip setpoint, and a half scram initiated by an IRM during reactor startup. The inspectors interviewed operators involved in the event, reviewed abnormal operating procedures, standard operating procedures, drawings, plant parameter strip chart recorder traces, and General Electric Transient Analysis Report System data.

b. Findings

No findings of significance were identified. However, the specifics of the emergent work issues that caused the event were discussed in Section 1R13 of this report.

2. Loss of Stator Cooling Water Pressure Causes Manual Reactor Scram

a. Inspection Scope

On December 6, 2001, the inspectors observed how control room personnel responded to an unexpected decrease in stator cooling water pressure that resulted in a manual scram of the reactor from 100 percent power. The inspectors interviewed operators involved in the event, reviewed abnormal operating procedures, standard operating procedures, drawings, plant parameter strip chart recorder traces, and General Electric Transient Analysis Report System data.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

On November 28, 2001, while performing surveillance tests of the HPCI system at 165 psig reactor steam pressure, a fluid transient occurred. The resulting flow peak of 1546 gallons per minute was less than the magnitude of previous non-damaging transients. A walkdown of the HPCI system was performed by the licensee. No damage to the piping or pipe supports was noted. Based on these observations and the bounded analysis there was no operability concern. A future modification to add a keep-fill system will prevent this condition from occurring.

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds (71111.16)

a. Inspection Scope

The inspectors reviewed the aggregate assessment of operator work-arounds for the third quarter of 2001. The inspectors reviewed the workaround impacts on reliability, availability, and potential for misoperation of any systems listed in the aggregate assessment of operator work-arounds. The review included the cumulative effects of operator work-arounds that could increase an initiating event frequency or that could affect multiple mitigating systems and the ability of operators to respond in a correct and timely manner to plant transients and accidents.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

Engineering design package 29068A, for replacing the emergency diesel generator 12 exciter, was reviewed and selected aspects were discussed with engineering personnel. This document was reviewed for adequacy of the safety evaluation and consideration of design parameters. The modifications were for equipment upgrades of existing equipment.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed the test data for the components for piping supports dedicated as American Society of Mechanical Engineering snubbers to ensure compliance with the code and Technical Specifications. The inspectors verified that the testing demonstrated that the snubbers were capable of performing their intended function.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage (71111.20)

a. Inspection Scope

The inspectors directly observed and verified whether operators appropriately followed standard operating procedures, implemented Technical Specifications, and conducted briefings correctly during the following activities:

- Reactor Criticality during Reactor Startup after Refueling Outage 8
- Main Turbine Generator Synchronization after Refueling Outage 8
- Reactor Criticality Point of Adding Heat during Reactor Startup after Forced Outage 01-01
- Main Turbine Generator Synchronization after Forced Outage 01-01
- Infrequently Performed Test or Evolution for Startup after Refueling Outage 8
- Drywell Inspection after Refueling Outage 8

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors witnessed and reviewed test data for the emergency diesel generator 14 loss of power, loss of coolant accident surveillance test. The inspectors reviewed the Technical Specifications to confirm that the surveillance activities had verified the equipment would perform its intended functions. The inspectors observed staffing levels of the control room and relay room, and in the field.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors reviewed the following drill critiques to evaluate the adequacy of the licensee's critique of performance in identifying weaknesses and deficiencies. The inspectors verified that the weaknesses were placed in the corrective action system and that all corrective actions for identified weaknesses were resolved for closed CARDs:

- Scenario 30.2, Radiological Emergency Response Preparedness Team Blue, Control Room Shift 5, May 1, 2001
- Scenario 31, Radiological Emergency Response Preparedness Team Red, Control Room Shift 1, July 17, 2001

b. Findings

There were no findings of significance identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS)

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 High Risk Significant, High Dose Rate Locked High Radiation Areas and Very High Radiation Areas

a. Inspection Scope

The inspectors reviewed the station's implementation of physical and administrative controls over access to High Radiation Areas, High Dose Rate Locked High Radiation Areas, and Very High Radiation Areas, including a discussion of these controls with the

Radiation Protection (RP) Manager and first line RP supervisors, to verify that revisions to procedures implementing these controls did not reduce the effectiveness and level of worker protection. Additionally, the inspectors selectively walked down the boundaries of Locked High Radiation Areas and Very High Radiation Areas reestablished since the completion of the station's recent refueling outage to verify adequate controls were in place.

b. Findings

No findings of significance were identified.

.2 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed several CARDs completed during the final month of the station's recent refueling outage related to radiation worker performance and RP technician proficiency. The inspectors reviewed these documents to assess the licensee's ability to identify repetitive problems, contributing causes, the extent of conditions, and corrective actions intended to achieve lasting results. Additionally, though the inspectors reviewed the licensee's High Radiation Area controls as discussed in Section 2OS1.1, the licensee did not identify any additional High Radiation Area access control issues during the inspection cycle.

b. Findings

No findings of significance were identified.

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

.1 Post-Outage ALARA Reviews

a. Inspection Scope

Due to the close proximity in time between the completion of the licensee's refueling outage and the inspection, the inspectors were only able to review two audits and self-assessments that focused on overall ALARA performance during the outage rather than assessing individual job ALARA performance. However, the inspectors reviewed the audit and self-assessment to assess the licensee's ability to identify repetitive problems, contributing causes, the extent of conditions, and corrective actions intended to achieve lasting results.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety (PS)

2PS2 Radioactive Material Processing and Transportation (71122.02)

.1 Review and Walkdowns of Radioactive Waste Systems

a. Inspection Scope

The inspectors reviewed the liquid and solid radioactive waste system description in the Updated Final Safety Analysis Report and the most recent radiological effluent release report (for calendar year 2000) for information on the types and amounts of radioactive waste (radwaste) generated for disposal.

The inspectors performed walkdowns of the liquid and solid radwaste processing systems located in the Radwaste and Onsite Storage Facilities to verify that the systems were as described in the Updated Final Safety Analysis Report and the Process Control Program, and to assess the material condition and operability of the systems. The inspectors also discussed the current operation of the system with members of the radioactive waste operations crew. In the case of abandoned radwaste equipment (i.e., asphalt extruder solidification system), the inspectors reviewed the licensee's administrative and physical controls implemented to isolate these systems to verify the equipment would not contribute to an unmonitored radioactive material release path and would not inadvertently affect operating systems.

b. Findings

No findings of significance were identified.

.2 Waste Characterization and Classification

a. Inspection Scope

The inspectors reviewed the licensee's method and procedures for determining the classification of radioactive waste shipments, including the licensee's use of scaling factors to quantify difficult-to-measure radionuclides (e.g., pure alpha or beta emitting radionuclides). Specifically, the inspectors reviewed the licensee's spring 2001 radio-chemical analysis results for the condensate resin, bead resin/charcoal, dry active waste, fuel pool cooling cleanup, and reactor water cleanup waste streams. The inspectors reviewed the report to verify that the licensee's scaling factors were accurately determined such that waste shipments were classified in accordance with the requirements contained in 10 CFR Part 61 and the licensee's Process Control Program. The inspectors also reviewed the procedure for transferring waste materials into shipping containers to determine if appropriate waste stream mixing and/or sampling procedures were utilized for the purposes of waste classification per 10 CFR 61.55.

The inspectors additionally reviewed the licensee's processes employed to ensure that changes in operating parameters, which may result in changes to the waste stream composition, are identified between the annual or biennial scaling factor updates.

b. Findings

No findings of significance were identified.

.3 Shipment Preparation

a. Inspection Scope

The inspectors were unable to directly observe shipments of radioactive material as the licensee was not conducting any radioactive material shipments during the inspection. Therefore, to ensure that the shipping activities were performed in accordance with the requirements of 49 CFR Parts 172 and 173, the inspectors examined the shipping packages described in Section 2PS2.4. For these shipments, the inspectors reviewed the final radiological surveys, labeling, placarding, vehicle inspections, and instructions to the driver. Additionally, the inspectors examined the training program provided to personnel responsible for the conduct of radioactive waste processing and radioactive material shipment preparation activities to assess the licensee's compliance with 49 CFR Part 172, Subpart H requirements. Specifically, the inspectors reviewed the lesson plans, student handouts, and course completion documentation for licensee and vendor-provided courses to ensure that personnel (shippers, RP technicians, and fuel handlers) had adequately completed both the awareness/safety training and function specific training applicable for their individual job functions.

b. Findings

No findings of significance were identified.

.4 Shipping Records

a. Inspection Scope

The inspectors reviewed a selection of non-excepted package shipments completed during calendar years 1999 - 2001 to verify compliance with NRC and Department of Transportation requirements (i.e., 10 CFR Parts 20 and 71; 49 CFR Parts 172 and 173). Specifically, the inspectors reviewed the following radioactive materials/waste shipment records:

- 99-001 Irradiated Reactor Hardware Liner (Type B, January 11, 1999)
- 99-041 Cs-137 Calibration Source (Type A, August 19, 1999)
- 00-013 Irradiated Hardware Liner (Type B, May 10, 2000)
- 00-040 High Pressure Turbine Rotor (Surface Contaminated Object II, April 18, 2000)
- 00-089 Dewatered Powdered, Charcoal, and Bead Resin (Low Specific Activity [LSA] II, September 27, 2000)
- 01-022 Powdered and Bead Resin (Unprocessed) Liner (LSA-II, May 8, 2001)
- 01-030 13 High Rad Drums (Compacted Dry Active Waste) (LSA-II, June 12, 2001)
- 01-077 Laundry (LSA-II, November 21, 2001)

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed self-assessments and CARDs completed during the previous 18 months which concerned the areas of radioactive waste processing and radioactive waste/material shipping. The inspectors reviewed these documents to assess the licensee's ability to identify repetitive problems, contributing causes, the extent of conditions, and corrective actions intended to achieve lasting results.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification (71151)

Mitigating System and Initiating Events Performance Indicator Verification

a. Inspection Scope

The inspectors reviewed licensee event reports, licensee memoranda, unit logs, and NRC inspection reports to verify the following performance indicators for second quarter 2001 through third quarter of 2001.

- Unplanned Scrams per 7000 Critical Hours
- Reactor System Activity
- Scrams with Loss of Normal Heat Removal
- Unplanned Power Changes per 7000 Critical Hours
- Safety System Unavailability, High Pressure Injection System
- Safety System Unavailability, RCIC
- Safety System Unavailability, RHR System
- Safety System Functional Failures
- Safety System Unavailability, Emergency AC Power

b. Findings

There were no findings of significance identified.

4OA5 Other

The inspectors reviewed the interim report for the May 2001 Plant Evaluation performed by an inspection team from the World Association of Nuclear Operators. No further inspection was deemed necessary by NRC inspectors, and no assessment was made of the results of the inspection.

.4OA6 Management Meetings

.1 Exit Meeting Summary

The inspectors presented the inspection results to Mr. O'Connor and other members of licensee management at the conclusion of the inspection on December 21, 2001. The licensee acknowledged the findings presented. No proprietary information was identified.

Specific Area Exits

Radiation Protection

Senior Official at Exit:	D. Cobb, Plant Manager
Date:	December 7, 2001
Proprietary (explain "yes"):	No
Subject:	Occupational Radiation Safety (Access Control and ALARA); Public Radiation Safety (Radwaste and Transportation)
Change to Inspection Findings:	No

KEY POINTS OF CONTACT

Licensee
H. Arora, Nuclear Licensing
M. Brown, Engineer, Nuclear Licensing
J. Carter, Supervisor, Radwaste
D. Cobb, Plant Manager
D. Craine, Supervisor, Radiological Engineering
J. Davis, Manager, Outage
T. Dong, Manager, In-Service Inspection
Q. Duong, Manager, Plant Support Engineering
S. Hassoun, Principle Engineer, Nuclear Licensing
R. Johnson, Supervisor, Nuclear Licensing
E. Kokosky, Manager, Radiation Protection
M. Kramer, Shift Manager, Operations
A. Mann, Manager, Operations
J. Moyers, Manager, Nuclear Assessment
D. Noetzel, Manager, System Engineering
W. O'Connor, Vice President, Nuclear Generation
N. Peterson, Manager, Nuclear Licensing
M. Philippon, Shift Technical Advisor, Operations
J. Priest, Nuclear Quality Assurance
S. Stasek, Director, Nuclear Assessment
J. Tibai, Manager, Maintenance Rule
B. Weber, Supervisor, Radwaste
J. Werner, Manager, Training
D. Williams, Assistant Manager, Radiation Protection

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

URI	50-341/01017-01	Fire Detection Equipment May not be Tested per Operability Test Procedures
NCV	50-341/01017-02	Inadequate Use of Procedures During System Restoration

Closed

NCV	50-341/01017-02	Inadequate Use of Procedures During System Restoration

Discussed

None

LIST OF ACRONYMS USED

AC	Alternating Current
ALARA	As-Low-As-Reasonably-Achievable
CARD	Condition Assessment Resolution Document
CFR	Code of Federal Regulations
HPCI	High Pressure Coolant Injection
IRM	Intermediate Range Monitor
LSA	Low Specific Activity
NRFP	North Reactor Feedwater Pump
SRFP	South Reactor Feedwater Pump
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RP	Radiation Protection

LIST OF DOCUMENTS REVIEWED

The following documents were selected and reviewed by the inspectors to accomplish the objectives and scope of the inspection and to support any findings.

1R01 Adverse Weather

Procedure 27.000.04	Freeze protection lineup verification	Revision 21
Procedure 24.000.02	Shiftly, Daily and Weekly required surveillances	Revision 100

1R04 Equipment Alignment

Dwg. 6M721-5714-1	Condensate System Functional Operating Sketch	Revision AB
UFSAR Section 10.4.7	Condensate and Feedwater System	Revision 8
Procedure 23.107	Reactor Feedwater and Condensate Systems	Revision 92
Procedure 23.104	Condensate Storage and Transfer System	Revision 60
Procedure 23.107	Condensate Filter Demin system	Revision 57
Alarm Response Procedure 5D108	Condensate System Low Flow	Revision 7
Alarm Response Procedure 5D130	North Hotwell Level Hi/Low	Revision 9
Procedure 24.202.04	HPCI System Offline Auto Initiation Time Response Test	Revision 38
Procedure 23.107.01	Standby Feedwater System	Revision 29
Emergency Operating Procedure	Reactor Pressure Vessel Control, Sheet 1	Revision 9
Dwg. 6M721-5707	Core Spray Functional Operating Sketch	Revision Z
UFSAR Section 6.2.3.3.2	Core Spray System	Revision 9
Procedure 24.203.03	Division 2 CSS Pump and Valve Operability and Automatic Actuation	Revision 40

Procedure 24.203.04	Core Spray Pump and Valve Operability and Position Verification Test	Revision 27
Procedure 24.203	Core Spray System	Revision 33
Procedure 43.203.005	Div 2 CSS Leakage Monitoring Test	Revision 25
Procedure 44.030.002	ECCS - Core Spray System Division 2 Logic Functional Test	Revision 34
Procedure 22.000.01	Plant Startup Master Checklist	Revision 48
Annunciator Response Procedure 2D3	Div 2 CSS Actuated	Revision 6
Annunciator Response Procedure 2D90	Div I/II Fill Line Pressure Low	Revision 8

1R05 Fire Protection

UFSAR Section 9A.4.1.8.1	Fire hazard Analysis: Reactor Building, Third Floor, Zone 7, El. 641 Ft 6 In.	Revision 10
Drwg 6A721-2400	Fire Protection Evaluation Plan Plot	Revision M
Drwg 6A721-2407	Fire Protection Evaluation Reactor and Auxiliary Buildings Third Floor Plan El-641"-6" and 643"-6"	Revision Q
UFSAR Section 9A.4.1.10	Fire Hazard Analysis: Reactor Building, Fifth Floor, Zone 9, El. 684 Ft 6 In	Revision 8
Drwg 6A721-2409	Fire Protection Evaluation Reactor and Auxiliary Buildings Fifth Floor Plan El-677"-6" and 684"-6"	Revision R
Drwg 6A721N-2042	Fire Protection Evaluation Residual Heat Removal Complex Upper Floor Plan El-617'-0"	Revision C
Drwg 6A721N-2041	Fire Protection Evaluation Residual Heat Removal Grade Floor Plan El-590'-0"	Revision E
Drwg 6A721-2401	Fire Protection Evaluation Reactor Building Subbasement Plan El-540'-0"	Revision K
UFSAR Section 9A.4.2.10	Fire Hazard Analysis: Control Room, Zone 9, El. 643 Ft 6 In, 655 Ft 6 In and 677 Ft 6 In	Revision 10
Procedure 20.000.22	Plant Fires	Revision 31

UFSAR Section 9A.4.4.1	Radwaste Building general description	Revision 11
Fire Brigade Drill Scenario No. 6	1st Floor RAD Waste Chemical Lab storage Area - El. 583'6"	

1R07 Heat Sink Performance

47.205.02	Residual Heat Removal Division 1 (South) Heat exchanger Performance Test	Revision 6
Job TG25010930	Perform 47.205.002, RHR Division 2 HX Performance Test	November 11, 2001
UFSAR Section 5.5.7	Residual Heat Removal	Revision 5
UFSAR Section 6.2.1.3.3	Recirculation Line Break Long Term Response	Revision 6
CARD 01-13239	Log Mean Temperature Differential (LMTD) Correction Factor Used in RHR Heat Exchanger Test Analysis	July 16, 2001
CARD 01-13240	RHR Heat Exchanger Test Acceptance Criteria	August 2, 2001
CARD 01-13241	RHR Heat Exchanger Design Fouling Less than Allowed in Heat Exchanger Performance Test	August 2, 2001
CARD 01-14727	NRC Concern - RHR Heat exchanger Monitoring	May 4, 2001

1R12 Maintenance Rule Implementation

NUMARC 93-01	Nuclear Energy Institute Industry Guideline for Monitoring Effectiveness at Nuclear Power Plants April 1996	Revision 2
	Maintenance Rule Desk Top Reference	July 2, 2001
PRA Ranking Table 4.1	Probabilistic Importance Measure	
Log 98-002	Maintenance Rule position Paper: Bases Summary for Maintenance Rule Performance Criteria, Table 1	Revision O, October 2, 1998
Log 96-01	Maintenance Rule Position Paper: Development of “Conditional Probability” for SSCs Modeled in the Fermi 2 PSA	Revision 1, October 2, 1998

Log 96-002	Maintenance Rule Position Paper: Development of Train and Divisional Level Conditional Probability, Allowed Number of Failures and Out-of-Service Hours, and Redundancy Factor	Revision 1, October 2, 1998
MR06, Section 5.2.1	Establishing Performance Criteria	Revision 6
MR06, Appendix H	Performance Criteria Summary	Revision 8
	Control Room Logs for Condensate Storage Tank (P1100), Mechanical Draft Cooling Towers (E1156), and the Turbine Building Closed Cooling Water System (P4300)	December 31, 1998 - December 5, 2001
	Condition Assessment Resolution Documents for the Condensate Storage System (P1100), Mechanical Draft Cooling Towers (E1156), and the Turbine Building Closed Cooling Water System (P4300)	December 31, 1998 - December 5, 2001
	Work Requests and Preventive Maintenance Task for the Condensate Storage System (P1100), Mechanical Draft Cooling Towers (E1156), and the Turbine Building Closed Cooling Water System (P4300)	December 31, 1998 - December 5, 2001
	Control Room Logs for the Safety Relief Valves (B2104), Residual Heat Removal Service Water (E1151), and Condensate System (N2000)	December 31, 1998 - December 19, 2001.
	Condition Assessment Resolution Documents for the Safety Relief Valves (B2104), Residual Heat Removal Service Water (E1151), and Condensate System (N2000)	December 31, 1998 - December 19, 2001.
	Work Requests and Preventive Maintenance Task for the Safety Relief Valves (B2104), Residual Heat Removal Service Water (E1151), and Condensate System (N2000)	December 31, 1998 - December 19, 2001.
	Critical Performance Evaluation Data for Maintenance Rule Functional Failures	December 31, 1998 - December 5, 2001
000Z004113	Minor Maintenance Form: Shaft Seal Leaking for Emergency Hotwell Pump	November 28, 2000
STR 00-4163	Safety Tagging Record to Replace Shaft Seal for the Emergency Hotwell Pump	November 28, 2000

6M721-5721-1	Condensate Storage and Transfer System Operating Sketch	Revision U
6M721-2006	Condensate Storage and Transfer System Diagram	Revision AZ
CARD 99-11515	CST Level Indication Lost Due to Freezing	January 5, 1999
000Z2990526	Change Oil in MDCT Fan C Gear Reducer	March 31, 1999
STR 99-0296	Safety Tagging Record to Change Oil in MDCT Fan C Gear Reducer	March 31, 1999
CARD 00-17280	North TBCCW Pump Failed to Start	May 17, 2000
WR V293960311	Refurbish 480 Volt Breaker 72M-2D, Test Relays, Power Shield and Ammeter	March 3, 2001
STR 01-0219	Safety Tagging Record to Refurbish 480 Volt Breaker 72M-2 D, Test Relays, Power Shield and Ammeter	March 3, 2001

1R13 Maintenance Risk Assessment and Emergent Work

WR 000Z991909	Perform ASME “As-Found” & “As-Left” Relief Valve Testing Per 43.000.002	November 16, 2001
System Operating Procedure 23.205	Residual Heat Removal System, Attachment 1B, “Div 2 RHR Initial Valve Lineup”	Revision 73
STR 2001-006932	Safety Tagging Record for E1100 Division 2 RHR System Outage	November 24, 2001
CARD 01-19330	Nuclear Operator Finds and Stops Leak From Division 2 RHR (Mispositioned Valve)	November 24, 2001
WR 000Z013640	5N Heater Tube Leak Identified During FW Heater Integrity Check During S/D	November 6, 2001
System Operating Procedure 23.108	Extraction Steam and Heater Drains	Revision 55
STR 2001-006851	Safety Tagging Record for N2003B003	November 20, 2001
CARD 01-19702	Mispositioned Valve N3016F355 Found Open During Investigation of High Offgas In Flow	November 28, 2001
Drawing 6M721-5717-2,	Main Turbine Extraction Steam System Functional Operating Sketch	Revision R
System Operating Procedure 23.107	Reactor Feedwater and Condensate Systems	Revision 91

STR 2001-007059	Safety Tagging Record for N. Reactor Feedwater Pump	November 28, 2001
CARD 01-22208	Level 8 Trip While Unisolating NRFP	November 28, 2001

1R14 Nonroutine Plant Evolutions

MLS 11	Licensing/Safety Engineering Conduct Manual, Chapter 11 - Post Event investigations	Revision 10
CARD 01-22208	Level 8 Trip While Unisolating the NRFP	November 28, 2001
ST-OP-315-0046-001	Figure 2: Feedwater System	
Procedure 23.107	System Operating Procedure, “Reactor Feedwater System”	Revision 65
GETARS	General Electric Transient Analysis Recording System Data: Wide Range Reactor Pressure, Narrow Range Reactor Level, Wide Range Level	November 28, 2001
DCS	Digital Control System Data: Feedwater Level Control Summary	November 28, 2001
DCS	Digital Control System Data: Feedwater Control System Flow Summary	November 28, 2001
DCS	Digital Control System Data: North Reactor Feedwater Pump Flows and Pressures	November 28, 2001
Scram 01-01	Post Scram Data Evaluation	December 6, 2001
Scram 01-01	Sequence of Events Recorder Data	December 6, 2001
Scram 01-01	Average Power Range Monitor Traces	December 6, 2001
Scram 01-01	Traces for Reactor Vessel Level	December 6, 2001
Scram 01-01	Traces B21-R623A, “Reactor Vessel Level/Pressure”	December 6, 2001
Scram 01-01	Traces B21-R623B, “Reactor Vessel Level/Pressure”	December 6, 2001
Scram 01-01	Traces B21-R613, “Core Flow”	December 6, 2001
Scram 01-01	Traces C32-R609, “Vessel Pressure”	December 6, 2001
Scram 01-01	Traces C32-R607, “Main Steam Flow”	December 6, 2001
Scram 01-01	Traces N30-R824, “Condenser Vacuum”	December 6, 2001

Scram 01-01	Traces B21-R007, “Vessel Metal Temperature”	December 6, 2001
Scram 01-01	General Electric Transient Analysis System Data	December 6, 2001
CARD 01-22371	Manual Scram due to loss of Stator Water Cooling System	December 6, 2001

1R15 Operability Evaluations

CARD 01-20890	HPCI fluid transient during performance of 24.202.02	November 28, 2001

1R16 Operator Work-Arounds

NPOP-01-0199	Aggregate Assessment of Operator Work Arounds	September 17, 2001
TMIS-01-0155	Risk Assessment of Revised Operator Work Arounds - September 2001	September 17, 2001
ODE-006	Operator Work Arounds (ODE-006)	October 2001

1R17 Permanent Plant Modifications

EDP 29068	Exciter-Regulator Replacement for EDG 11, 12 and 14	Revision A
ECRs 29068-1 through 9	Changes for Packages to The Exciter-Regulator Replacement for EDG 11, 12 and 14	Revisions A, B, C and O
MES 19	Preparation and Control of Engineering Design Packages	Revision 13

1R19 Post Maintenance Testing

Log No. 01-048	ISI/NDE-IST Program Evaluation Sheet: Functionality of Snubber E11-3158-G30	November 8, 2001
Log No. 98-008	Pacific Scientific Snubber Visual Examination for Snubber B21-4093-G13	September 8, 1998
CARD 01-21931	SST Job Performance Records Do Not Reflect Actual Completed Performance	December 5, 2001
CARD 01-21930	Missing Documentation for Testing Snubber 810064	December 5, 2001

Form NIS-2	ASME Section XI Owner's Report for Repairs or Replacements of EESW Pipe Support P45-3353-G14	November 10, 2001
DER 91-0010	Deviation Event Report: Generic Letter 90-09, "Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions"	December 29, 1990
TSR-3155	Technical Service Request: EESW Column Separation Mitigation	July 13, 2001
RID 70971	Replacement Installation Document: Restore Function of Strut P45-3353-G14	November 8, 2001
Log 01-030	Pacific Scientific Snubber Visual Examination for Snubber B21-2593-G13	November 2, 2001
Procedure 43.000.011	Snubber Functional Test	Revision 37
ISI-NDE Program	Part C: Inservice Inspection-Nondestructive Examination (ISI-NDE) Program (Plan) for Snubbers	Revision 2
Log No. 98-008	Pacific Scientific Snubber Visual Examination for Snubber B21-2593-G13	September 7, 1998
WR 000Z932245	Test Reactor Core Isolation Cooling (RCIC) E51-3174-G33 Snubber	June 16, 1993
WR 000Z930020	Remove Snubbers, Install Struts and Modify Insulation on Drywell Piping	January 11, 1993
WR 000Z973173	Refurbish Snubbers as Required During RF-06	December 22, 1997
Job ID A334930628	Rebuild Hydraulic Snubbers During Refuel to Satisfy EQ, TS and Perform Preventive Maintenance	May 11, 1994
WR 000Z951759	Snubber E11-3152-G21 Has Cracked Tack Weld but is Operable	February 17, 1995
WR 000Z968255	Rebuild Hydraulic Snubbers During Refuel 06 to Satisfy EQ, TS and Perform Preventive Maintenance	December 6, 1996
WR 000Z946846	Snubber E11-3164-G26 is Leaking Oil At Noticeable Rate (Puddle on Floor)	September 26, 1994
WR 000Z013579	Pipe Clamp Yoke is Bent and Clamp is Loose, and Snubber Bushing is Wedged	November 1, 2001

WR 000Z953605	Replace the Other 3 Clamp Studs and Nuts For N30-3529-G36	May 26, 1995
WR 000Z945303	Investigate and Repair Snubber N3059G084	June 11, 1994
Job ID 009C891031	Embedded Pipe plate for Snubber P1166G009 is Damaged	November 11, 1989
Generic Letter 90-09	Alternative Requirements for Snubber Visual inspection Intervals and Corrective Actions	December 11, 1990

1R20 Refueling and Outages

Operations Conduct Manual MOP19	Reactivity Management	Revision 0
Procedure 23.623	Reactor Manual Control System	Revision 45
Procedure 22.000.02	Plant Startup to 25% Power	Revision 53
Infrequently Performed Test or Evolution IPTE 01-01	Cycle 9 Startup Test Program	Revision 0

1R22 Surveillance Testing

Procedure 24.307.04	EDG 14 Loss of Offsite Power and ECCS Start with Loss of Offsite Power Test.	Revision 32
Technical Specifications	3.8.1 AC Sources - Operating 3.8.2 AC Sources - Shutdown	

1EP6 Drill Evaluation

Scenario 30.2	Drill/Exercise Critique Summary, RERP Blue Team, Shift 5, May 1, 2001	May 31, 2001
CARD 01-10171	EOF Related Followup Actions As a Result of the March 7, 2001 RERP Drill	March 19, 2001
CARD 01-10190	Problems with Medical Response When the Plant Nurse or First Responder is not Available	May 18, 2001
Scenario 30.2	Drill/Exercise Critique Summary, RERP Red Team, Shift 1, July 17, 2001	August 9, 2001

CARD 01-10195	RERP: Evaluate RP Concerns for Security Personnel Response During Emergencies	January 11, 2001
CARD 01-16624	RERP Telephone System	July 31, 2001

2OS1 Access Control to Radiologically Significant Areas

	EF2 Radiation Protection Organization Self-Assessment of Plant Conditions and Personnel Performance during RF08 from Tours Conducted by RPO Personnel	November 12, 2001
CARD 01-17799	RRA Access Denial	November 9, 2001
CARD 01-21706	Worker Leaves Areas After Alarming PCM	November 16, 2001
CARD 01-21708	Accessing LHRA Gates	December 2, 2001
MRP06	Accessing and Control of High Radiation, Locked High Radiation, and Very High Radiation Areas	Revision 4

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

	EF2 Radiation Protection Organization Self-Assessment of Plant Conditions and Personnel Performance During RF08 from Tours Conducted by RPO Personnel	November 12, 2001
Audit Report 01-0115	Nuclear Quality Assurance Audit Report 01-0115 - Radiation Protection Program	October 22 - November 26, 2001

2PS2 Radioactive Material Processing and Transportation

	Fermi 2 UFSAR Sections 11.2 and 11.5	Revision 7 and Revision 8
	Radioactive Material Shipment Logs	1999 - 2001
CARD 00-12105	HAZMAT Training Requirements	January 3, 2000
CARD 01-12016	Liner LH-01-001 has 4 Defective Dewatering Elements	February 27, 2001
CARD 01-17909	RWCU Demin A Would Not Go Into Service	November 25, 2001
CARD 01-19082	Lockout Occurred Unexpectedly	November 23, 2001
Fermi 2 Technical Manual	Fermi 2 Process Control Program Manual	Revision 19
LP-GN-528-0003	Hazardous Material (HAZMAT) Orientation, Function Specific Training - Level 1	Revision 0
MRP24	Fermi 2 10CFR61 Compliance Manual	Revision 1

NRC-01-0031	Annual Radioactive Effluent Release and Radiological Environmental Operating Reports	May 1, 2001
NRPC-01-0166	Scaling Factors Report Dated April 25, 2001	May 29, 2001
NRPC-01-0168	Validation of Stainless Steel Laundry Container Shipment Using DAW Scaling Factors, Sample Reference Date - January 12, 2001	May 30, 2001
Plant Technical Procedure 20.000.27	Transportation Accidents Involving Radioactive Material from Fermi 2	Revision 7
Plant Technical Procedure 65.000.506	Shipping Low Specific Activity (LSA) Radioactive Material	Revision 16
Plant Technical Procedure 65.000.508	Shipping Less Than or Equal to A1, A2 Quantities of Radioactive Material	Revision 11
Plant Technical Procedure 65.000.509	Shipping Greater Than A1, A2 Quantities of Radioactive Materials	Revision 14
Plant Technical Procedure 65.000.515	Receipt, Storage, Inventory, Inspection and Packing of Radioactive Material Shipping Packages	Revision 9
Plant Technical Procedure 65.000.522	Shipping Surface Contaminated Object Radioactive Material	Revision 4
Plant Technical Procedure 65.000.523	Radwaste Shipments	Revision 4
Radioactive Material Shipment 99-001	Irradiated Reactor Hardware Liner #95455-6-3/4	January 11, 1999
Radioactive Material Shipment 99-041	Cs-137 Calibration Source (L-96-0027)	August 19, 1999
Radioactive Material Shipment 00-013	Irradiated Hardware Liner L91-001	May 10, 2000
Radioactive Material Shipment 00-040	HP Turbine Rotor	April 18, 2000
Radioactive Material Shipment 00-089	Dewatered Powdered, Charcoal, & Bead Resin LH-94-009	September 27, 2000
Radioactive Material Shipment 01-022	Powdered and Bead Resin (Unprocessed) Liner LH-00-005	May 8, 2001

Radioactive Material Shipment 01-030	13 High Rad Drums (Compacted DAW)	June 12, 2001
Radioactive Material Shipment 01-077	Laundry	November 21, 2001
RWP 01-1006	Survey, Segregate, and Compact Dry Active Waste. Perform Maintenance, Handling, Preparation, and Shipping of Radioactive Material	Revision 1
RWP 01-1014	Transfer and Process Water, Oil, Filter Media, and Filters. Hook-up, Tear Down, and Repair Equipment Associated with Dewatering and Solidifying Liners	Revision 2
Vendor Procedure FO-OP-032-483	Set Up and Operating Procedure for the RDS-1000 Unit at Detroit Edison Fermi-2	Revision 20

4OA1 Performance Indicator Verification

	Second and Third Quarter Performance Indicators for HPCI, RCIC, RHR, and Emergency AC Power Safety System Unavailability	
	Second and Third Quarter Performance Indicators for Safety System Functional Failures	
TMTE-01-0125	NRC Performance Indicators for HPCI, RCIC, RHR, and Emergency AC Power Systems Second Quarter 2001 Safety System Unavailability	July 17, 2001
TMTE-01-0186	NRC Performance Indicators for HPCI, RCIC, RHR, and Emergency AC Power Systems Third Quarter 2001 Safety System Unavailability	October 10, 2001
Dwg 6M721-5706-1	Residual Heat Removal (RHR) Division II Functional Operating Sketch	Revision X
Dwg 6M721-5706-2	Residual Heat Removal (RHR) Division I Functional Operating Sketch	Revision V
Dwg 6M721-5706-2	RHR Service Water Make Up Decant and Overflow Systems Functional Operating Sketch	Revision U
Dwg 6M721-5708-1	High Pressure Coolant (HPCI) Injection System Functional Operating Sketch	Revision AC
Dwg 6M721-5708-2	HPCI Lube Oil/Control Oil System Functional Operating Sketch	Revision H

Dwg 6M721-5709-1	Reactor Core Isolation Cooling (RCIC) System Functional Operating Sketch	Revision AC
Dwg 6M721-5709-2	RCIC Lube Oil/Control Oil System Functional Operating Sketch	Revision E
Procedure 44.030.155	ECCS - HPCI Torus Level Functional Test	Revision 34
Procedure 44.030.400	ECCS - HPCI/RCIC Condensate Storage Tank Level Loop, E41-N061B Calibration/Functional	Revision 21
Procedure 44.020.219	NSSSS -HPCI Exhaust Diaphragm Pressure, Division I Functional Test	Revision 27
Procedure 44.020.227	NSSSS - HPCI and RCIC Room Area Temperature Channel A Functional Test	Revision 29
	Unit Logs for HPCI (E41), RCIC (E51) and RHR (E11)	April 1, 2001 - September 30, 2001
	CARDs for HPCI (E41), RCIC (E51) and RHR (E11)	April 1, 2001 - September 30, 2001
STR 2001-005498	Safety Tagging Record: Repack HPCI Cooling Water to Lube Oil Pressure Control Valve	April 23, 2001
STR 2001-005472	Safety Tagging Record: Adjust Torque Switch for HPCI turbine Exhaust Stop Check Valve	April 20, 2001
STR 2001-005758	Safety Tagging Record: Clean Orifices D008 and 009 on HPCI Barometric Condenser	April 16, 2001
STR 2001-006431	Safety Tagging Record: Test/ inspect MOV MCC E5150F001	August 7, 2001
STR 2001-006644	Safety Tagging Record: Implement EDP 30202 to Replace GEMAC Flow Control Station	September 24, 2001
STR 2001-007122	Safety Tagging Record: Troubleshoot Cause for Valve E5150F044 not Opening	December 17, 2001
STR 2001-005659	Safety Tagging Record: Test Thermal Overloads for Valve E1150F611A	April 10, 2001
STR 2001-005935	Safety Tagging Record: Repack Valve E1100F086	April 27, 2001
STR 2001-005964	Safety Tagging Record: Repack Pump E1156C003	May 7, 2001
STR 2001-006041	Safety Tagging Record: Electrical Maintenance on Pump B	May 18, 2001

STR 2002-006041	Safety Tagging Record: Electrical Maintenance on Pump D	May 18, 2001
STR 2001-006022	Safety Tagging Record: Test Thermal Overloads for Valves E1150F024B and F027B	May 15, 2001
STR 2001-06122	Safety Tagging Record: Check Torque on Blade Clamping Hardware Bolting, Clean Blades, lubricate motor	June 4, 2001
STR 2001-006423	Safety Tagging Record: Test Thermal Overloads for E1150F007A and F027A and Test MCC Positions and Valves E1150 F004A, F004C and F016A. Lubricate E1150F034A and C	July 31, 2001