U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-341/94007(DRP)

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

Licensee: Detroit Edison Company
2000 Second Avenue
Detroit, MI 48226

Facility Name: Fermi 2 Nuclear Power Plant

Inspection At: Fermi Site, Newport, Michigan

Inspection Conducted: April 1 through May 27, 1994

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Inspection Summary

Inspection from April 1, 1994, through May 27, 1994
(Report No. 50-341/94007(DRP))

Areas Inspected: Routine, unannounced safety inspection by the resident inspectors of action on previous inspection findings, operational safety verification, engineered safety features systems, onsite event followup, current material condition, housekeeping, radiological controls, security, augmented inspection team (AIT) followup, licensee event report followup, maintenance activities, surveillance activities, plant modification review group meetings, technical surveillances, motor-operated valve (MOV) program improvement task force, system engineering, generic letter 89-10 MOV program testing status, chemistry, refueling/spent fuel pool activities, and licensee generated report reviews.

Results: Of the 21 areas inspected, one violation was identified that pertained to failure to follow procedures concerning an improper implementation of a tagout (paragraph 3.a). One non-cited violation was identified that involved the failure to properly document fuel movements (paragraph 8.a). Two unresolved items were identified that pertained to the falsification of logs by a contract RP technician (paragraph 3.f) and a crane accident that occurred during the movement of a steel liner that contained

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contaminated material (paragraph 3.a). Four Inspection Followup Items were identified pertaining to a turbine building heating, ventilation, and air conditioning (TBHVAC) failure (paragraph 3.c); an residual heat removal (RHR) MOV failure (paragraph 3.h); water chemistry concerns (paragraph 3.h); and fire protection performance concerns (paragraph 3.h). the following is a summary of the licensee's performance during this inspection period:

Plant Operations

The licensee's performance in this area was mixed. While good operator support contributed to a successful core offload, an improperly implemented tagout forced control room operators and security personnel to respond to the loss of two electrical buses and their associated equipment. In addition, events associated with the mispositioning of fuel in the spent fuel pool (SFP) were not communicated to the NRC resident office in a timely manner.

Plant Support

The licensee's performance in this area was good. Security personnel responded appropriately to the loss of an electrical bus that supplied security equipment. Aggressive radiological protection initiatives such as system flushes, additional temporary shielding, and pre-evolution drills were implemented in support of outage drywell work. During supervisory review of high radiation door verifications, licensee personnel determined that a contract radiological protection (RP) technician falsified some rounds sheets during the inspection period. Prompt corrective measures were taken.

Maintenance

The licensee's performance in this area was excellent. All maintenance and surveillance activities observed were properly performed and no problems were noted during the performance of these activities.

Engineering

The licensee's performance in this area was mixed. While nuclear engineers provided excellent support to the core offload, a relaxation of controls associated with spent fuel movements contributed to the mispositioning of a fuel bundle in the spent fuel pool during fuel sipping operations. During the fuel bundle mispositioning event, the refuel floor coordinator's focused observations and good questioning attitude resulted in the prompt identification of the error. In addition, corrective actions were immediately initiated to prevent recurrence, hence, the violation was not cited. During the investigation of a previously-identified NRC inspection item, the licensee discovered additional instrumentation that had inadequate seismic verifications. Verifications for these instruments were in progress at the end of the inspection period.

DETAILS

1. Persons Contacted

   Detroit Edison Company

   *S. Bartman, Supervisor, Chemistry
   *W. Colonnello, Director, Safety Engineering
   *J. Conen, Lead Compliance Engineer
   *J. Contoni, Supervisor, Design
   *R. Delong, Superintendent, Radiation Protection
   *R. Eberhardt, Assistant to Plant Manager
   *P. Fessler, Plant Manager, Operations
   *L. Fron, Section Head, Turbine Group
   D. Gipson, Vice President, Nuclear Generation
   *L. Goodman, Director, Licensing
   *E. Hare, Senior Compliance Engineer
   *J. Korte, Director, Nuclear Security
   *P. Lynch, Nuclear Shift Supervisor, Operations
   R. McKeon, Assistant Vice President/Manager, Operations
   *D. Nordquist, Director, NQA
   *R. Newkirk, Supervisor, Licensing
   *J. Nolloth, Superintendent, Maintenance
   *M. Odell, NMJ, Maintenance
   *G. Pierce, Supervisor, Work Control
   *J. Plona, Superintendent, Operations
   *W. Romberg, Assistant Vice President and Manager, Technical
   *R. Russell, Supervisor, Nuclear Training
   *R. Szkotnicki, Supervisor, Inspection & Surveillance
   *J. Thompson, Supervisor, Electrical Maintenance
   *J. Thorson, Reactor Engineer
   *N. Thrift, NSO, Operations
   *E. Vinske, Supervisor, Maintenance
   *R. Zavala, NPPO

   *Denotes those attending the exit interview conducted on May 27, 1994.

   The inspectors also had discussions with other licensee employees, including members of the technical and engineering staffs, reactor and auxiliary operators, shift supervisors, and electrical, mechanical and instrument maintenance personnel, and security personnel.

2. Action on Previous Inspection Findings (92701)

   a. (Open) Unresolved Item (341/94005-03(DRP)): Inadequate seismic qualification of transmitter piping. As a result of NRC concerns with work request WR 000Z932716, the licensee investigated the seismic qualification of a Barton flow switch associated with the EDG Service Water System. The investigation concluded that even though the instrument was marked as being seismically qualified, the switch mounting portion of the installation was not seismically qualified. The instrument was originally classified as QA level non-Q Seismic Category II/1 but was upgraded to QA level 1, Seismic Category I in as built notice (ABN) 8908-1 dated December 17, 1991. The licensee's data base stated that the Barton instrument mounting was seismically qualified in design calculation (DC)-3027. In actuality, DC-3026 addressed only the Seismic I qualification of the pipe stand and not the instrument mounted to the stand. The licensee performed an engineering analysis to verify that the instrument in question was actually seismically qualified. However, during the licensee's investigation, other components covered by ABN 8908-1 were found to be in the same status; i.e., classified as seismically qualified when in fact they were not. The licensee developed a time table to resolve seismic qualification concerns attributed to DC-3026 and determine the adequacy of the seismic information. Pending licensee completion of these activities, and NRC review of the results, this item will remain open.

3. Plant Operations

Fermi 2 remained in cold shutdown for the inspection period due to the extensive outage activities required as a result of the December 25, 1993 turbine generator failure. During the inspection period, the reactor was placed in Mode 5 (refuel) to commence the plant's fourth refueling outage.

a. Operational Safety Verification (71707)

The inspectors verified that the facility was being operated in conformance with the license and regulatory requirements, and that the licensee's management control system was effective in ensuring safe operation of the plant.

On a sampling basis, the inspectors verified proper control room staffing and coordination of plant activities; verified operator adherence with procedures and technical specifications; monitored control room indications for abnormalities; verified that electrical power was available; and observed the frequency of plant and control room visits by station management. The inspectors reviewed applicable logs and conducted discussions with control room operators throughout the inspection period. The inspectors observed a number of control room shift turnovers. The turnovers were conducted in a professional manner and included log reviews, panel walkdowns, discussions of maintenance and surveillance activities in progress or planned, and associated LCO time restraints, as applicable. The inspectors had the following observations:

1. On May 5, 1994, nuclear power plant operators (NPPO) hung tags in accordance with abnormal lineup sheet (ALS) 94-1185. The ALS was established to support 18 month testing (PM R016930217) of 4160 volt breaker 68K. Seven tags were

required for the tagout. The first six tags were to be hung on bus 68K position K4. The last tag was required to be hung on bus 69K position K6. The intent of the tagout was to prevent tripping breaker 6 during testing of breaker 4. The first NPPO hung the tagout and a second NPPO independently verified that the tagout was correct. However, in the field, the last tag was hung and verified as being correct, on bus 68K instead of the required location on bus 69K. Since the tagout was not properly implemented, bus 68K tripped unexpectedly during the PM and the control room operators entered abnormal operating procedures to deal with the loss of the bus. The electric fire pump and GSW pumps five and six were lost as a result of losing bus 68K. Operators entered the appropriate LCO for the loss of the electric fire pump. In this case, there were no adverse consequences as a result of losing the general service water (GSW) pumps because these pumps were already secured. Security bus 72T was also lost as a result of the tagout error (reference paragraph 3.g for additional details on the security department's response to the loss of the bus).

Technical Specification 6.8.1.a. requires that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Item 1.c of Appendix A of Regulatory Guide 1.33, Revision 2, February 1978, requires administrative procedures for "equipment control (e.g., locking and tagging)." Abnormal Lineup Sheet (ALS) 94-1185 for equipment control tagging specified hanging tag E-7 on Bus 69K, position K6. In addition, Section 4.11.2.1 of Fermi Management Directive, FMD PR1, "Procedures, Manuals, And Orders", specifies that independent verification shall be performed in accordance with Fermi Interfacing Procedure FIP-OP1-07, "Independent Verification". Section 5.2.1.4. of FIP-OP1-07 requires independent verification to be completed before the activity requiring it is started, and further defines activities as release of equipment for maintenance. Step 5.2.2.1.c. of FIP-OP1-07 requires that independent verification shall be performed by direct determination that affected equipment is in the correct position by a second individual. The operators' failure to follow licensee procedures with respect to the proper implementation of this tagout is considered an example of a Violation of Technical Specification 6.8.1 (341/94007-01).

2. On May 21, 1994, at approximately 9:30 a.m., a 35 ton truck crane tipped over on its side while lifting a 19,000 pound steel resin liner to a transport truck. Three steel liners were located in the basin surrounding the condensate storage tanks. The first of the three liners to be moved was lifted. After the lift, the crane was rotated to the right

to allow the liner to be set in the truck. As the load moved from the front of the crane to the side of the crane, the crane began to tip over. The boom struck the liner as the crane tipped over. Damage to the liner consisted of a partial crushing of the top and bottom; however, no tears were sustained in the liner. There was no release of radioactive material during this event. The preliminary investigation by the licensee indicated that the boom was extended too far for the load being lifted. Pending the licensee's investigation of the matter, and NRC review of the investigation results, this is considered an Unresolved Item (341/94007-02).

b. <u>Engineered Safety Feature (ESF) Systems</u>

During the inspection, the inspectors selected accessible portions of several ESF systems to verify status. Consideration was given to the plant mode, applicable Technical Specifications, Limiting Conditions for Operation requirements, and other applicable requirements.

Through observation, the inspectors verified that the following was acceptable: installation of hangers and supports; housekeeping; freeze protection, if required, was installed and operational; valve position and conditions; no potential ignition sources; and major component labeling, lubrication, cooling, etc. The inspectors also verified that instrumentation was properly installed and functioning and that significant process parameter values were consistent with expected values; that instrumentation was calibrated; that necessary support systems were operational; and that locally and remotely indicated breaker and valve positions agreed.

During the inspection, the accessible portions of the following ESF systems were walked down:

- Divisions I and II of Emergency Diesel Generators

c. <u>Onsite Event Follow-up</u> (93702)

During the inspection period, the licensee experienced several events, some of which required prompt notification of the NRC pursuant to 10 CFR 50.72. The inspectors pursued the events onsite with licensee and/or other NRC officials. In each case, the inspectors verified that any required notification was correct and timely. The inspectors also verified that the licensee initiated prompt and appropriate actions. The specific events were as follows:

1. <u>Turbine Building Heating Ventilation and Air Conditioning (TBHVAC) Exhaust Fan Damage</u>

On May 21, 1994, at approximately 8:25 a.m. the North TBHVAC Supply and Exhaust Fan tripped followed by the tripping of the Center TBHVAC Supply and Exhaust Fans approximately 12 seconds later as required by design. Loud noises were reported by workman in the area. Inspections performed following the event revealed several exhaust fan blades had been sheared off on the Center TBHVAC Exhaust Fan. Preliminary results from the licensee's investigation indicate that one blade failed due to high-cycle fatigue and, upon failure, sheared off adjacent fan blades on impact. These units did not have a history of vibration problems or of similar types of failures.

The licensee was continuing their investigation and plans to:

1) Remove the Center Exhaust Fan Hub and send the blade fragments to a remote metallurgical facility for inspection and failure determination.

2) Determine why the North HVAC Exhaust Fan tripped when the damage appears to be confined to the center unit.

Based on the results of these activities the licensee will determine what, if any, other actions are required. Pending the licensee' investigation of the matter and NRC review of the investigation results, this is considered an Inspection Followup Item (341/94007-03).

2. <u>Division 2 Control Air Compressor Automatic Start</u>

On May 21, 1994, at approximately 2:15 a.m., while restoring from safety related Electrical Distribution Bus 64B maintenance, the Division II Noninterruptible Control Air system (NIAS) Air Compressor unexpectedly started. The air compressor automatically started due to a NIAS low air pressure (85 psi) signal which occurred when normal station air was isolated when Valve P500F402 unexpectedly shut when power was restored to the control circuity. Following automatic start of the compressor, Valve P500F402 was subsequently reopened and the air compressor was returned to standby. The licensee initiated an investigation of the event, which was still in progress at the end of this inspection period. The inspectors will review investigation results, including the root cause evaluation and corrective actions, upon completion of the licensee's investigation.

d. Current Material Condition (71707)

The inspectors performed general plant as well as selected system and component walkdowns to assess the general and specific material condition of the plant, to verify that work requests had been initiated for identified equipment problems, and to evaluate housekeeping. Walkdowns included an assessment of the buildings, components, and systems for proper identification and tagging, accessibility, fire and security door integrity, scaffolding acceptability, radiological controls, and any unusual conditions. Unusual conditions included but were not limited to water, oil, or other liquids on the floor or equipment; indications of leakage through ceiling, walls or floors; loose insulation; corrosion; excessive noise; unusual temperatures; and abnormal ventilation and lighting. No significant discrepancies were identified; however, the inspectors observed several pipe supports on the condensate storage and transfer system that were not properly configured. Specifically, nuts were missing or not installed on several u-bolt pipe supports. The inspectors notified the licensee of the discrepancies. At the end of the inspection period licensee evaluation on system operability and further inspections of system pipe supports were in progress.

e. Housekeeping and Plant Cleanliness

The inspectors monitored the status of housekeeping and plant cleanliness for fire protection and protection of safety-related equipment from intrusion of foreign matter.

During plant tours the inspectors noted discrepancies such as extension cords and other items placed in electrical cable trays, drop lights hanging from valve handwheels, and one case where a cart was tied off to a Main Steam Isolation Valve Leakage Control System (MSIVLCS) instrument rack. The above items were discussed with appropriate licensee personnel and resolved. Early in the inspection period the inspector toured the torus room with the assistant Radiation Protection Manager and concluded that housekeeping in the torus room was poor. Scaffolding material, ladders, tools, flashlights and miscellaneous debris items were scattered throughout the area. The licensee initiated a cleanup effort in the room and its condition improved by the end of the inspection period. Overall, while housekeeping was adequate, it was below the standards typically maintained by the plant.

f. Radiological Controls (71707)

The inspectors verified that personnel were following health physics procedures for dosimetry, protective clothing, frisking, posting, etc., and randomly examined radiation protection instrumentation for use, operability, and calibration.

Plant personnel implemented several initiatives aimed at reducing radiation exposure in the drywell. System flushes were performed to lower general area dose rates, temporary shielding was increased by approximately one third, and mockups and drills were conducted prior to performing work activities in the drywell. While too early to accurately assess the results of the radiation exposure reduction initiatives, they appeared to be beneficial and demonstrated aggressive licensee actions to keep personnel exposure as low as reasonably achievable.

During the inspection period, the licensee reported an instance of records falsification by a contract radiation protection technician. The technician was to verify the status of high radiation area doors in the Turbine Building and Rad Waste Building. One of the doors under his purview had been de-posted from a Locked High Radiation Area (locked closed) to a General Radiation Area (door open). During the performance of his rounds, the individual "verified" that the door was still locked shut and posted as Locked High Radiation Area. The discrepancy was discovered when Fermi 2 radiation protection personnel reviewed the logs. The reviewer, who was cognizant of the fact that the area had been recently de-posted, brought the matter to RP management. During a critique of the event, the individual admitted that he had completed the log sheets without actually physically verifying the status of the area in question. The licensee pulled the individual's key card access and performed a review of the other work conducted by the individual. No other problems or discrepancies were identified during the review. Pending further NRC followup of this matter this is considered an Unresolved Item (341/94007-04).

g. Security (71707)

Each week during routine activities or tours, the inspectors monitored the licensee's security program to ensure that observed actions were being implemented according to the approved security plan. The inspectors noted that persons within the protected area displayed proper photo identification badges, and those individuals requiring escorts were properly escorted. Additionally, the inspectors also observed that personnel and packages entering the protected area were searched by appropriate equipment or by hand.

On May 5, 1994, security personnel responded to the loss of an electrical bus that supplied several items of security equipment (reference paragraph 3.a for event details). The bus failure affected one explosive detector and the assessment capabilities of two interior closed circuit television (CCTV) cameras. The affected explosive detector was not used for processing personnel and other operable detectors were available. The loss of CCTV assessment was not due to a failure of the cameras; lights in a building went out and security personnel were not able to assess the monitored zone by camera. A member of the security force was posted as a compensatory measure within ten minutes of losing assessment ability. The inspector determined that the security department's response to the loss of the bus was appropriate and timely.

h. AIT Followup

MOV Failure: Subsequent to the December 25, 1993, turbine generator failure event, Valve E1150F611B (residual heat removal (RHR) low pressure coolant injection mode bypass valve) failed to close when the plant was being placed in the RHR shutdown cooling mode on the day after the accident. The contactor failed to energize the motor-operated valve (MOV) twice, then closed on the third attempt. This valve was used to warm up the RHR system prior to placing the system in shutdown cooling.

The licensee identified the contactor as the component that failed. The contactor was replaced on January 7, 1994. Tentative causes appeared to include the use of a Cramolin Cleaner (R-5) solvent or the failure to completely engage a spring catch on the contact cover after maintenance activities . As interim corrective actions the licensee discontinued use of the cleaner, removed the cleaner when it was found on contactors, and performed inspections to ensure full engagement of spring catches. The problem with contactors for MOVs had been identified as a generic problem that had caused other MOV failures at this plant and was already being investigated at the time of the failure. Preliminary investigation of the defective contactor for valve E1150F611B confirmed the presence of excessive amounts of Cramolin (the oily solvent was dripping from the contactor). The results of this investigation will be reviewed by the inspectors during future inspections. Pending licensee final evaluation and disposition of the MOV failure, and NRC review of the results, this is an Inspector Followup Item (341/94007-05).

Reactor Water Chemistry: The reactor water chemistry was adversely effected by the December 25, 1993, turbine generator failure event. Prior to December 25, 1993, the reactor chemistry parameters were excellent. Reactor water conductivity was approximately 0.08 microSiemen/centimeter (uS/cm) and the concentration of chlorides was less than 2 parts per billion. After the event, the conductivity increased to over 185 uS/cm and the chloride concentration exceeded 10 parts per million (ppm). These levels exceeded the TS required shutdown reactor chemistry of 10 uS/cm and 0.5 ppm for conductivity and chlorides, respectively. The effects of the poor water quality on reactor components and fuel have not been completely analyzed. Pending licensee evaluation of the effects of the chemistry excursion, and NRC review of the results, this is an Inspector Followup Item (341/94007-06).

Fire Protection System: Fire protection personnel responded to the December 25, 1993, turbine generator failure event. Overall, Fermi 2 fire protection personnel and equipment performance was adequate. However, the AIT identified the following:

- The full fire brigade did not function as a team to respond to the turbine building to deal with the potential for existing fires until approximately thirty-seven minutes after the event. While, for this event, thirty-seven minutes taken to respond did not result in a delay in suppressing the actual fires, a more timely brigade response could have more significant impact in dealing with future fires.

- Communications problems caused delays in assessing the fire's extent. These problems are attributable to the use of hand-held radios by personnel wearing self contained breathing apparatus, water wetting communications equipment, and difficulties encountered using face mask microphones.

- There was no abnormal procedure for turbine building flooding. This delayed attempts to control flooding.

- Plant personnel experienced difficulty securing systems that were causing flooding. The difficulties could be traced to lack of instructions regarding equipment location and the lack of training for certain plant personnel in operating valves and electrical equipment.

- Plant personnel did not have in their possession a procedure for manually aligning the CO2 system to purge the generator. In addition, brigade members were unable to operate the CO2 system valves. This was due to either the water and oil on the valve handles or mechanical binding within the valves. At the AIT's completion, the exact cause of this problem had not been determined.

- Motion detectors worn by plant personnel during the response to the event (man down alarms) kept malfunctioning. This contributed to communication problems.

Pending licensee evaluation/disposition of the above mentioned items, and NRC review of the results, this is considered an Inspector Followup Item (341/94007-07).

Confirmatory Action Letter: By Confirmatory Action Letter (CAL) dated December 28, 1993, (CAL-3-93-018), the licensee committed to certain actions with respect to the turbine generator failure.

Items three through six of CAL 3-93-018 have been completed by the licensee and no longer remain as active commitments on the part of the licensee. Specifically, the following items documented in CAL 3-93-018 are considered closed:

- Place components, equipment, and data sources associated with the turbine-generator failure event, including those components that subsequently failed to perform as expected, under in situ quarantine to preserve evidence and data until released by you and discussed with the NRC's Augmented Inspection Team (AIT) or the NRC's Senior Resident Inspector.

- When developed, provide a copy of your proposed course of action for investigation of the event, recovery of the facility, and proposed corrective actions to the NRC Region III office for review.

- Maintain documentary evidence of your investigation effort and make this available to the AIT.

- Make those staff members on shift at the time of the event available for interview.

The two CAL items that remain to be completed are as follows:

- Complete an internal investigation to determine the sequence of events and root cause(s) of this event. Followup on this item will be conducted as part of NRC's evaluation and closure of LER 93-014.

- Complete an evaluation of the effects of the abnormal water chemistry experienced in the reactor on the fuel and reactor internals. Review of this CAL item will be conducted as part of NRC's evaluation and closure of inspection followup item 341/94007-06 discussed above.

One violation was identified in the area of Plant Operations.

4. <u>Safety Assessment/Quality Verification</u> (40500 and 92700)

   a. <u>Licensee Event Report (LER) Follow-up</u> (92700)

      Through direct observations, discussions with licensee personnel, and review of records, the following event reports were reviewed to determine that reportability requirements were fulfilled, that immediate corrective action was accomplished, and that corrective action to prevent recurrence had been or would be accomplished in accordance with Technical Specifications (TS):

(Closed) LER 341/92003: Technical Specification Required Shutdown - Stuck Open Drywell to Torus Vacuum Breaker. The licensee replaced the actuator cylinder and piston rod and satisfactorily retested the device. Additionally, the licensee requested, and the NRC approved, an amendment that revised the Technical Specification surveillance frequency for periodic cycling of the suppression chamber - drywell vacuum breakers from once per 31 days to once per cold shutdown. This LER is closed

(Closed) LER (341/92005): Standby Gas Treatment System (SGTS) Division 1 and 2 Inoperable. The licensee provided enhanced training to the licensed operators on operation of the PE relays and the associated LER was issued to the operators under required reading. Additionally, the licensee installed a modification to remove the interface between the safety related SGTS controls and its non safety related Cardox System PE Relays. This modification defeated the ability of the relay to inhibit an automatic start of SGTS. This LER is closed.

(Closed) LER (341/92006): Emergency Equipment Cooling Water (EECW) Automatic Initiation Due to Low Differential Pressure. The licensee revised system operating procedures, alarm response procedures, and training material to reflect that normal operation of the system requires two heat exchangers in service. The event and procedure revisions were presented to the operators during licensed and non-licensed operator re-qualification classes. This LER is closed.

In addition to the foregoing, the inspector reviewed the licensee's Deviation Event Reports (DER) generated during the inspection period. This was done in an effort to monitor the conditions related to plant or personnel performance, potential trends, etc. DERs were also reviewed to ensure that they were generated appropriately and dispositioned in a manner consistent with the applicable procedures. No discrepancies were noted.

No violations or deviations were identified.

5. Maintenance (62703 & 61726)

a. Maintenance Activities (62703)

Selected station maintenance activities were observed and reviewed to ascertain that they were conducted in accordance with approved procedures, regulatory guides and industry codes or standards, and in conformance with technical specifications.

The following items were also considered during this review: limiting conditions for operation were met while components or systems were removed from service; approvals were obtained prior to initiating the work; functional testing and/or calibrations were performed prior to returning components or systems to service; quality control records were maintained; and activities were accomplished by qualified personnel.

Portions of the following maintenance activities were observed or reviewed:

- 000Z943076 Replace 29 LPRMs/Control Rod Blade Shuffle
- 000Z921193 Remove Existing Piping, Valve E4150F011
- 000Z934970 Replace E4150F011 HPCI Test Return to CST
- N255911003 Inspect and Repair North RFP Turbine
- 000Z940451 Repair Generator Stator
- L639930223 Inspect Main Unit Condenser Internal Surfaces
- 000Z931127 Removed Damaged Tubes and Trapped Foreign Material From Condenser
- 000Z931129 Cut and Remove LP-1 and LP-3 Extraction Steam Piping
- 000Z930665 Install New Stainless Steel Manways for Main Condenser Water Boxes
- R016930217 4160V Switchgear Bus 68K Preventative Maintenance

No violations or deviations were identified.

b. Surveillance Activities (61726)

During the inspection period, the inspectors observed technical specification required surveillance testing and verified that testing was performed in accordance with adequate procedures, that test instrumentation was calibrated, that results conformed with technical specifications and procedure requirements and were reviewed, and that any deficiencies identified during the testing were properly resolved.

The inspectors also witnessed or reviewed portions of the following surveillances:

- E508921203 Perform MOV Thrust Test on E5150F045 RCIC Turbine Steam Inlet Valve
- 0001940427 Perform 24.307.016 EDG-13 Start and Load Test - Slow Start
- 000Z940164 Perform Fuel Sipping
- 000Z943076 Offload Core
- 43.000.017 ISI Inspection Dryer/Separator

No violations or deviations were identified.

6. Engineering (37700)

a. Plant Modification Review Group (PMRG) Meeting

On April 5, 1994 the inspectors attended a PMRG meeting held to discuss the conceptual design for a proposed modification to the RHR Inboard Isolation Valve E1150F015A and the Core Spray Inboard Isolation Valve E2150F005A. These modifications were intended to eliminate susceptibility to pressure locking. The meeting was initiated and chaired by a plant engineer. The PMRG meeting notice informing plant staff potentially impacted by the modification of this meeting requested that representatives from eight plant groups (Tech Group, Maintenance, Outage Planning, Operations, Modifications, Radiation Protection and ISI/LLRT) attend the meeting to ensure that all potential technical concerns were addressed and that there was a clear understanding among involved parties regarding the design change and testing requirements.

The inspectors noted that except for the ISI, work planning, and maintenance group representative, none of the other plant group representatives attended. During the meeting, the inspector determined that the lead engineer had not contacted, in preparation for the meeting, the valve manufacturer, the plant valve group engineers, or other utilities that had completed such a unique design change to ensure that the proper design is used to modify these important valves.

The inspectors concluded that lack of interest in attending the PMRG meeting by the multidisciplined individuals invited and the lack of adequate preparation by the lead engineer could adversely affect the modification process and team work. The licensee stated that similar problems relative to attending PMRG meetings were noted in the past and that the concerns raised by the inspectors would be examined and addressed.

b. Licensee Technical Surveillances Identified Errors in Design Documents

Although plant personnel errors appeared to show some improving trend overall, recently completed QA surveillances 94-0253 and 93-0361 noted errors in design documents which reflected inattention to detail. The inspectors were concerned that these types of errors could potentially lead to configuration control problems, to installation errors, and to work delays during implementation of EDPs.

To address the concerns noted above, Plant Engineering management established an error reduction task force to determine root cause, correct the problem, and recommend corrective actions to prevent recurrence. The inspectors will review this issue during followup inspections.

c. MOV Program Improvement Task Force

On April 11, 1994, the licensee established an MOV Review Team to assure that Fermi's MOVs would function properly and reliably following the 1994 outage. Personnel responsible for engineering, maintenance, testing, analysis, scheduling, and coordination of MOV work were organized into a single, separate group. The team was tasked with resolving recent MOV failures from all causes, completing generic letter (GL) 89-10 testing in a timely and satisfactory manner, resolving programmatic/process problems, and addressing all regulatory issues associated with MOVs. The inspectors considered this to be a positive management initiative in addressing MOV problems at Fermi. These organizational changes should provide opportunities for improved coordination of all related activities. The results of the MOV improvement task force will be evaluated by the NRC as the program progresses.

d. System Engineering

The inspectors interviewed engineers and supervisors in the engineering organization mainly in the Plant and System Engineering groups. The individuals interviewed appeared to be knowledgeable in their area of responsibility and eager to improve the engineering process. During the interviews, one of the system engineers was not aware of the open DERs on his systems. On request, the engineer provided the inspector with his latest available printout (from his systems file) of open DERs on his system. The list was dated September 18, 1993. The inspectors determined that six DERs were issued after September 18, 1993 on that system but the engineer was not made aware of this information. A list of open DERs was not easily retrievable on the data base used by the engineering staff. The licensee was in the process of upgrading the computer program to make this information easily retrievable on the database and available to the staff.

e. Investigation of Failure of MOV B3105F031B

Reactor Recirculation Pump Discharge Valve B3105F031B failed to close on demand subsequent to the December 25, 1993, automatic shutdown following failure of the main turbine/generator system. The valve failure resulted from three broken wires within the MOV operator.

The inspectors reviewed the examination, research, and analysis applied by the licensee in the determination of root cause of the F031B failure. Fracture surfaces of the wire strands showed a unique pattern which the licensee established to be caused by high cycle fatigue. The fracture surface was duplicated in the laboratory by flexing similar wire in the manner proposed as the failure mechanism. Fractures which were produced in the wire by other plausible methods of failure demonstrated a different fracture surface from that observed with the failed wires.

Through a combination of finite element analysis and physical testing of the valve and operator, the licensee determined that the resonant frequency (23-24 Hz) of the system and valve was similar to that of a wire bundle made to duplicate the failed bundle. The pump generated vibration in this frequency range when operated at 83-86% of maximum speed. A credible source of vibration was shown to be present because the pump was operated in this range for approximately 80 days during the previous three years and for a week during the month prior to failure. The failure analysis was considered acceptable and the proposed corrective actions were appropriate.

f. GL 89-10 MOV Program Testing Status

The licensee now includes 147 MOVs in their GL 89-10 program. This included the addition of one gate valve which would be installed to replace a check valve. Testing of these valves had been scheduled as follows:

Static Testing: All MOVs were to be static tested. Prior to this refueling outage, static testing of 86 MOVs was completed. However, repeat testing was required for a number of valves as a result of required post-maintenance testing and adjustments to torque switches after completion of the original tests. These adjustments were mandated by new data from Limitorque Part 21's, industry experience, and Fermi experience. Whenever a torque switch change was made, a new static test was necessary in order to determine the actual results of the change and to provide data for comparison with subsequent dynamic tests. Because of this, 116 static tests were scheduled for this outage, although only 61 of these were first-time tests.

Dynamic Testing: Of the 59 testable MOVs, 42 are scheduled for differential pressure (dp) testing during this outage. Prior to this outage, 14 satisfactorily completed testing. Several other dp tests were completed, but later disqualified. The remaining 3 MOVs are located in the drywell and can be tested only during start-up, at which time the drywell is inerted.

No violations or deviations were identified.

7. Chemistry

a. CST February 24-25, 1994, Discharge (IP 84750)

Prior to the February 24-25, 1994, discharge of the contents of the condensate storage tank (CST), the licensee provided aliquots of the pre-discharge sample to the NRC for confirmatory analyses. An aliquot of this sample was analyzed by the Radiological and Environmental Sciences Laboratory (RESL), US Department of Energy for the NRC. RESL analyzed the sample for gamma emitting nuclides, gross beta and alpha, hydrogen-3, strontium-89, strontium-90, and iron-55. The RESL results were in agreement with the NRC results, documented in Inspection Report No. 50-341/94003(DRSS), as shown in the attached Table 1.

b. CST Cleanup and April 15-17, 1994, Discharge (IP 84750)

Following the March 15 - 16, 1994, discharge of water from the CST, the licensee transferred additional volumes of water related to the December 25, 1993, turbine-generator event to the CST. The water was processed through several portable demineralizer vessels (vendor supplied) designed to provide filtration and ion exchange as described in Inspection Report No. 50-341/94003(DRSS). The discharge pathway consisted of diverting flow from the CST pump, through temporary carbon steel piping, to a filtration unit located in the Auxiliary Boiler House (ABH). Downstream of the filter (inside the ABH), a manually operated isolation valve, offline radiation monitor, and ultrasonic flow meter were installed. The flow was then routed to the neutralization tank waste water discharge line, which was a discharge pathway addressed in the Updated Final Safety Analysis Report.

The licensee sampled the CST (under NRC observation) during tank recirculation at 8:21 p.m. (EST) on April 14, 1994, and, as required, prior to discharge at 6:37 a.m. (EST) on April 15, 1994, and provided an aliquot of each sample to the NRC for confirmatory analysis and dose calculations. The licensee's analytical results of the required pre-discharge sample were in good agreement with NRC results (Table 2), with the acceptance criteria in Attachment 1. Additional sampling during the discharge presented no evidence of stratification within the CST tank. As shown in Table 3, the concentration of radioactive materials in the CST was below the limits of 10 CFR 20.1302(b)(2)(i), and the total activity was less than 1 curie (37 gigaBecquerels). The associated dose calculations were consistent with earlier results (Inspection Report No. 50-341/94003(DRSS). The estimated maximum, individual doses of 0.02 millirem (mrem) (0.2 microSieverts (uSv)) to the whole body and 0.03 mrem (0.3 uSv) to the liver were well below the appropriate federal annual limit of 100 mrem (1 mSv) total effective dose equivalent to a member of the public (10 CFR 20.1301(a)(1)) and the more restrictive licensee Technical Specification annual limit of 3 mrem (30 Usv), based on 10 CFR 50, Appendix I.

The licensee proceeded to discharge the CST contents (about 560,000 gallons) over a 30-hour period starting at about 9:00 p.m. (EST) on April 15, 1994. The controls described in Inspection Report No. 50-341/94003(DRSS) were employed throughout the evolution. The total activity discharged was about 700 millicuries (26 gigaBecquerels).

No violations or deviations were identified.

8. Refueling Activities (60710)

During the refueling outage, the inspectors observed the licensee's fuel handling operations and discussed refueling operations with plant operators and fuel handling personnel. The licensee used approved procedures for fuel accountability and movements. Communications between the control room and fuel handlers were established and, with one exception, effective. The inspectors witnessed fuel handling operations during several shifts from the control room, and in the reactor building.

On April 14, 1994, Unit 2 began its fourth refueling outage (RF04). The fuel pool gates were removed on April 18, 1994 and the first fuel assembly was moved from the reactor at 7:30 a.m. on April 20, 1994.

The inspectors verified that outage staffing was implemented in accordance with the Technical Specifications and approved plant procedures. These procedures delineated specific responsibilities of key personnel, interface and coordination among different organizations, control of contractor personnel, etc. The inspectors also verified that controls had been established for water level control, radiation monitoring, decay heat removal, containment integrity, and shutdown margin and reactivity monitoring. Equipment check requirements and required surveillance testing prior to fuel handling activities were reviewed and confirmed to have been completed in a satisfactory manner.

The inspector witnessed two shifts of fuel handling operations with no problems noted. Personal performance of individuals involved in these activities was good. The inspector also attended the licensee's morning outage meetings noting that they were detailed with inputs from all plant departments.

During this refuel outage, all of the fuel was unloaded from the reactor, moved to the spent fuel pool, and tested for indications of fuel leaks. Reactor engineering personnel identified a potential leaking fuel bundle earlier in the fuel cycle by performing a "flux tilt" evolution. Fuel sipping evolutions performed during this inspection period confirmed the identity of the leaking bundle. The licensee tested 100 percent of the bundles. In addition to confirming the identify of the suspected leaker, the fuel sipping evolution allowed the licensee to flush the bundle assemblies of any water chemistry impurities left as a consequence of the December 25, 1993, turbine failure event.

Fuel Bundle Mispositioning Event: On May 10, 1994, licensee and contractor personnel engaged in fuel sipping operations mispositioned a fuel bundle in the spent fuel pool (SFP). The specific sequence of events associated with the bundle mispositioning were as follows: 1) Bundle LYS341 was moved from the sipping can to the SFP. The bundle was erroneously placed in position J-24 when it should have been placed at position H-0. 2) Bundle LYS297 was moved from the sipping can to the SFP. The bundle was designated to go in position J-24 but this slot was now occupied. 3) Bundle LYS297 was temporarily placed in position H-27. 4) Bundle LYS341 was moved to position H-0 (this bundle was now located in the correct location as specified per procedure). 5) Bundle LYS297 was moved from position H-27 to position J-24 (this bundle was now located in its correct position per procedure). The initial error occurred when bundle LYS341 was placed in the wrong location in the SFP. The mistake was compounded when, contrary to procedural requirements, the bundles were manipulated to relocate them to their correct locations without the use of a SNM/Component Transfer Form Change Request. The error was initially discovered by the Refuel Floor Coordinator (RFC) who noted that the refueling crane was located over a different portion of the SFP than expected for where he anticipated the crew to be in the sequence of sipping operations. The RFC questioned the crew on the refueling bridge about the status of the sipping evolution and sequence of moves that had just occurred. The above errors were identified, the control room was notified of the status on the refuel floor, and fuel sipping evolutions were secured.

The following contributing factors were discovered during the inspector's review of the event: a non user-friendly (difficult to follow) procedure was utilized; there was no "spotter" available on the refueling bridge; the pre-evolution brief was inadequate; and poor communications existed between the key groups. In order to build flexibility into the procedure, the fuel movement directions did not specify which of the two sipping cans the fuel was to be placed in when it was moved from the SFP to the sipping location. In essence, this required the Station Nuclear Engineer (SNE) to develop the location details of procedural steps as the evolution was ongoing as well as direct and verify fuel movements. These details then needed to be copied into several procedural blanks; at the same time directing which sipping can the bundle was to be placed into. The SNE then had to jump ahead in the procedure and specify which can the bundle was to be removed from when it was returned to the SFP. In addition, the procedural steps were not set up in direct sequential order. In order to keep both sipping cans in use at all times, the SNE was required to move around in the procedure and work steps out of numerical order (this was administratively allowed by the procedure). The additional procedural or administrative burdens placed on the SNE distracted him from his other duties and may have contributed to the mispositioning. In addition to the procedural difficulties, there was one less contractor on the refuel bridge during sipping activities as compared to the core offload evolution. A spotter, whose task was to verify the crane operator correctly located and positioned fuel, was utilized during the core offload. However this individual was not initially stationed on the bridge for the fuel sipping tests and an opportunity to catch the error was lost. Finally, the pre-evolutionary briefing held for the fuel sipping test was inadequate. The personnel directly involved in performing the tasks (the SNE and refuel bridge driver) were not in attendance at the main briefing held in the main control room prior to commencement of fuel sipping. The importance of proper communications between the refuel floor and operators in the control room was discussed at the main brief. However, when the SNE and bridge operator were later briefed, communication requirements were not discussed. As a consequence, the main control room operators were not informed when the sipping evolution started nor were they initially kept informed of the step by step progress of the fuel movements. By not keeping the control room operators appraised of the status of the fuel movements, as was discussed at the main sipping brief, an additional opportunity to prevent the error was lost.

The licensee secured fuel sipping operations and implemented corrective actions prior to recommencing fuel sipping. Corrective actions included adding a spotter to the refuel bridge, revising the procedure to make it more user friendly, establishing communications between the refuel floor and the control room, and discussing the event with all involved personnel prior to recommencing fuel sipping. The licensee subsequently completed the fuel sipping operations and no further errors were noted.

The inspector reviewed the circumstances surrounding this violation against the criterion specified in 10 CFR Part 2, Section VII.B and determined that this event qualified for mitigation of enforcement sanctions. Therefore, a Notice of Violation will not be issued.

One non-cited violation was identified.

9. Report Review

During the inspection period, the inspector reviewed the licensee's Monthly Operating Status Report for March 1994. The inspector confirmed that the information provided met the requirements of Technical Specification 6.9.1.6 and Regulatory Guide 1.16.

The inspector also reviewed the licensee's Monthly Performance Reports for the months of March and April 1994.

No violations or deviations were identified.

10. Inspection Followup Items

Inspection Followup items are matters which have been discussed with the licensee, which will be reviewed by the inspector and which involve some action on the part of the NRC or licensee or both. Inspection Followup Items disclosed during the inspection are discussed in Paragraphs 3.c and 3.h.

11. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, violations, or deviations. Unresolved items disclosed during the inspection are discussed in paragraphs 3.a and 3.f.

12. Meetings and Other Activities

a. Management Meetings (30702)

On April 7 and 8, 1994 Mr. Ledyard, B. Marsh, Project Directorate II-1 toured the Fermi plant and met with licensee management to discuss plant performance and maintenance related issues.

b. Exit Interview (30703)

The inspectors met with the licensee representatives denoted in paragraph 1 during the inspection period and at the conclusion of the inspection on May 27, 1994. The inspectors summarized the scope and results of the inspection and discussed the likely content of this inspection report. The licensee acknowledged the information and did not indicate that any of the information disclosed during the inspection could be considered proprietary in nature.

Attachments:
1. Table 1
2. Table 2
3. Table 3
4. Attachment 1

Table 1

Fermi 2 Nuclear Station

Condensate Storage Tank
February 24, 1994

Nuclide	NRC Value (uCi/ml)[3]	NRC Error[1] (uCi/ml)	RESL[2] Value (uCi/ml)	RESL Error (uCi/ml)
CR-51	2.76E-07	9.uGH-08	< MDA[4]	
CO-58	6.47E-08	1.29E-08	< MDA	
CO-60	4.26E-07	2.36E-08	4.3E-07	6E-08
I-131	3.02E-08	1.08E-08	< MDA	
CS-134	1.47E-07	1.69E-08	1.0E-07	4E-08
CS-137	1.68E-07	1.69E-08	2.1E-07	4E-08
H-3	4.75E-04	5.9E-06	4.81E-04	6E-06
GROSS BETA	1.2E-06	6E-08	1.1E-06	9E-08
GROSS ALPHA	< MDA		9E-09	1.1E-08
FE-55	NA[5]		2.9E-07	4E-08
SR-89	NA		5.6E-08	9E-09
SR-90	NA		6E-09	4E-09

[1] The one standard deviation error attributable to counting statistics.

[2] The Radiological and Environmental Sciences Laboratory at the US Department of Energy's Idaho Operations Office.

[3] Micro-Curies per milliliter

[4] MDA = Minimum Detectable Activity

[5] NA: The NRC did not analyze for this nuclide.

Table 2

Fermi 2 Nuclear Station
Confirmatory Measurements

SAMPLE	NUCLIDE	NRC VAL.[1]	NRC ERR.[1]	LIC.VAL.[1]	LIC.ERR.[1]	RATIO[2]	RES[3]	RESULT[4]
CST	MN-54	5.63E-07	4.00E-08	6.57E-07	4.70E-07	1.17	14.1	Agreement
TANK	CO-58	9.37E-08	2.26E-08	9.19E-08	2.29E-08	0.98	4.2	Agreement
4/15/94	CO-60	4.15E-07	4.14E-08	5.71E-07	5.03E-08	1.38	10.0	Agreement
0637hrs	ZN-65	1.69E-07	4.18E-08	< MDA[5]			4.0	No Comparison
	SB-125	2.02E-07	6.33E-08	2.14E-07	7.49E-08	1.06	3.2	Agreement
	CS-134	1.54E-07	3.17E-08	1.62E-07	3.84E-08	1.05	4.9	Agreement
	CS-137	1.80E-07	3.16E-08	1.93E-07	3.23E-08	1.07	5.7	Agreement

1 These quantities are in the units of microcurie per milliliter.

2 Ratio = Licensee Value / NRC Value

3 Resolution = NRC Value / NRC Error (one standard deviation)

4 Result : The result of the comparison is based on the criteria in Attachment 1 and is expressed by the following:

Agreement	* = Criteria Relaxed
Disagreement	No Comparison

5 MDA = Minimum Detectable Activity

Table 3

Fermi 2 CST Discharge Activity Calculation
(April 15-17, 1994, Discharge)

Date of analysis: April 15, 1994

Volume(gallons) = 558360
(liters)= 2.114E+06

Flow Rates: (gallons per minute)
Dilution= 15600
CST dchg= 400

Isotope	EC[1] uCi/ml	Result[2] uCi/ml	Conc./EC[3]	Activity (mCi)
Mn-54	3.000E-05	5.635E-07	4.696E-04	1.191E+00
Co-58	2.000E-05	9.372E-08	1.171E-04	1.981E-01
Co-60	3.000E-06	4.146E-07	3.455E-03	8.763E-01
Zn-65	5.000E-06	1.687E-07	8.435E-04	3.566E-01
Sb-125	3.000E-05	2.020E-07	1.683E-04	4.270E-01
Cs-134	9.000E-07	1.541E-07	4.281E-03	3.257E-01
Cs-137	1.000E-06	1.795E-07	4.488E-03	3.794E-01
H-3	1.000E-03	3.260E-04	8.150E-03	6.890E+02
Totals[4]		3.278E-04		6.928E+02
(w/Dilution)[5]		8.194E-06	2.197E-02	

[1] Effluent concentrations for release to unrestricted areas as listed in 10 CFR 20, Appendix B, Table 2, Column 2.

[2] Result of gamma isotopic and tritium analyses of Condensate Storage Tank performed in NRC Region III Laboratory.

[3] Fraction of 10 CFR 20 effluent concentrations. This fraction is calculated as the concentration of effluent as it enters the lake, including the dilution flow.

[4] Total, undiluted activity from condensate storage tank.

[5] Totals with dilution credit from recirculation water.

ATTACHMENT 1

CRITERIA FOR COMPARING ANALYTICAL MEASUREMENTS

This attachment provides criteria for comparing results of capability tests and verification measurements. The criteria are based on an empirical relationship which combines prior experience and the accuracy needs of this program.

In these criteria, the judgement limits are variable in relation to comparisons of the NRC's value to its associated one sigma uncertainty. As that ratio, referred to in this program as "Resolution", increases, the acceptability of a licensee's measurement should be more selective. Conversely, poorer agreement should be considered acceptable as the resolution decreases. The values in the ratio criteria may be rounded to fewer significant figures reported by the NRC Reference Laboratory, unless such rounding will result in a narrowed category of acceptance.

RESOLUTION	RATIO = LICENSEE VALUE/ NRC REFERENCE VALUE
	AGREEMENT
< 4	NO COMPARISON
4 - 7	0.5 - 2.0
8 - 15	0.6 - 1.66
16 - 50	0.75 - 1.33
51 - 200	0.80 - 1.25
> 200	0.85 - 1.18

Some discrepancies may result from the use of different equipment, techniques, and for some specific nuclides. These may be factored into the acceptance criteria and identified on the data sheet.