U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-341/92021 (DRP)

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

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

Facility Name: Fermi 2

Inspection At: Fermi Site, Newport, Michigan

Inspection Conducted: December 8, 1992, through January 19, 1993

Inspectors: W. J. Kropp
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Approved By: M. P. Phillips, Chief
Reactor Projects Section 2B

1/22/93
Date

Inspection Summary

Inspection from December 8, 1992, through January 19, 1993
(Report No. 50-341/92021(DRP)
Areas Inspected: Routine, unannounced safety inspection by the resident inspectors of action on previous inspection findings; operational safety verification; cold weather preparation; onsite event follow-up; current material condition; housekeeping and plant cleanliness; radiological controls; security; regional requests; LERs; deviation event reports; maintenance activities; surveillance activities; onsite review; and engineering design packages.
Results: Of the fifteen areas inspected, no violations were identified. Two Unresolved Items were identified that pertained to unnecessary challenges to engineered safety features (paragraph 3.a), concerns with an LER (paragraph 5.a), and work planning (paragraph 6.a). In addition, four inspector followup items were identified that pertained to overflow of a phase separator tank (paragraph 3.f); operability of a test line valve (paragraph 5.b); testing of a fuel oil transfer pump (paragraph 6.b); and engineering calculation (paragraph 7.b). The following is a summary of the licensee's performance during this inspection period.

Plant Operations

The licensee's performance in this area was good. The operator's response to the failure of a relay in the high pressure coolant injection (HPCI) system was good. Also, the operator's response to abnormal noises identified in a condenser pump was conservative and well managed. However, the operators did not adequately monitor torus level during a surveillance to prevent HPCI pump automatic suction transfer to the torus.

Safety Assessment/Quality Verification

Overall, the licensee's performance in this area was excellent. Five of the licensee event reports (LER) were reviewed without any problems noted. However, concerns were identified with two LERs that pertained to root cause analysis (LER 341/92010) and the description of the event (LER 341/92008).

Maintenance and Surveillance

During this inspection period, the licensee's performance in this area was mixed. The team work exhibited between maintenance, operations, radiation protection, engineering and other licensee organizations during the maintenance outage to repair an extraction steam line was excellent. However, concerns were identified with work planning for other maintenance activities and with the testing of the emergency diesel generator fuel oil transfer pumps.

Engineering and Technical Support

The licensee's performance in this area was good. The onsite reviews conducted for assessing the restart of the unit after a refueling outage and for the scram on November 18, 1992, were thorough. Also, the system engineer and Inservice Inspection and Testing Group's decision to place an emergency equipment cooling water pump on increased testing frequency was conservative. There was concern identified with an assumption used in a calculation fc determining the minimum voltage expected to be seen by a motor operated valve.

DETAILS

1. Persons Contacted

Detroit Edison Company

* J. Conen, Senior Engineer, Plant Safety
* R. Eberhardt, Superintendent, Radiation Protection
* P. Fessler, Director, Nuclear Training
* J. Green, Superintendent, I&C
* E. Hare, Senior Compliance Engineer, Licensing
* J. Korte, Director, Nuclear Security
* A. Kowalczuk, Superintendent, Maintenance and Modifications
* R. McKeon, Plant Manager, Nuclear Production
* W. Miller, Superintendent, Technical Engineer
* R. Newkirk, General Director, Regulatory Affairs
* W. Orser, Senior Vice President, Nuclear Operations
* J. Plona, Superintendent, Operations
* L. Schuerman, General Supervisor, Plant Engineering
* A. Settles, Director, Nuclear Licensing
* R. Stafford, General Director, Nuclear Assurance
* F. Svetkovich, Superintendent, Radwaste
* R. Szkotnicki, Supervisor, Production Quality Assurance
* J. Tibai, Supervisor, Compliance
* G. Trahey, Director, Nuclear Material Management

*Denotes those attending the exit interview conducted on January 19, 1992.

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

2. Action on Previous Inspection Findings (92701)

a. (Closed) Open Item (341/90009-04(DRP)): The licensee had recommended a design change, Engineering Design Package (EDP) 11249, to improve accessibility of cabinets H11P627 and H11P626. The inspectors reviewed Work Request 000Z913527 and 000Z913528 that addressed the design changes specified in EDP 11249. The inspectors identified no concerns. This matter is considered closed.

b. (Closed) Open Item (341/90010-02(DRP)): The licensee committed to install a trip on the feed breaker to the recirculation pump motor/generator set to improve the reliability of the ARI/RPT system. The modification (EDP 5173) to add the trip has been scheduled for Refueling Outage, RF04. The inspectors have no further concerns in this area. This matter is considered closed.

c. (Closed) Inspection Followup Item (341/92017-03(DRP)): Simulator response comparison with actual plant response. The inspectors verified, via interviews with licensee personnel, that the simulator accurately reflected actual plant response to the November 18, 1992, loss of feedwater event and subsequent scram. Licensee personnel provided the inspectors with simulator charts and printouts that were generated as a result of inputting actual plant parameters and initial conditions from the November 18 event. The inspector's review of the items, and comparison of simulator response to actual plant data, verified that the simulator accurately modeled plant response. This item is considered closed.

d. (Closed) Inspection Followup Item (341/92017-04(DRP)): Control of fire doors. The inspectors verified Fire Doors RA2-7 and RA2-6 were in proper working order. The inspectors also reviewed the administrative controls for fire doors and have no further concerns in this area. This item is considered closed.

e. (Closed) Inspection Followup Item (341/92017-06(DRP)): Furtner NRC review of maintenance procedures 44.030.219 and 46.000.044. Procedure 44.030.219, used for a scale change out, required steps 6 through 13 to be completed if the "As Found" data was found out of tolerance. For Work Request 920902, steps 6 through 13 were performed even though the "As Found" data was acceptable. In discussions with the licensee, the inspectors determined that steps 6 through 13 were required regardless of the "As Found" data. This subject was discussed during a pre-job brief. Since this job was unique, no changes to the procedure were needed.

Work Request 920516 utilized procedure 46.000.044 calibrating the charcoal absorber temperature controller for the standby gas treatment system. While observing the work, the inspectors noted that the maintenance personnel lifted leads to defeat the lockout function when testing the control unit relays. The lifting of the leads was not delineated in the work request and procedure 46.000.044. The licensee has committed to change procedure 46.000.044 to address the lifting of the leads. Based on the above, this item is closed.

3. Plant Operations

The licensee completed a maintenance outage on December 13, 1992, when the unit was synchronized to the grid. The maintenance outage started on December 4, 1992, to repair a failure of the extraction steam line to the Number 4 North Feedwater Heater which had caused a runback of the recirculation pumps on December 1. Subsequent investigation by the licensee determined that the bellows between the extraction steam line and the low pressure turbine, along with a failure of the extraction steam line near an elbow, was the cause of the event. Other items worked during the outage included repairs to the moisture separator

reheater drain line bellows and repairs to the South Reactor Feed Pump Turbine High Pressure Isolation Valve. A drywell entry was also done to perform work on a drywell equipment drains sump pump. The unit has operated since December 13, 1992, at power levels up to 98 percent.

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:

- During performance of a High Pressure Coolant Injection (HPCI) Surveillance, 24.202.01, "HPCI Pump Time Response and Operability Test at 1025 PSI," the operators did not adequately monitor torus level to prevent an automatic HPCI pump suction transfer from the condensate storage tank (CST) to the torus. The suction of the HPCI pump during an engineered safeguards feature (ESF) actuation automatically switches from the CST to the torus if either a high level in the torus or low level in the CST occurs. To prevent unnecessary actuations of automatic suction transfer from the CST to the torus during the surveillance, procedure 24.202.01 had a statement in the "Precautions and Limitation" Section that required the torus level to be maintained below +1 inch. The operators did not maintain torus level below the +1 inch. Level was allowed to increase to +1.2 inches. Since the residual heat removal system was cooling the torus, expected wave motion in the torus caused a spike in the torus level to the 2 inch setpoint for the automatic HPCI suction transfer. The event was documented in Deviation Event Report (DER), 92-0647 (see paragraph 5.b of this report). This DER identified a near miss earlier in the year during the performance of the same surveillance. However, the operators tripped the HPCI pump prior to reaching the setpoint for the auto suction transfer

to the torus. The inspectors were concerned with the unnecessary challenges to ESF features caused by lack of adequately controlling plant conditions during surveillances. The inspectors will continue to monitor this area of operations. This matter is considered unresolved pending further NRC review (341/92021-01(DRP)).

- On December 25, 1992, during rounds, a nuclear power plant operator (NPPO) noticed abnormal noises associated with the North Condenser Pump. The pump was subsequently taken off line. Since the plant was at 98 percent power, the operators utilized pump curves and simulator information to ramp the unit to a power level commensurate for two condenser pump operation. The inspector considered the operator's response to the abnormal noises in the North Condenser Pump as prudent, conservative, and well planned.

b. Cold Weather Preparation (71714)

The inspectors completed review of the licensee's process to ready the unit for cold weather operations. The inspectors' review included direct observation of components or systems potentially affected by cold weather, log reviews to check for cold weather related problems, interviews with licensee personnel, and documentation review of the licensee's cold weather preparation procedure, NPP-27.000.04, "Freeze Protection Lineup Verification." No substantive concerns were identified as a result of the review. Safety-related as well as balance-of-plant (BOP) equipment and systems that would be sensitive to cold weather conditions appeared to have been adequately addressed by the licensee's procedures and preparations.

c. Onsite Event Follow-up (93702)

During the inspection period, the licensee experienced two events that required prompt notification of the NRC pursuant to 10 CFR 50.72. The inspectors pursued the events onsite with the licensee. The inspectors verified that any required notifications were correct and timely. The inspectors also verified that the licensee initiated prompt and appropriate actions. The specific events were:

- The failure of a relay in the high pressure coolant injection (HPCI) logic that could have prevented the HPCI turbine steam inlet valve, E4150F001, from opening either automatically or manually. As a result, the licensee declared HPCI inoperable at 5:11 p.m. (EST) on January 4, 1993. The relay was replaced and the HPCI system declared

operable at 3:30 a.m. (EST) on January 5, 1993. The inspectors will review the licensee's root cause and subsequent corrective action during the review of the associated LER.

- During performance of surveillance test on the HPCI system on January 14, 1993, the HPCI turbine steam control valve, E4150F068, failed to open. At 1:51 p.m. (EST), the HPCI was declared inoperable. Investigation by the licensee determined that a resistor box in a power supply for the governor control circuit failed. The resistor box was replaced and the surveillance test was successfully performed. The HPCI system was declared operable at 2:30 a.m. (EST) on January 15, 1993. The inspectors will review the licensee's root cause and subsequent corrective actions during the review of the associated LER.

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, 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. During a plant tour, the inspectors identified a packing leak on Reactor Core Isolation Cooling Valve E51-F054. Further investigation revealed that Work Request 000Z92150 had been initiated on May 9, 1992. The licensee stated that the work request was not worked during the recently completed refueling outage because the repairs required to stop the packing leak could be accomplished during power operations. However, the licensee stated the work should have been considered as part of the leak reduction program. Overall, the inspectors considered the material condition of the plant during this inspection period as satisfactory.

e. Housekeeping and Plant Cleanliness (71707)

The inspectors monitored the status of housekeeping and plant cleanliness for fire protection and protection of safety-related equipment from intrusion of foreign matter. Housekeeping was considered above average. However, there were several areas in the plant that required increased licensee attention.

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.

During the backwashing of Reactor Water Cleanup (RWCU) Demineralizer "A," level indication was lost to the phase separator tanks. Level was lost when the radwaste computer interface system malfunctioned. Shortly after losing tank level, the backwashing of RWCU Demineralizer "A" was secured. The licensee subsequently determined that overflow of a phase separator tank had occurred which resulted in contamination around Residual Heat Removal (RHR) Pump "A." The overflow line for the phase separator tanks was routed to a reactor building sump in the RHR Pump "A" Room. The seal line for RHR pump "A" was routed to the overflow line. The licensee suspects that overflow of a phase separator tank occurred resulting in contaminated water being routed from the overflow line to the RHR pump "A" seal line and eventually to the room. Resin was found in the area of the contamination. The licensee has initiated an investigation of the event. Pending review of the licensee's investigation, this matter is considered an Inspection Followup Item (341/92021-02(DRP)).

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.

No violations or deviations were identified.

4. Regional Request (92701)

a. Types of Insulation in Drywell

The inspectors requested information from the licensee pertaining to types of insulation used in the drywell. Also included in the information was the configuration and dimensions of safety related intake strainers located within the suppression pool as well as pump flow rates associated with each strainer. The licensee furnished the inspectors with the requested information. The information was then sent to the Region III Office.

b. Spent Fuel Pool Storage

The Region III office requested information pertaining to availability of storage space for complete core offload for spent fuel. The licensee stated that there would be enough space to completely offload the core until the year 2000. The licensee has considered short term actions to extend the time that included new high density fuel racks and other alternative designs.

5. Safety Assessment/Quality Verification (40500 and 92700)

a. Licensee Event Report (LER) Follow-up (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 (341/89017) Revision 1: The analysis of the postulated feedwater line break in the steam tunnel described in the Updated Final Safety Analysis Report (UFSAR) was completed in the early 1970s but documentation could not be retrieved. A new analysis was conducted with the results submitted in a UFSAR update submitted in March 1990.

(Closed) LER (341/92007) Revision 1: During periodic leak rate testing of primary containment isolation valves and penetrations, twenty-eight valves exceeded the administrative allowable leakage rate. As a result, the combined leakage of the twenty-eight valves exceeded the limits as defined in the Technical Specifications (TS). The valves were reworked to eliminate or reduce any leakage.

(Closed) LER (341/92008): On September 19, 1992, during the performance of surveillance 24.307.13, "Emergency Diesel Generator No. 14-ECCS Start and Load Rejection Test," Bus 65F was inadvertently deenergized. The loss of Bus 65F resulted in numerous engineered safety features actuations and the loss of shutdown cooling. The inspectors had the following concerns with the LER:

- LER 92008 did not fully describe how shutdown cooling was lost when Bus 65F was deenergized. The logic for the Shutdown Cooling Valve, E4150F008, was powered by Modular Power Unit (MPU)-2. MPU-2's normal power source was from a motor control center (MCC) supplied from Bus 65F. MPU-2's alternate power source was a MCC supplied from Bus 65E. Therefore, as a result of the loss of Bus 65F, the logic for Shutdown Cooling Valve, E4150F008, was lost and the valve

went shut. MPU-2 was scheduled for maintenance on September 22, 1992, with temporary power supplying the MPU-2 loads. Also, at the time of the event, maintenance on the Division II Reactor Protection System (RPS) was in progress which required the use of the alternate power supply for Division II of the RPS. The normal power source was Bus 65E and the alternate was Bus 65F. Therefore, as a result of the loss of Bus 65F, the RPS Division II was deenergized which caused the E11-F008 Shutdown Cooling Valve to close on a Group 4 Isolation Signal. With the closure of E11-F008, the primary backup decay heat removal system (Division I, Residual Heat Removal) and the licensee's alternate method of decay heat removal (fuel pool clean & cooling system) were lost. The inspectors were concerned that the licensee's investigation of this event did not assess the event from a shutdown risk perspective for possible lessons learned.

- LER 92008 did not identify that the failure to repeat back surveillance steps was a contributing factor to the event. The licensee's investigation of the event did identify that the surveillance test leader, a nuclear supervising operator (NSO), did not demand precise, formal communication be utilized during the surveillance. Discussion with the licensee determined that even though the LER did not specifically identify inadequate repeat backs as a contributing factor, the training specified in the LER as part of the corrective action addressed the importance of repeat backs, including the importance of demanding precise and formal communications by NSOs.

- Onsite Safety Review Organization (OSRO) reviewed LER 92-008 during the OSRO meeting on October 14, 1992. The OSRO recommended approval with no unreviewed safety questions. The inspectors were concerned that OSRO did not pursue the loss of shutdown cooling and address possible impact on the shutdown risk program for possible lessons learned.

- The LER stated both Shutdown Cooling Valves, E4150F008 and E4150F009, went shut during the event, when in fact only valve E4150F008 went shut.

These concerns are considered an unresolved item pending further NRC review of licensee actions. (50-341/92021-07(DRP))

<u>(Closed) LER (341/92009)</u>: Eight safety relief valves (SRV) failed to actuate within specified TS limits during surveillance testing and one SRV failed to actuate. The SRV pilot assemblies have been replaced by refurbished and certified spare assemblies.

(Closed) LER (341/92010): The licensee issued a voluntary licensee event report when, on September 18, 1992, station header pressure dropped to a level that resulted in the isolation of the noninterruptible air supply (NIAS). The control air compressors automatically started and supplied the NIAS loads as designed. The NIAS system was actuated when Modular Power Unit 3 (MPU-3) was de-energized as part of a planned evolution for preventive maintenance (PM). When MPU-3 was de-energized, Valve P50-401 closed and isolated the station air loads. This was expected by the operators. However, at the time the MPU-3 was de-energized, the radwaste control room operator notified the Control Room Nuclear Supervising Shift Operator (CRNSO) that radwaste tank level indication instrumentation was lost. Steps were taken to reduce the radwaste inflow when the Nuclear Assistant Shift Supervisor (NASS) heard the radwaste operator communicate with the CRNSO. Also, the CRNSO directed a nuclear power plant operator (NPPO) to bypass the station air header isolation valve, P50-F041, by opening Manual Bypass Valve P50-F535 to restore radwaste tank level. The action resulted in a decrease in station header pressure since the running West Station Air Compressor was unable to maintain header pressure on the bypass valve. Also, the central air compressor did not auto start because the auto start logic was also powered by MPU-3. Station air header pressure eventually decreased to a point where the NIAS system isolated and the control air compressors auto started to supply NIAS loads.

The LER identified the root cause of the event as personnel error with inadequate communication between shift personnel. The inspectors did not agree with the licensee's conclusion. The failure of Abnormal Operating Procedure NPP 20.129.01, "Loss of Stations and/or Control Air," to identify radwaste loads affected by a loss of station air contributed to inadequate work planning. If there was adequate work planning, the plant evolution which involved inflow to radwaste, would have been terminated prior to deenergizing MPU-3 or the PM on MPU-3 could have been postponed until the radwaste activities requiring tank level indication were completed. The inspectors agree that personnel error contributed to the actuation of the NIAS system. Since, the licensee's corrective action to revise procedure NPP 20.129.01, "Loss of Station and/or Control Air," to identify radwaste loads affected by a loss of station air, should improve work planning for future MPU-3 outages, the inspectors have no further concerns with this LER.

(Closed) LER (341/92011): During a Reactor Pressure Vessel leakage test, (water solid), and a pressure of 1045 psig, an engineered safety feature actuation occurred due to invalid reactor water level 3, 2, and 1 signals. The invalid signals were caused by the bumping of an instrument rack by a radiation protection technician. At the time of the event, the plant was in cold shutdown and control rods were fully inserted.

(Closed) LER (341/92012): Manual Scram of the reactor by the Control Room Nuclear Supervising Operator when reactor pressure vessel water level decreased on loss of feedwater. The loss of feedwater was caused by personnel error, when the wrong valve was opened during the backwashing of a condensate polishing demineralizer.

b. Deviation Event Reports

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. Deviation Event Reports were also reviewed to ensure that they were generated appropriately and dispositioned in a manner consistent with the applicable procedures. During the review of DERs, the inspectors had one observation. Deviation Event Report 92-0647 issued on November 6, 1992, documented the inadvertent auto transfer of the high pressure injection coolant (HPCI) suction to the torus during the performance of surveillance 24.202.01, "HPCI Pump Time Response and Operability Test at 1025 PSI." The HPCI pump was on recirculation to the condensate storage tank (CST), the normal discharge path during the surveillance. When torus level was allowed to increase above the +1 inch level during the surveillance, the Torus Suction Valves for HPCI, E4150F042 and E4150F041 opened as designed when the high torus level auto suction transfer setpoint of 2 inches was reached. The setpoint was reached due to expected waves in the torus caused by the operation of residual heat removal (RHR) pumps in the torus cooling mode. When the HPCI torus suction valves opened, the HPCI pump discharged to the CST Outboard Test Line Isolation Valve, E4150F008, and the Downstream Test Return Valve, E4150F011, started to close per design. However, Valve E4150F008 did not close completely which allowed approximately 1800 gallons per minute (gpm) to continue to flow to the CST. The 1800 gpm flow to the CST prevented the opening of the HPCI Recirculation Valve, E4150F012, to torus. Valve E4150F012 would have opened when HPCI discharge flow dropped to less than 1200 gpm. The HPCI system was designed to realign to inject into the reactor pressure vessel (RPV) upon a valid engineered safety system (ESF) signal when the HPCI system was in the test mode. However, the portion of the surveillance procedure 24.201.01 performed on November 6, 1992, was an 18 month Technical Specification (TS) surveillance that verified the HPCI system would realign to inject into the RPV on a RPV low level ESF signal. To prevent an undesirable injection into the RPV, the auto open feature of the HPCI Injection Valve, E4150F006, was defeated. Therefore, the HPCI system was not capable of automatically realigning to the emergency mode if an ESF signal was received during the performance of surveillance 24.202.01 on November 6, 1992. The inspectors had no concern with the method of testing the HPCI system, during the 18 month TS surveillance, to verify the HPCI would realign within the required

time response with a low level in the RPV. However, the TS surveillance performed every 92 days to verify proper HPCI pump flow would be performed without defeating the auto opening feature of the HPCI Injection Valve, E4150F006. Thus, the design of the HPCI system to realign to the emergency mode upon a valid ESF signal would occur. Deviation Event Report 92-0647 did not adequately address the operability of the HPCI system during future TS surveillances, performed every 92 days to verify adequate HPCI pump flow requirements. Since the cause of the failure of the Outboard Test Line Isolation Valve, E4150F008, to close had not yet been determined, the DER should have clearly addressed the operability of the HPCI system during future test modes when the 92 day TS surveillance was performed. The licensee subsequently revised the DER to clearly state the basis for operability of HPCI in the test mode. Discussion with licensee personnel determined that the ability of Valve E4150F008 to close would be resolved prior to the performance of future HPCI TS surveillances. The ability of Valve E4150F008 to isolate the test line to the CST during future HPCI Technical Specification surveillances is considered an inspection followup item pending NRC review of completed DER 92-0647 (341/92021-03(DRP)).

No violations or deviations were identified.

6. Maintenance/Surveillance (62703 & 61726)

a. Maintenance Activities (62703)

Routinely, station maintenance activities were observed and/or 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 and reviewed:

- Y274920616 Solenoid Replacement
- 000Z923405 Troubleshoot/Repair Division I CCHVAC Precipitater Power pack

- A546920611 Inspect/Test 490V MCC and Aux Relays, Lube Blower and Motor, Check Belts and Ionizing Wires

- T099920516 Replace CCHVAC Division I Chiller Compressor B009 Solenoid Purge Valve

- 000Z92150 Repair packing leak on valve E51-F054

- 000Z920627 Repair or replace valve on purge unit on CCHVAC Div. I compressor

- A566920303 Perform annual inspection (CCHVAC)

The inspectors had the following observations pertaining to work planning:

- The Division I Outage for the Control Center Heating, Ventilation, and Air Conditioning (CCHVAC) System commenced on January 5, 1993, without corrective maintenance work activity, 000Z920627, being ready for issuance to the field. This work package was a significant work activity for the CCHVAC Division I Outage.

- While observing work on preventive maintenance activity A546920611, the inspectors noted that not all the required tools and materials were pre-staged (grease gun and lubrication).

- Step 4 of Work Request T099920516, which required placement of a "nitrogen cap" on the CCHVAC chiller unit, was deleted and placed in Work Request A566920303.

Even though the above observations did not result in increased Division I CCHVAC outage time, the inspectors were concerned with the lack of effective work planning for a division outage that had been scheduled for several weeks. Based on the above and the inadequate work planning which caused the actuation of NIAS (LER 341/92010), the effectiveness of work planning is considered an Unresolved Item pending further review by the NRC (341/92021-04(DRP)).

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 portions or reviewed the following surveillances:

- 20.000.19 Shutdown from Outside the Control Room
- 24.202.01 HPCI Pump Time Response and Operability Test at 1025 PSI
- 24.202.04 HPCI System Suction Valve Auto Transfer
- 24.207.008 Division II EECW Pump and Valve Operating Test
- 24.207.02 EECW Pump and Valve Operability Test
- 24.307.14 Emergency Diesel Generator 13 Start and Load Test
- 24.307.34 DGSW, DGOT and Starting Air Operability Test - Emergency Diesel Generator No. 11
- 24.630 Remote Shutdown Instrument Channel Checks
- 44.030.151 HPCI System Logic Functional Test

During performance of 24.207.02, "EECW Pump and Valve Operability Test - Division I" on December 4, EECW Pump "A" differential pressure (dp) appeared to fall into "High dp alert" range. Pump discharge and suction pressures were consistent with data from prior satisfactory test runs. Due to problems encountered with the sonic flow measurement device, comparison of the pump flow dp to the pump curves yielded a dp that was 102.1 percent of the pump curve. All other system parameters were normal and consistent with prior runs. Conservative action was taken by the system engineers to declare the pump to be in the high alert range and the system was put on an increased frequency for testing. The licensee has experience some difficulties establishing a steady consistent flow reading on the sonic flow measurements device and is currently evaluating methods to resolve the problem.

During the review of Surveillance Procedure 24.307.14, "Emergency Diesel Generator (EDG) 13 Start and Load Test," the inspectors had the following observations:

- Each EDG has two fuel oil transfer pumps (A and B) with each one having 100 percent capability to supply the EDG day tank with enough diesel fuel oil for full load operation. Technical Specification (TS) 4.8.1.1.2.a.3, requires that every 31 days the licensee verifies a fuel transfer pump start and transfers fuel from the storage system to the day fuel oil tank. The design of the EDG fuel oil system allows

for selecting, by means of a diesel fuel oil transfer (DFOT) pump selector switch, the DFOT pump (A or B) that would auto start on any start of the EDG. The DFOT pump that was not selected would be on standby and would start automatically on low day fuel oil level. Step 5.2.1.0 of procedure 24.307.14, required that the transfer DFOT pump selector switch be repositioned to the opposite pump at the start of the surveillance. Therefore, this method of testing results in testing each DFOT pump once every 62 days. The inspectors had a concern that if a DFOT selected to auto start was taken out of service (OOS) for maintenance, the associated EDG could be required to be started to verify that the other DFOT would auto start and supply fuel to the day fuel oil tank. The test on the non-OOS DFOT pump could be required because Technical Specification requirements for verifying DFOT starts might not have been performed within the preceding 31 days (plus 25 percent). The inspectors expressed this concern to the licensee. The licensee issued a temporary change to EDG surveillance procedure that requires testing of both DFOT pumps during the monthly TS surveillance.

- To ascertain the standby start feature of the DFOT pump on a low day fuel oil tank level, the inspectors reviewed 24.307.14, "DGSW, DFOT and Starting Air Operability Test - EDG 11." Procedure 24.307.14 did not verify that the DFOT pumps for EDG 11 would auto start on low day fuel tank. Subsequent discussion with the licensee determined that procedure 46.000.118, "Magnetrol Level Switches - Displacer Type," calibrates the level switch for the EDG day fuel tanks every 36 months. The calibration verifies, by draining the day fuel oil tank, that the level switch contacts close at the required day fuel tank level to auto start the standby DFOT. Also, the level switch contacts that close for low day fuel oil tank level alarm, was verified to close at the required low level. However, the complete loop was not tested because the relay and associated contacts, which close in each of the DFOT pump control circuits to start the pumps on low level, were not verified to energize when a low level was reached. Also, the annunciator in the main control room, for low day fuel oil tank level, was not verified to actuate when the associated level switch contact was closed on reaching the low level setpoint for the day fuel oil tank. This matter is considered an Inspector Followup Item pending further NRC review (341/92021-05(DRP)).

No violations or deviations were identified.

7. Engineering & Technical Support (37700)

a. Onsite Review

The inspectors reviewed the following OSRO meeting minutes:

- 489E Restart
- 489G Restart
- 490A Restart
- 492C Scram

Based on the review of the above OSRO Meeting Minutes, the inspectors considered the OSRO meetings to be thorough and discussed the necessary relevant pivots for the subjects. The inspectors considered the OSRO process effective in controlling plant restarts after refueling outages. Also, the OSRO Meeting Minutes (492C) that documented the investigation into the cause of the scram on November 18, 1992, was thorough with a good discussion on probable causes of the scram.

b. Engineering Design Package

The inspectors reviewed Engineering Design Package (EDP) 1364 that pertained to replacing the 60 foot pounds (ft-lbs) direct current (DC) motor for Valve E4150F008 with a 80 ft-lb motor. Valve E4150F008, the Test Line Return Valve to the condensate storage tank (CST) for high pressure coolant injection (HPCI), would be opened during surveillances of the HPCI system. The normal position for this valve was closed. The Updated Final Safety Analysis Report (UFSAR), Section 6.3.2.2.1, states that during tests, a signal to initiate HPCI automatically stops the test mode and starts injection to the reactor via the feedwater line. This transfer from the test mode to the accident mode requires the automatic closure of valve E4150F008. The inspectors reviewed Design Calculation (DC) 4943, Revision C, in package EDP-13464. This calculation was updated to demonstrate that the minimum voltage seen by the 80 ft-lb motor would be greater than 80 percent of the motor rated voltage. The minimum battery voltage used in this calculation was based on the minimum voltage expected four hours after a design base accident (DBA). The calculated battery voltage at four hours was 235 Vc. Using 235 Vc, the calculation concluded that the minimum voltage seen by the motor was 80.5 percent of the motor's rated voltage. The HPCI system could be in the test mode upon initiation of a DBA. Since the UFSAR states that HPCI was designed to realign from the test mode to the accident mode, Valve E4150F008 could be open at the beginning of the DBA. Therefore, the use of a minimum battery voltage based on four hours after the initiation of a DBA might

not be a valid assumption to use in a calculation to determine the minimum voltage to be seen by Valve E4150F008. This matter is considered an inspector followup item pending further review by the NRC and the licensee (341/92017-06(DRP)).

No violations or deviations were identified.

8. 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 is discussed in paragraphs 3.f, 5.b, 6.b, and 7.b.

9. 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., 5.b., and 6.a.

10. Meetings and Other Activities

a. Management Meetings (30702)

On January 19 and 20, 1993, Mr. Wayne Shafer, Chief, Branch 2 toured the Fermi Plant and met with licensee management to discuss plant performance and plant material condition.

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 January 19, 1993. 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.