U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-341/85-15(DRP)

Docket No. 50-341 License No. DPPR-87

Licensee: Detroit Edison Company
6400 North Dixie Highway
Newport, MI 48166

Facility Name: Fermi Energy Center - Unit 2

Inspection At: Fermi Energy Center, Newport, MI

Inspection Conducted: March 4-6, 1985

Inspectors: S. DuPont

S. Guthrie

S. Stasek

T. Tongue

Approved By: N. J. Chrissotimos
Team Leader

3-21-85
Date

Inspection Summary

Inspection on March 4-6, 1985 (Report No. 50-241/85-15(DRP))
Areas Inspected: Special, announced team inspection of safety-related systems and functions through comparison of the Final Safety Analysis Report, proposed Technical Specifications, and as-built configurations. The inspection involved a total of 137 inspector-hours onsite by five NRC inspectors including 37 inspector-hours onsite during offshifts.
Results: No items of noncompliance or deviations were identified.

DETAILS

1. Persons Present at Exit

Detroit Edison Company

F. Agosti, Manager, Nuclear Operations
W. Colbert, Director, Nuclear Engineering
G. M. Trahey, Director, Nuclear QA
R. S. Lenhart, Superintendent, Nuclear Production
*E. Preston, Jr., Operations Engineer
*A. Wegele, Compliance Engineer
G. Debner, Senior Startup Engineer
*J. H. Plona, Technical Engineer
*J. E. Conen, Engineer
L. B. Collins, Systems Engineer
M. K. Deora, Systems Engineer
L. F. Wooden, Systems Engineer

Nuclear Regulatory Commission

N. J. Chrissotimos, Section Chief
T. Tongue, Senior Resident Inspector, Dresden
S. Guthrie, Senior Resident Inspector, Big Rock Point
P. M. Byron, Senior Resident Inspector, Enrico Fermi 2
S. G. Dupont, Regional Inspection Specialist
M. Parker, Resident Inspector, Enrico Fermi 2
S. Stasek, Resident Inspector, Dresden

*Denotes those persons contacted during the inspection.

2. Independent Inspection Effort - Comparison of FSAR, Proposed Technical Specifications, and As-Built Configurations

The inspection was conducted to compare selected safety-related systems, components and structures as described in the Final Safety Analysis Report (FSAR) with the final draft Technical Specifications and as-built configurations for likeness and compatibility.

In addition, existing procedures for the systems were reviewed for comparison of setpoints, etc., to assure that the FSAR and Technical Specifications were appropriately addressed. The following systems listed with FSAR Section and Technical Specification were reviewed.

Core Spray System (CSS)

FSAR Section 6.3.2.2.3 and FSAR Table 6.3-12.B.(B.2); Technical Specifications 3.5.1.a and 4.5.1.b.1.

Low Pressure Coolant Injection (LPCI) Mode of Residual Heat Removal (RHR) System

FSAR Section 6.3.2.2.4 and FSAR Table 6.3-12.B.(B.1); Technical Specifications 3.5.1.b, 4.5.1.b.2, and 4.5.1.c.1.

High Pressure Coolant Injection System (HPCI)

FSAR Section 6.3.2.2.1 and FSAR Table 6.3-12.B.(B.3); Technical Specifications 3.5.1.C, and 4.5.1.b.3. It was noted by the inspector during his review that an apparent inconsistency existed between the FSAR and the Technical Specification concerning the reactor low water level initiation setpoint for HPCI. The licensee, it was found, had already addressed the inconsistency and currently had an FSAR change request in place to remedy it.

Emergency Equipment Cooling Water (EECW) System

FSAR Sections 6.3.2.2.6 and 9.2.2.2; Technical Specifications 3.7.1.2 and 4.7.1.2.

Standby Liquid Control System

FSAR Section 7.4.1.2; Technical Specifications 3.1.5 and 4.1.5.

Safety Relief Valves (S/RV's)

FSAR Section 5.5 and Technical Specification 3/4.4.2.

Main Steam Isolation Valves (MSIV's)

FSAR Sections 5.5 and 10.3.2 and, Technical Specification 3/4.4.7.

Containment Systems

FSAR Section 6.2 and Technical Specification 3/4.6.2.3 and Technical Specification Table 3.6.3-1 (Primary Containment Isolation Valves). Technical Specification Table 3.6.3-1 lists automatic, remote-manual, manual and other isolation valves by function and isolation groups. The inspector compared the list with the FSAR sections as stated and verified closure times. The inspectors walked down and verified flow paths and actual installation of about 75% of the 243 valves listed in the table.

The following related systems were reviewed:

Suppression Pool Cooling

Technical Specification 3/4.6.2.3.

Vacuum Relief - Suppression Chamber - Drywell Vacuum Breakers

Technical Specification /4.6.4

Vacuum Relief - Reactor Building to Suppression Chamber Vacuum Breakers

Technical Specification 3/4.6.4.2

Reactor Protection System

FSAR Section 7.2 and FSAR Table 7.2-1; Technical Specification Table 2.2.1-1.

The inspector verified that the trip setpoints contained in Technical Specification Table 2.2.1-1 were also listed within the applicable surveillance procedures by verifying that instrument output values (Voltage DC or AC or milliamperes) correlated to the Technical Specification setpoints and by reviewing instrumentation specification sheets for the following functions:

Turbine Stop Valve - Closure

Main Steam Line Isolation Valve - Closure

Intermediate Range Monitor - Neutron High Flux

Main Steam Line Radiation - High

Reactor Vessel Steam Drain Pressure - High

Reactor Vessel Low Water Level - Level 3

Scram Discharge Volume - High Water Level

Drywell Pressure - High

Electrical Power Systems

The inspector reviewed Technical Specifications Section 3/4.8.1 through 3/4.8.4, Electrical Power Systems, to determine if the commitments set forth in the FSAR are accurately reflected in the requirements of the Technical Specifications.

The inspector verified that the requirements for separate and independent onsite A.C. electrical power sources were met by inspecting the four emergency diesel generators (EDG) and their electrical distribution switchgear. The inspector verified that the fuel system availability and capacity requirements of Technical Specification 3.8.1.1.b and 3.8.1.2.b were met. In addition, the inspector verified the presence of other systems and components not required by Technical Specifications but essential for the operability of the diesel generators including lube oil, air start, cooling water, service water, local control panels, and 130 VDC control power. The inspector verified the availability of manual controls and instrumentation in the control room including the "EDG remaining loading capability" and EDG load instrumentation required by operators in loss of offsite power situations. The inspector verified that procedures were in place to perform the surveillances required by Technical Specifications to verify operability of the EDGs.

The inspector verified the availability of the D.C. electrical power sources, including batteries and chargers, required by Technical Specifications 3.8.2.1 and 3.8.2.2. The surveillances necessary to verify the operability of the batteries were determined to be in place.

Through review of drawings and visual inspection of switchgear components as installed, the inspection determined that the A.C. and D.C. power distribution system requirements of Technical Specification 3.8.3.2 were satisfied and that surveillance requirements of section 4.8.3.2.1 and 4.8.3.2.2 were addressed in operational procedures.

By visual inspection of installed switchgear, the inspector verified the availability of power to pumps and heat trace cables for the standby liquid control system required by Technical Specification Table 3.8.4.5-1.

For the Emergency Core Cooling Systems (ECCS) and other safety-related systems required to bring the plant to a safe shutdown condition and/or mitigate the effects of accidents, the inspector verified that the required switching and distribution components and necessary control room instrumentation for major electrical components were in place. The components included pumps in the Core Spray, Residual Heat Removal, and Residual Heat Removal Service Water Systems. Instrumentation for operator verification of operation of the High Pressure Coolant Injection system and for control room indication of the valve position of valves in the Automatic Depressurization system.

During the review of Technical Specification 3.8.1.1, which requires a minimum of "two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution systems," the inspector discussed with the applicant his view that the electrical distribution system as constructed does not appear to meet the requirements of Technical Specifications or Criterion 17 of 10 CFR 50, Appendix A. Guidance for implementation of these requirements are provided in Regulatory Guides 1.93, Availability of Electric Power Sources, and 1.32, Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants, which is consistent with IEEE Std. 308-1974. These documents describe requirements for two sources of offsite power from the transmission network to the onsite distribution system. Regulatory Guide 1.32, Section C.1.a, describes a preferred design that would include two immediate access circuits, and notes that designs that substitute a delayed access circuit for one immediate access circuit conforms to Criterion 17.

The inspector's concern is based on the physical separation of incoming sources of offsite power when those sources reach the Class 1E distribution system. The Class 1E system is separated into two divisions which are physically isolated from each other. Division I (Busses 64B and 64C) is fed by one incoming line which joins the transmission network at transformer no. 64 and derives power from the 120 KV system. Division II (Busses 65E and 65F) is fed by one incoming line which joints the 345 KV system at the secondary winding of transformer no. 65. Both the 120 KV and the 345 KV systems are each fed by several lines from the Detroit Edison distribution network. The inspector in his analysis only considered the incoming line from the point where it joins the transmission network as described above, in keeping with the specifications of Criterion 17. Fermi 2 does not have a unit auxiliary transformer. During operation it is not possible to connect either power source to a division other than the one it normally serves. The inspector expressed his view that the present configuration actually provides two separate and independent Class 1E distribution systems, presently designated Divisions I and II, each with its own single source of power and each unable to be fed by a second immediate access or delayed access circuit.

The applicant acknowledges that except at very low power operation, a loss of either offsite source would almost certainly result in a plant scram. A loss of either offsite source would result in a half scram signal and an automatic start of the EDG(s) serving the dead bus(es). A loss of the 345 KV line at rated power would result in loss of recirc. pump bus 65G, and a loss of the 120 KV line would result in a major reduction in main condenser circulating water. Either situation imposes an undesirable transient on the unit, unnecessarily challenges safety systems, and challenges the EDG(s).

The inspector questioned the conclusions of the Safety Evaluation Report, Section 8.1, prepared by the staff of NRR. This issue was discussed telephonically with the office of Nuclear Reactor Regulation on March 5, 1985, in which they confirmed that this design was acceptable and met Criterion 17 of 10 CFR, Appendix B. The absence of a unit transformer does not imply that fast transfer to offsite power sources is not required, but rather that it is not possible. The immediate access of the entire 1E distribution system to two sources of preferred (offsite) power is not provided to offset the lack of unit transformer availability because of the separation of incoming sources.

The applicant committed to provide the inspector with data they believe will address the concerns expressed, by demonstrating that this question has already been analyzed. The applicant agreed to provide this information within 90 days of the date of this report or prior to commencement of commercial operation, whichever occurs first. The inspector will review the applicant's submittal.

Following review of the information, the inspector will determine if any further followup is required (50-341/85-15-01(DRP)). This open item does not effect the current license activities.

3. Open Items

Open items are matters which have been discussed with the licensee, which will be reviewed further by the inspector, and which involve some action on the part of the NRC or licensee or both. Open items disclosed during the inspection are discussed in Paragraph 2.

4. Exit Interview

The inspectors met with licensee representatives (denoted in Paragraph 2) at the conclusion of the inspection on March 6, 1985, and summarized the scope and findings of the inspection activities.

After discussions with the licensee, the inspectors have determined there is no proprietary data contained in this inspection report.