NUREG-0798
Supplement No. 4

Safety Evaluation Report
related to the operation of Enrico Fermi Atomic Power Plant, Unit No. 2

Docket No. 50-341

Detroit Edison Company, et al.

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

September 1984

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ABSTRACT

Supplement No. 4 to the Safety Evaluation Report related to the operation of the Enrico Fermi Atomic Power Plant, Unit 2, provides the staff's evaluation of additional information submitted by the applicant regarding outstanding review issues identified in Supplement No. 3 to the Safety Evaluation Report, dated January 1983.

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1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

The "Safety Evaluation Report Related to the Operation of Enrico Fermi Atomic Power Plant, Unit No. 2" (NUREG-0798) (SER), prepared by the staff of the Nuclear Regulatory Commission (staff), was issued on July 10, 1981. The SER provided a summary and results of the staff's radiological safety review of the application by the Detroit Edison Company (applicant) for an operating license for Fermi-2. The SER concluded that on favorable resolution of outstanding matters described therein, the plant could be operated without endangering the health and safety of the public.

Supplements 1, 2, and 3 to the SER provided: (1) the staff's evaluation of additional information provided by the applicant regarding outstanding review issues identified in the SER; and (2) the staff's evaluation of additional information provided by the applicant regarding revised designs. Supplement 1 also provided the staff's response to the comments in the report by the Advisory Committee on Reactor Safeguards (ACRS).

By Amendments 45 through 58 to the Final Safety Analysis Report (FSAR) and by letters identified in Appendix A to this supplement, the applicant has provided additional information, including information regarding several of the outstanding issues identified in Supplement 3 to the SER.

This supplement (Supplement 4 to the SER) provides the staff's evaluation of additional information provided by the applicant in FSAR amendments through Amendment 58 and by letters identified herein.

Each section and appendix of this supplement is designated and titled the same as the corresponding section or appendix of the SER that has been affected by the additional evaluation. Except as noted, each section is supplementary to the corresponding section in the SER. Appendix A to this supplement is a continuation of the chronology of principal actions related to the staff's safety review of the application. The NRC licensing project manager for the review of the Fermi-2 operating license application is Mr. M. David Lynch. Mr. Lynch may be contacted by calling (301) 492-7050 or writing:

Mr. M. David Lynch
Division of Licensing
Nuclear Regulatory Commission
Washington, DC 20555

This SER is a product of the NRC staff. NRC staff members who were principal contributors to this report are identified in Appendix G.

A number of consultants assisted the staff in the review. The organization which provided consultants to the staff is listed below. The individual consultants are listed in Appendix G.

Idaho National Engineering Laboratory (EG&G Idaho, Inc.)

Appendix K contains our consultant's report on their evaluation of the Fermi-2 inservice testing program for pumps and valves.

Copies of this supplement are available for public inspection at the Commission's Public Document Room at 1717 H Street, NW, Washington, DC, and at the Monroe County Library System, 3700 South Custer Road, Monroe, Michigan 48161. They are also available for purchase from the sources indicated on the inside front cover.

1.8 Summary of Outstanding Issues

1.8.1 Prelicensing Issues

The partial or complete resolution of some of the outstanding issues identified in Supplement 3 to the SER is described in appropriate sections of this supplement. The outstanding issues remaining in the staff operating license review are listed below together with the number of the appropriate section in the SER or in Supplement 1, 2, 3, or 4 to the SER, in which the issues are discussed, including the status, and plans for their resolution. We will complete our review of these items before the operating license is issued. The resolution of these outstanding items will be discussed in a future supplement to the SER.

(1) Seismic reassessment of piping systems attached to torus (SER Supplement 3, Section 3.7.3)

(2) Mark I containment analyses (SER Supplement 3, Section 3.8.1)

(3) Seismic and dynamic qualification of equipment (this supplement, Section 3.10)

(4) Environmental qualification of safety-related electrical equipment (SER Supplement 2, Section 3.11; SER Supplement 3, Section 3.11)

(5) Hydrodynamic loads in the control rod drive system (this supplement, Section 4.6.2)

(6) Emergency preparedness (Supplement 3, Section 13.3; this supplement, Section 13.3)

(7) TMI Issues (Supplement 3, Section 22)

   (a) I.C.7 NSSS-vendor review of procedures (SER)
   (b) II.B.3 Chemistry procedures for postaccident sampling (SER Supplement 3)
   (c) II.D.1 Testing of safety/relief valves (SER and SER Supplement 3)
   (d) II.E.4.2 Containment isolation dependability (SER and SER Supplement 3)

1.8.2 License Conditions

In our review of outstanding issues identified in the SER and in Supplements No. 1 through 3, we have resolved three of the issues that were identified in Supplement 3 as license conditions. The license conditions remaining in our licensing review are listed below, with the number of the appropriate section in the SER or in Supplement 1, 2, or 3 to the SER in which we discuss the license condition.

(1) Modifications to piping and equipment attached to Mark I containment (SER Supplement 1, Sections 3.10 and 18; SER Supplement 3, Section 3.8.1).

(2) Environmental qualification of equipment (SER Supplement 2, Section 3.11).

(3) Hydrodynamic stability analysis (SER Section 4.4.1).

(4) Study of multiple control system failures (SER Section 7.2.2).

(5) Modifications to fire protection equipment (SER Supplement 2, Section 9.5.1).

(6) Low-pressure turbine-disc inspection (SER Section 10.2.2).

(7) Retention of persons with BWR operating experience on shift until 100% power is achieved (SER Section 13.1; SER Supplement 1, Section 18).

(8) Implementation of safeguards contingency plan, guard training plan, and physical security plan (SER Supplement 2, Section 13.5).

(9) Final procedure for postaccident sampling (SER Supplement 2, Section 22, Item II.B.3).

(10) Instrumentation for detection of inadequate core cooling (SER Section 22, Item II.F.2; SER Supplement 1, Sections 18 and 22, Item II.F.2).

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, AND COMPONENTS

3.9 Mechanical Systems and Components

3.9.6 Inservice Testing of Pumps and Valves

In the SER which we issued in July 1981, we stated that a detailed review of the Fermi-2 inservice testing (IST) program for pumps and valves had not been completed. Therefore, we performed a preliminary review of the IST program and in Supplement 1, granted interim relief from certain pump and valve testing requirements of Section XI of the ASME Boiler and Pressure Vessel Code required by Sections 50.55a(g)(2) and 50.55a(g)(4)(i) of 10 CFR Part 50 for that portion of the 120-month period until we completed our detailed review. Our detailed review of the Fermi-2 IST program for pumps and valves is now complete; this report contains an evaluation of the IST program proposed in Revision 2 to Detroit Edison Company Report No. DET-16-0201, "Inservice Testing Program, Enrico Fermi Atomic Power Plant Unit 2." We reviewed the Fermi-2 IST program in accordance with the guidelines in Section 3.9.6 of the Standard Review Plan (NUREG-0800, July 1981).

The applicant will test the pumps and valves within the scope of the Fermi-2 IST program in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1980 Edition, through the Winter 1980 Addenda, except for certain items where the applicant has requested relief from the testing requirements of the code. We have reviewed these relief requests with the following findings:

a. The applicant has requested specific relief from measuring the inlet pressure, differential pressure, and flow rate for the diesel fuel oil transfer pumps in accordance with the requirements of Section XI of the code. The applicant's basis for requesting this relief is that it has not installed instrumentation to measure either the inlet pressure or the flow rate for these pumps. In addition, the applicant states that it is unable to calculate the pump flow rate and the differential pressure as the present system is designed. It is our position that without the measurement of these parameters, the applicant cannot adequately monitor the hydraulic characteristics of these pumps and, therefore, detect possible pump degradation.

   Accordingly, the applicant's requested relief from the requirements of Subsection IWP of Section XI of the ASME Code is denied. We require the applicant to modify the diesel fuel oil transfer system to permit the measurement of the inlet pressure, the differential pressure, and the flow rate of the diesel fuel oil transfer pumps in accordance with the requirements of Section XI of the code.

b. The applicant has requested specific relief from the requirement to evaluate the stroke times of all active Category A and Category B solenoid operated valves in compliance with the requirements of Section XI of the code. The applicant's basis for requesting relief is that it is impractical to apply the requirements of IWV-3413(b) to valves with short stroke times (i.e., less than five seconds) and has proposed to verify that the stroke time does not exceed five seconds. We do not agree with the applicant's basis for requesting relief from the stroke time measurement requirements of the code. It is our position that rapid-acting valves are defined as those valves with stroke times of two seconds or less and that valves with stroke times greater than two seconds should be tested in accordance with the appropriate requirements of Section XI.

Accordingly, the applicant's requested relief from the requirements of Subsection IWV of Section XI of the ASME Code is denied. We require that the applicant designate as rapid-acting valves only those active Category A and Category B solenoid operated valves which have stroke times of two seconds or less. We require that valves which do not fit this category should be tested for stroke times in accordance with the requirements of Section XI of the ASME Code.

c. The applicant has requested specific relief from exercising valves C11-115 and C11-138 which are control rod drive charging water header and cooling water header check valves for each of the 185 hydraulic control rod units, in accordance with the requirements of Section XI of the code. As an alternative, the applicant has proposed verifying closure of these check valves during individual control rod scram insertion testing which will be performed in accordance with the requirements of the Technical Specifications. The applicant's basis for requesting relief is that proper operation of each of these check valves is demonstrated during scram testing. Specifically, the applicant's position is that if a particular control rod drive scram insertion time is less than the limit specified in the Technical Specifications, the check valves are functioning properly. The NRC staff does not agree with the applicant's basis for requesting relief for the check valves since proper valve closure cannot be assured unless the control rod drive charging header and cooling water header are depressurized.

   Accordingly, the applicant's requested relief from the exercising requirements of Subsection IWV of Section XI of the ASME Code for check valves C11-115 and C11-138 is denied. We require that the applicant test these valves in conformance with the appropriate sections of the ASME Code.

d. The applicant has requested specific relief from exercising valves T48-F001A and T48-F001B, which are the isolation valves for the water supply to the cooler in the combustible gas control system, in accordance with the requirements of Section XI of the ASME Code. As an alternative, the applicant has proposed full-stroke exercising of these valves during the combustible gas control system operability tests which are performed every six months in accordance with the requirements of the Technical Specifications. The applicant's basis for requesting relief is that there is no manual means of stroking the valves which automatically open upon initiation of the combustible gas control system. However, the applicant has not submitted any technical justification for not performing the system operability tests on a quarterly basis to demonstrate proper operability of these valves. We do not agree with the applicant's basis for requesting relief since testing of the valves can be performed quarterly in accordance with the frequency specified by the ASME Code.

Accordingly, the applicant's requested relief from the exercising requirements of Subsection IWV of Section XI of the ASME Code for isolation valves T48-F001A and T48-F001B is denied. We require that the applicant test these valves in conformance with the appropriate sections of the ASME Code.

e. The applicant has requested specific relief from the exercising requirements and stroke time measurement requirements of Section XI of the code for primary system safety-relief valves (SRVs) B21-F013E, B21-F013H, B21-F013J, B21-F013P and B21-F013R. As an alternative, the applicant has proposed exercising these valves once every 18 months in accordance with the requirements of the Technical Specifications. The applicant also proposes to observe changes in steam flow and/or turbine bypass valve positions to ensure that these valves have stroked in less than or equal to, five seconds. Although we do not agree with the applicant's basis for requesting relief from testing of the valves quarterly in accordance with the frequency specified by the ASME Code, there are other safety-related reasons for exercising these valves only at 18-month intervals. Specifically, if these valves were to fail to reclose after testing, the plant effectively would be placed in a loss-of-coolant accident (LOCA) condition. In addition, a recent study (BWR Owners Group Evaluation of NUREG-0737, Item II.K.3.16, "Reduction of Challenges and Failures of Relief Valves") recommends that the number of SRV valve openings be reduced as much as possible.

   Based on these considerations, we grant relief to the applicant to exercise these valves once every 18 months as specified in paragraph 4.5.1.d.2 of the Fermi-2 Technical Specifications.

f. The applicant has requested specific relief from exercising check valve E41-F045 in the high pressure coolant injection suction line from the suppression pool, in accordance with the requirements of Section XI of the ASME Code. As an alternative, the applicant has proposed disassembly of this valve to verify freedom of disc movement during each refueling outage to demonstrate proper valve operability until sufficient data is accumulated to justify an inspection interval between tests longer than each refueling outage. The applicant's basis for requesting relief is that normal system tests utilize the condensate storage tank for pump suction rather than the suppression pool. Taking suction from the suppression pool during testing is undesirable since this water is not demineralized and thus the entire water inventory of the suppression pool and condensate storage tank would have to be processed after the test. We agree with the applicant's basis for requesting relief for check valve E41-F045.

   Accordingly, the applicant's requested relief from the exercising requirements of Subsection IWV of Section XI of the ASME Code for check valve E41-F045 is granted. However, we require that the applicant provide us with the results of its inspections before any inspection interval between tests longer than each refueling outage, can be accepted.

We find that it is impractical within the limitations of the Fermi-2 plant design, geometry, and accessibility for the applicant to meet certain requirements of Subsections IWP and IWV of Section XI of the ASME Code. Imposing these requirements would result in hardships or unusual difficulties without a compensating increase in either the level of quality or safety. Therefore, pursuant to Section 50.55a of 10 CFR Part 50, the relief requested by the applicant from the pump and valve testing requirements of Sections 50.55a(g)(2) and 50.55a(g)(4)(1) of 10 CFR Part 50 is granted for the initial 120-month period of the IST program except for those items identified above. Our detailed discussion of these additional matters is contained in Appendix K to this report.

In summary, we find that the IST program proposed by the applicant for the Fermi-2 safety-related pumps and valves is acceptable and in conformance with the Commission's regulations, except for Items (a), (b), (c), and (d) discussed above.

3.10 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment Important to Safety

In Supplement No. 3 to the SER, we identified both generic as well as equipment specific items of concern that remained to be resolved. Since then, we have further reviewed the previously submitted information provided in the applicant's submittal dated March 18, 1982. In response to our concerns arising from our further review of its submittal of March 18, 1982, the applicant provided additional information to resolve all outstanding matters affecting the seismic and dynamic qualification of equipment important to safety in its letters dated April 7, June 10, and October 4, 1983. Our review indicates that all items are resolved to our satisfaction, though two items require confirmation. The results of our review and the two remaining confirmatory items are summarized below.

In Supplement No. 3, we identified six items (i.e., Items (2)(a) through (2)(f)) which were to be resolved by the applicant no later than three months prior to the scheduled fuel load date. These items are discussed using the prior numbering system of Supplement No. 3.

Item (2)(a)

The qualification reassessment of mechanical equipment using a floor response spectra corresponding to a five percent structural damping and an equipment damping of five percent or lower was completed satisfactorily as documented in the applicant's submittal of March 18, 1982. This item is now resolved.

Item (2)(b)

We evaluated those pieces of safety-related equipment which were already installed but required hardware modifications because of reevaluation and qualification activity related to the seismic qualification program. The only equipment which fell into this category was the engine instrument panel for the emergency diesel generators in the residual heat removal complex. Field Modification Request (FMR) 4287 was issued by the applicant to implement the necessary fix; the fix has been implemented. We find that this item is now resolved.

Item (2)(c)

Work on confirming that the acceleration values used in qualifying valves are consistent with those used in the as-built piping analysis, is in progress. The applicant has indicated that some initial problem of exceedance in valve accelerations had been encountered as a result of the piping analysis. Accordingly, the applicant has committed to provide us with a summary report describing the status of the confirmatory process as well as the approach which will be taken to resolve of any such exceedance. This commitment is found acceptable to the staff. The applicant subsequently did submit a report but was unable to qualify one of the valves; a further submittal will be made prior to fuel load. We will review this report when it is submitted and report on its acceptance in a future supplement to the SER. While we consider this matter resolved, we require the applicant to submit this confirmatory report prior to fuel loading.

Item (2)(d)

For equipment installed or qualified after the Seismic Qualification Review Team (SQRT) audit, some SQRT forms have been provided for our review and found acceptable. The applicant has committed to submit any remaining required SQRT forms. We find the applicant's commitment on this matter to be acceptable. On this basis, we find this item resolved. We will review these remaining SQRT forms, if any, and report on our acceptance in a future supplement to the SER.

Item (2)(e)

With respect to the installation work on the RHR mechanical draft cooling tower, the applicant states that this work has been completed. We find that this item is now resolved.

Item (2)(f)

The applicant also states that the installation of the hydraulic control unit attached piping has been completed. We find that this item is resolved.

In Supplement No. 3 to the SER, we stated that our review of the seismic and dynamic qualification of torus-attached equipment was waiting for a submittal from the applicant. The applicant subsequently submitted a report on this matter attached to its letter dated June 10, 1983. In this report, the applicant presented the results of its reevaluation of the qualification of both mechanical and electrical equipment which were subject to suppression pool hydrodynamic loads. The torus attached piping systems which terminate at the equipment consist of the residual heat removal (RHR), core spray, high pressure coolant injections (HPCI), and reactor core isolation cooling (RCIC) systems. The equipment considered in the reevaluation includes pumps and valves of various types, including electrical equipment such as electrical torus penetrations and thermocouples.

Based on our review of the applicant's methods of reevaluation and the acceptance criteria for this equipment, we find that both the structural integrity and operability of this equipment have been addressed. In all cases, the equipment was found to be qualified under the combined effects of seismic load and the hydrodynamic loads associated the suppression pool. We, therefore, conclude that the issue of equipment supported by the torus-attached piping system is resolved.

We stated in Supplement No. 3 to the SER that the applicant provided its response in its letter of October 11, 1982, to our July 30, 1982, request for additional information concerning long-term operability of deep draft pumps. The specific information we requested was:

(a) A general description of the pump design and the performance characteristics of these pumps as well as their natural frequencies, either estimated by analysis or measured by tests.

(b) The approach taken to assure initial alignment and shaft straightness during installation.

(c) The details of how the vibration data base will be established in light of the guideline provided in Division I of ASME Code Section XI.

(d) The acceptance test procedure prior to full power operation as well as after repair and reassembly.

(e) The in-service surveillance test procedure, including the vibration monitoring instruments to be used and their location, as well as an explanation of how pump degradation can be detected by this program.

Based on our review of the applicant's responses to our request cited above and on the information contained in the applicant's earlier submittals of August 16, 1979, October 25, 1979, and November 23, 1981, responding to IE Bulletin 79-15, "Long-Term Operability of Deep Draft Pumps," we feel that the long-term operability program as adopted by the applicant is in compliance with the procedure which we have recently suggested for use in detecting problems with deep draft pumps. Based on the above findings, we conclude that the issue of long-term operability of deep draft pumps for the Fermi-2 facility is now resolved.

3.10.2 Summary of Evaluation Findings

Based on our review of the applicant's overall program for the seismic and dynamic qualification of equipment important to safety, we find that the applicant has resolved all issues related to this program and, accordingly, that this program is acceptable. However, there are three specific matters for which we require the applicant to submit confirmatory information prior to fuel load. These are:

a. Provide a summary report for our review prior to the fuel load, confirming that the values of valve acceleration are acceptable.

b. Submit any remaining required SQRT forms for equipment qualified after the SQRT audit.

c. Additionally, we require the applicant to submit justification for interim operation for any safety-related equipment which cannot be completely qualified prior to fuel load.

4 REACTOR

4.2 Fuel System Design

4.2.3 Design Evaluation

Seismic and LOCA Loadings

In our SER issued in July 1981, we stated that the issue of the fuel assembly dynamic response under blowdown conditions which could occur in the reactor pressure vessel for a postulated loss-of-coolant accident (LOCA), remained as a confirmatory item. Since then, we have reviewed and approved the General Electric Company topical report, NEDE-21175-3. This report describes an analytical method for evaluating the combination of seismic and LOCA loads. We have reviewed the plant specific liftoff movement and vertical acceleration under these combined loads (i.e., 2.8 g) and found that the vertical liftoff motion (i.e., 0.015 inches) is insignificant. For this motion, there would be very little change in reactivity. We, therefore, conclude that the applicant has satisfactorily resolved this confirmatory issue.

Control Blade Stress Corrosion Cracking

In our SER issued in July 1981, we discussed the generic problem of cracking of the control rod blade tubing due to stress corrosion and the subsequent loss of boron carbide from some of the cracked tubes which were examined. Our concern regarding the loss of boron carbide by leaching from cracked control blade tubing is addressed in IE Bulletin No. 79-26, Revision 1, which requires operating BWRs to perform various actions including, but not limited to, tests to determine shutdown margins. In Supplement No. 3 to the SER, we identified a licensing condition which would require a surveillance program to determine and quantify stress corrosion cracking and boron depletion. (Refer to Item 4 in Section 1.8.2 of Supplement No. 3 to the SER.)

In its letter dated June 25, 1984, the applicant proposed a surveillance program for Fermi-2 which includes a plan to inspect and replace control blades when necessary. The applicant's criteria to replace control blades when the need arises due to boron depletion, is consistent with the intent of IE Bulletin 79-26, Revision 1. The applicant has incorporated this proposed surveillance program into the plant operating guidelines and procedures.

Based on the applicant's proposed control blade surveillance program which we find to be consistent with IE Bulletin 79-26, Revision 1, we conclude that our concern regarding control blade stress corrosion cracking and boron depletion has been resolved in an acceptable manner. Accordingly, we are removing surveillance of the control rod blades as a licensing condition for Fermi-2.

5 REACTOR COOLANT PRESSURE BOUNDARY

5.2 Integrity of the Reactor Coolant Pressure Boundary

5.2.2 Overpressurization Protection

In our SER, we stated that the applicant's commitment in its letter dated June 4, 1981, to a two-year maintenance period for the safety-relief valves (SRV), was acceptable. However, the applicant later modified its commitment on this matter in its letter dated September 15, 1983. The applicant now proposes that:

a. Fifty percent of the SRVs will be removed from service and tested and serviced at any given refueling outage (nominally 18 months).

b. The remaining 50 percent of the SRVs will be tested during the subsequent refueling outage.

The applicant's proposal on this matter is consistent with the BWR Owners Group response to TMI Item II.K.3.16 which we found to be acceptable. Accordingly, we find the revised SRV maintenance interval proposed by the applicant in its letter dated June 4, 1981, to be acceptable.

6 ENGINEERED SAFETY FEATURES

6.3 Emergency Core Cooling System

6.3.4 Evaluation Findings

6.3.4.1 Safety Concerns Associated with Pipe Breaks in the BWR Scram System

In the SER we issued in July 1981 and in Supplements 1 and 3 to the SER, we discussed our concern regarding a postulated break in the control rod drive scram discharge volume. This safety concern was discussed in detail in NUREG-0803, "Generic Safety Evaluation Report Regarding the Integrity of BWR Scram System Piping," which is our generic safety evaluation of postulated pipe breaks in BWR scram systems. In Supplement No. 1 to the SER, we concluded that the recommendations of NUREG-0803 should be satisfactorily resolved before an operating license for Fermi-2 is issued.

In Supplement No. 3 to the SER, we reported on the applicant's response to the unresolved issues identified in NUREG-0803. In its response, the applicant concluded that the probability of core damage due to a postulated failure in the scram discharge volume (SDV) is sufficiently low that further environmental qualification or design modifications to mitigate the consequences of a postulated pipe break are not necessary. The applicant adopted by reference, a probabilistic risk assessment (PRA) contained in the GE topical report NEDO-24342, of the SDV piping system as justification for this conclusion. We also reported in this supplement that this PRA was currently under review and that we would address the acceptability of the applicant's response to the recommendations in NUREG-0803 in a future supplement to the SER.

The BWR Owners Group has since submitted additional informaton regarding this PRA in the GE Topical Report NEDO-22209 which updates the probabilistic approach presented in NEDO-24342 and which presents probabilistic arguments as an alternative to the NRC staff's criteria in NUREG-0803. We conclude from our review of NEDO-22209 that resolution of our concern regarding a postulated SDV break requires more detailed consideration of the applicable pipe break mechanisms than can be obtained by a probabilistic analysis. Accordingly, we have requested further specific information of the BWR Owners Group regarding a deterministic fracture mechanics evaluation of the scram system piping, including a request for discussions of the associated realistic leak rate, leak detection and mitigation capability. We are currently reviewing the BWR Owners Group responses to our request for additional information and will provide a generic evaluation that identifies any additional design requirements developed as a result of our review. (There is one item regarding radiation exposure as a result of routine tests and inspections which is plant specific and which will be evaluated on a case-by-case basis following completion of the generic evaluation.)

Since this is a multi-plant action item, we have not made a determination as to what design changes, if any, are necessary for the Fermi-2 facility. This determination will not be made until our review of the BWR Owners Group responses is complete. We conclude, therefore, that our requirement for satisfactory resolution of the NUREG-0803 recommendations prior to the issuance of an operating license as first stated in Supplement No. 1 to the SER, should be eliminated. After the issuance of the generic SER containing the resolution of our concerns in NUREG-0803, the applicant will be required to make any changes necessary to protect the health and safety of the public.

7 INSTRUMENTATION AND CONTROLS

7.3 Engineered Safety Feature Systems

7.3.2 Specific Findings

Reactor Vessel Water Level Sensing Line Failures

In Supplement No. 3 to the SER, we discussed our concern that the postulated failure of an instrument tap or sensing line in conjunction with a single electrical failure could defeat the automatic initiation of certain safety systems. We identified this in Item (8) of Section 1.8.1 of Supplement No. 3 as an outstanding issue. The basis for our concern on this matter is that operating reactor experience indicates that a number of failures have occurred in BWR reactor vessel level sensing lines. In some cases, these sensing line failures resulted in erroneously high reactor vessel level indication. Since the Fermi-2 design includes instrument sensing lines common to feedwater control system sensors, reactor trip sensors, and engineered safety feature sensors, we reviewed the Fermi-2 design to determine whether the failures cited above could adversely affect the operation of the Fermi-2 facility. We first expressed our concern regarding the consequences of common sensing line failures in our letter to the applicant dated September 21, 1982.

We subsequently requested additional information regarding the design of the Fermi-2 reactor vessel level measurement instrumentation in our letter to the applicant dated September 12, 1983. Specifically, we requested the applicant to provide an analysis of the consequences for each case where the postulated failure of a reactor vessel level tap or instrument sensing line, concurrent with an additional random electrical failure, could induce a transient and preclude the automatic operation of the reactor protective systems.

In response, the applicant provided the requested information in a series of letters dated November 16, 1982; September 23, 1983; and April 23, 1984. Based on its review of the Fermi-2 design, the applicant identified the instrument reference leg sensing lines common to both the feedwater control and the protective system sensors. For each identified common line, the applicant performed an analysis to determine the consequences of a sensing line failure concurrent with additional single failures in the protective channels not dependent on the failed sensing line.

The "worst-case" failure path identified by the applicant from this analysis is initiated by a failure of the Division I vessel level instrument reference line combined with a random failure of the B21-N091D level transmitter indicating a "high" water level. The immediate consequences of such concurrent postulated failures would be a feedwater trip and a main turbine stop valve closure due to a false reactor vessel signal indicating a high water level (level 8). Although the turbine steam bypass system would continue to operate, the reactor would be tripped via the turbine stop valve closure input to the reactor trip system. Additional consequences of these postulated failures would include:

(1) failures of both the automatic high pressure coolant injection (HPCI) initiation and of the automatic reactor core isolation cooling (RCIC) initiation on low reactor vessel level (level 2); (2) unavailability of the HPCI manual initiation capability; (3) failure of the automatic recirculation pump trip on low reactor vessel level (level 2); and (4) failure of the automatic depressurization system (ADS) initiation, automatic low pressure coolant injection (LPCI) initiation, and automatic core spray (CS) initiation on low reactor vessel level (level 1).

As stated above, the immediate systems response to the postulated instrument line break includes a feedwater trip, a turbine trip, and a reactor trip with a resultant decrease in vessel water level. Alarms would annunciate at the level-4 and level-3 setpoints as the vessel water level decreased. However, the automatic initiation of the emergency core cooling systems (ECCS) and the RCIC would be precluded by the postulated failure, as well as the capability to manually initiate HPCI. To prevent the reactor fuel from being uncovered, operator action would be required. For the purpose of this analysis, the applicant has assumed no operator action for ten minutes following the reactor scram. Following manual initiation of the RCIC at ten minutes into the postulated accident scenario, the water level stops decreasing and slowly starts increasing. No fuel failures would occur and the core would remain covered at all times.

Failure paths identified by the applicant which could affect other low reactor vessel level (level 3) circuits, are less limiting than the "worst case" discussed above. Postulated failures of other level circuits result in an automatic reactor trip via the new anticipated transient without scram (ATWS) mitigating system, and either automatic HPCI or automatic RCIC initiation on low reactor vessel level (level 2).

Although such postulated failures do not necessitate operator action to initiate protective systems, successful automatic termination of the transient is dependent on the ATWS mitigating system and, in some cases, the RCIC system. The latter system is not classified as an emergency core cooling system (i.e., safety-related).

Following the accident at Three Mile Island, both we and industry focused attention on various improvements to enchance safety. The BWR vessel level measurement systems were identified as an area with the potential for cost-effective improvement. In January 1982, the BWR Owners Group (BWROG) met with the NRC staff to discuss BWR vessel level instrumentation systems. At this meeting, the BWROG agreed to evaluate the need and desirability of design changes to level measurement systems to supplement procedural modifications. In accordance with this agreement, the BWROG prepared and submitted a report on vessel level measurement systems entitled "Review of BWR Reactor Vessel Water Level Measurement System," SLI-8211, July 1982. Based on their review of the vessel level measurement system, the authors of SLI-8211 identified areas of concern and recommended design improvements. The improvements recommended in SLI-8211 for plants with design features such as those in the Fermi-2 facility included a modification of the protection systems logic to lessen reliance on operator action in mitigating the transient resulting from a postulated rupture or break in an instrument sensing line. However, the applicant has chosen not to propose any modifications to implement the recommendations of the BWROG report, SLI-8211. At this time, based on our review of the vessel level instrumentation system at Fermi-2, we believe that modification of the protection systems logic would be a cost-effective measure which would enhance safety. Although the recommended design improvements in SLI-8211 were proposed by the BWROG, the promulgation of regulatory requirements to implement such modifications requires a comprehensive NRC staff review. Our review of this issue is being addressed within the Generic Issue Management Control System (Generic Issue No. 50).

Based on our review of the applicant's analysis of the consequences of instrument sensing line postulated failures, we find that although certain postulated failures would require operator action, the operator has sufficient time and available information to diagnose the problem and initiate corrective action. In addition, we find that the applicant has administrative procedures to aid and direct the operator in the event of a sensing line failure. Therefore, we find that the Fermi-2 level measurement system design is acceptable. If following the completion of our review of the recommendations contained in the SLI-8211 report, we find that modifications to the protection systems logic are required, the implementation of these modifications will be addressed in a future supplement to the SER.

9 AUXILIARY SYSTEMS

9.5 Fire Protection, Communication, Lighting, and Emergency Diesel Engine Systems

9.5.1 Fire Protection

In Supplement No. 3 to the SER, we stated that we had not completed our review of the additional information submitted by the applicant in Amendment No. 44 to the FSAR regarding the addition to the radwaste building to be used for interim storage of low-level radwastes. We have now completed our review of the fire protection features of this addition. Our evaluation is summarized below.

Introduction

The onsite storage facility for low-level radwastes is in a separate building (i.e., the radwaste building). The walls, floor, and ceiling of this on-site facility are either reinforced concrete or concrete block. Door openings to the radwaste building and the rooms housing the asphalt storage tank and pumps have Class A, three-hour rated fire doors. Fire-detection equipment in this building is designed to annunciate and alarm locally in the control rcom of this on-site storage facility. An automatic sprinkler system which conforms with National Fire Protection Association (NFPA) Std. 13 storage area and the control room, where the amount of in-situ combustibles is negligible. A manual hose station with enough hose to reach all areas in the facility is located in the truck-bay areas. The heating, ventilating and air-conditioning (HVAC) system for the on-site storage facility will automatically shut down on sensing smoke in the outside air supply and exhaust air ducts.

The rooms housing the asphalt storage tank and pumps are separated by three-hour fire-barrier walls and doors, even though no safety-related equipment is located there. They are also protected by an automatic fire-detector and sprinkler system. The HVAC systems for these rooms are completely separate from the rest of the storage facility, with no interactions possible; fusible links automatically shut down the entire HVAC systems in the rooms in the event of a fire. Smoke detectors are provided in all areas of the facility except for the radwaste storage areas, where all combustibles are contained in sealed steel drums.

We recommend in Section C.7.n of BTP CMEB 9.5-1 that radwaste and decontamination areas be provided with fire barriers, automatic fire suppression and detection and ventilation controls. In this regard, we note the following four design features have been proposed by the applicant.

a. Three-hour rated fire barriers are provided for all areas to separate the radwaste storage facility from other areas of the Fermi 2 facility.

b. Automatic sprinkler protection is provided for all areas of the facility except the building control room and the empty drum storage area. We have evaluated the combustible loading in the two non-sprinklered areas and find that the fuel load is negligible. In case of a fire in this area, the release of radioactive materials would be a very small fraction of 10 CFR Part 20. We, therefore, conclude that the addition of automatic sprinklers would not significantly enhance the level of fire protection.

c. Smoke detection capability is provided in all areas of the facility except for the radwaste storage areas. The applicant has chosen to not provide detectors in these areas due to potential problems of detector insensitivity resulting from the elevated radiation field in this localized area. We have evaluated the fire hazard in these areas and find that combustibles are contained in sealed steel drums. Automatic sprinklers, which provide an alarm are provided. We, therefore, conclude that the addition of smoke detectors would not significantly enhance the level of fire protection.

d. Ventilation isolation controls are provided which automatically shut down the HVAC upon detection of smoke in the outside air supply and exhaust air ducts.

Based on our evaluation above, we conclude that the level of fire protection provided for the new addition to the radwaste storage facility for interim storage of low-level radwastes is in accordance with our guidelines in paragraph C.7n of BTP CMEB 9.5-1, and is, therefore, acceptable. We find that this matter is now resolved.

13 CONDUCT OF OPERATIONS

13.3 Emergency Preparedness Evaluation

13.3.1 Introduction

The staff's initial evaluation of the Fermi, Unit 2, Radiological Emergency Response Preparedness Plan, Revision 1, dated June 1982, was provided in Section 13.3 of Supplement No. 3 to the SER. We identified 20 items for which additional information and commitments were required from the applicant for resolution of these matters. In addition, we requested additional information regarding the Fermi-2 emergency preparedness program in a letter to the applicant dated July 22, 1983. The applicant's responses were provided in Revision 2 to the Fermi-2 emergency plan and in the procedures implementing this emergency plan, both of which were submitted in September 1983; additional information was submitted in a letter to the NRC dated February 23, 1984.

We have reviewed the additional information provided in Revision 2 to the Fermi-2 emergency plan, and the supplemental information in the emergency plan implementing procedures, against the same requirements and guidance criteria identified in Supplement No. 3; namely, Section 50.47 of 10 CFR Part 50, Appendix E to 10 CFR Part 50, and NUREG-0654/FEMA-REP-1, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," dated November 1980. The results of our review are discussed in Section 13.3.2 of this report.

Findings and determinations on the adequacy of offsite emergency preparedness for the Fermi-2 facility have been provided to the NRC by the Federal Emergency Management Agency (FEMA) in several reports; the latest of these reports is dated July 18, 1983. The FEMA findings and determinations are discussed in Section 13.3.3 of this report. Additional supplemental findings have been requested by the NRC in a letter to FEMA dated March 28, 1984.

A two-week onsite evaluation to assess the applicant's capability to implement the emergency plan was conducted by the NRC during the period October 11-21, 1983. The results of that appraisal are contained in Inspection Report No. 50-341/83-24, dated November 28, 1983.

13.3.2 Evaluation of the Emergency Plan

This evaluation addresses those items identified in Supplement No. 3, which required resolution. The order of presentation and the numbering of the sections corresponds to the listing of the items in Section 13.3.2 of Supplement No. 3.

13.3.2.1 Assignment of Responsibility (Organization Control)

There were two items in this category requiring resolution. For the first, we requested the applicant to identify the agencies with emergency plan responsibilities in the ingestion exposure pathway (i.e., within 50 miles) Emergency Planning Zone (EPZ) and to provide a map which clearly illustrates the States, provinces and cities within the ingestion exposure EPZ. In its letter to the NRC dated February 23, 1984, the applicant identified the State of Ohio, the State of Michigan and the Province of Ontario, Canada, as the agencies with responsibilities for emergency plans in the ingestion exposure pathway EPZ. The applicant states that this information will be included in Revision 3 of the Fermi-2 emergency plan. A map of the ingestion exposure EPZ is shown in Figure A-2 of Revision 2 to the Fermi-2 emergency plan. Based on the information provided by the applicant, and the applicant's commitment to revise the plan accordingly, we find that this item has been satisfactorily resolved. We will confirm the applicant's implementation of this matter in a future supplement to the SER.

For the second item related to the assignment of responsibility in the event of an emergency, we requested the applicant to identify the agency or agencies responsible for notifying the appropriate Canadian officials of an emergency at the Fermi-2 facility and to describe the arrangements made to notify these Canadian officials.

In response, the applicant stated in Section A.2 of Revision 2 to the Fermi-2 emergency plan that the Department of State Police, the lead agency for emergency response in the State of Michigan, is responsible for notifying and providing periodic information updates to the Province of Ontario, Canada, through the Ministry of the Solicitor General in Toronto, Canada. In addition, the applicant states in emergency plan implementing procedure EP-290, "Emergency Notification," that upon classification of an Alert or higher emergency, the onsite emergency organization will notify the Sandwich West Police Station in Canada. Based on our review of the information in the emergency plan and procedures, we find that the applicant has provided an acceptable response to this item.

13.3.2.2 Onsite Emergency Organization

We requested the applicant to provide information on the shift staffing augmentation capability at the Fermi-2 site. The applicant was also requested to provide, if there were significant differences in the Fermi-2 plan from the staffing objectives of Table B-1 of NUREG-0654, justification for these differences.

In response, the applicant indicated in Section B.1.2 of Revision 2 to the Fermi-2 emergency plan that it intends to comply with the 30-minute and 60-minute augmentation criteria of Table B-1 of NUREG-0654 as a goal for staffing the emergency response facilities. The applicant also states that during off-hours, on the average, 60 minutes is required to staff the key emergency response positions. On this basis, we find that the applicant has provided an acceptable response to this item in that the applicant's shift staffing augmentation objectives, as reflected in the Fermi-2 emergency plan, meet the guidance contained in both NUREG-0654 and in Supplement 1 to NUREG-0737, "Clarification of TMI Action Plan Requirements," December 1982.

13.3.2.4 Emergency Classification System

We identified four items in this section of Supplement No. 3 which required resolution by the applicant. They are discussed in the same order as in Supplement No. 3.

The applicant was requested to provide the specific instrument readings and other indicators which are to be used as emergency action levels (EAL's) in the emergency classification system. In addition, we transmitted to the applicant detailed comments on the EAL's in our letter dated July 22, 1983.

The applicant provided a response to this item in the EAL's contained in Section D of Revision 2 to the Fermi-2 emergency plan and in the implementing procedure EP-101, "Classification of Emergencies". We found in our review of the emergency classification system that, in general, the EAL's have been substantially revised in response to our comments and that the majority of the specific instrument readings and other indicators have been provided. However, the applicant states in Revision 2 that a number of EAL's will be provided at a later date. These are the EAL's related to high main steam line radiation, high off-gas activity at the steam jet air ejector, containment high-range radiation monitor readings, and a high containment radiation reading as verified by a portable instrument reading.

In its submittal to the NRC, dated February 23, 1984, the applicant states that calculations were currently in progress to determine the specific radiological monitoring readings which will be used in the EAL's. The applicant has also committed to incorporate these EAL's into implementing procedure, EP-101, prior to fuel load. Based on a review of the revised EAL's in the emergency plan and procedures and the applicant's commitment to incorporate the remaining radiological EAL's into the procedures prior to fuel load, we find this item to be resolved in an acceptable manner. We will confirm the applicant's implementation of this matter in a future supplement to the SER.

We requested the applicant to correlate the containment high-range radiation monitor and other key instrument readings, if applicable, to a range of degraded core conditions. Specifically, we requested that the applicant include selected values from this analysis along with other indicators of core and containment conditions in the EAL's and that these be used to initiate protective actions in accordance with the guidance in NUREG-0654, Appendix 1, for general emergencies.

In response to our request on this matter, the applicant has incorporated the concept of utilizing core indicators, containment indicators, and other plant system indicators as EAL's in the emergency classification scheme to classify events and to initiate protective actions. Information on the containment high-range radiation monitors (CHRRMs) was provided in the applicant's submittal of February 23, 1984. Redundant monitors are being installed at elevation 605 in the drywell about seven feet from the reactor shield wall. The monitors, which meet the requirements of Item II.F.1 of NUREG-0737, measure the radiation levels resulting from nuclides emitting gamma radiation in the vicinity of the detectors.

The applicant has determined the CHRRM readings, in units of rem/hour, for various fractions of the core inventory of radioactive noble gases and radioiodine assumed to be airborne in the drywell. This inventory contains a source term consisting of 100 percent of the noble gases and 25 percent of the radioiodine assumed to occur in the event of a loss-of-coolant accident (LOCA). Selected CHRRM readings have been incorporated into the emergency classification scheme as EAL's, thus enabling the plant operators to make a rapid assessment of the severity of an incident and to appropriately classify the emergency. The applicant has committed to include the information on the correlation between the CHRRM readings and the reactor source terms in an emergency plan implementing procedure. Based on our review of the applicant's response and the applicant's commitment to include appropriate information on the CHRRMs in a specific procedure, we find that this item has been satisfactorily resolved. We will confirm the applicant's implementation of this matter in a future supplement to the SER.

We requested the applicant to develop the methodology needed to classify serious events in case the containment monitor or other key radiation monitoring instrumentation are either offscale or inoperable.

In its submittal to the NRC dated February 23, 1984, the applicant states that it has developed the methodology needed to assess a radiological emergency condition in case key monitoring instrumentation (e.g., the CHRRM's) is either offscale or inoperable. This methodology includes using a survey meter and appropriate conversion factors to obtain an estimate, through the biological shield, of the activity in the drywell. Other techniques involve using the post-accident sampling system and sampling the stack effluent to obtain estimates of the activity in the drywell. The applicant has committed to incorporate this methodology in an emergency plan implementing procedure. Based on our review of the applicant's response and the applicant's commitment to include this information in a specific procedure, we conclude that this item has been resolved in an acceptable manner. We will confirm the applicant's implementation of this matter in a future supplement to the SER.

We requested the applicant to revise the emergency plan and implementing procedures to indicate that offsite authorities will be notified within 15 minutes after an emergency has been declared.

In response to our request on this matter, the applicant has revised Section D in Revision 2 to the Fermi-2 emergency plan and also revised procedure EP-290, "Emergency Notification," to state that offsite authorities will be notified within 15 minutes following the declaration of an emergency condition at the Fermi-2 site. We find this response acceptable and, on this basis, we conclude that this item has been resolved.

13.3.2.5 Notification Methods and Procedures

We requested the applicant to develop a notification form for initial messages to offsite organizations and to revise Emergency Plan Implementing Procedure EP-290, "Emergency Notifications," to provide assurance that notifications will be made within the 15-minute period specified in Section IV.D.3 of Appendix E to 10 CFR Part 50.

In response to this particular item, the applicant revised Procedure EP-290 to specify that notifications will be made within the required 15-minute time period following declaration of an emergency. A generic State of Michigan notification form has been developed for notification of offsite response organizations. This notification form is included in implementing procedure EP-290. However, the form requires a large amount of information. While the applicant indicates in Section E.1 of the Fermi-2 emergency plan that the notification form contains provisions for initial and follow-up messages, only the generic State of Michigan notification form is shown in EP-290. (Initial messages contain minimum but essential information while follow-up messages contain more detailed information.) In its submittal dated February 23, 1984, the applicant states that EP-290 will be revised to describe those portions of the notification form to be used for initial messages and those portions to be used for follow-up messages. We find the applicant's response to this item acceptable based on our review of the information contained in the emergency plan and in EP-290, and the applicant's commitment to further revise EP-290 to indicate which portion of the generic notification form is to be used for an initial message and which portion will be used for follow-up messages. We will confirm the applicant's implementation of this matter in a future supplement to the SER.

We requested the applicant to provide a commitment to have a prompt alert and notification system which is in accordance with the guidance of Appendix 3 to NUREG-0654. We further requested that this system be installed and operational prior to fuel load. Alternatively, the applicant could develop interim compensatory measures to provide emergency instructions to the public within the plume exposure pathway EPZ. The applicant was also requested to revise the emergency plan to reflect the deletion of mobile sirens from the alert and notification system.

As described in Section E.2 of Revision 2 to the Fermi-2 emergency plan, the public will be alerted by a system of fixed electronic sirens posted throughout the plume exposure pathway EPZ; i.e., over a 10-mile path. Reference to mobile sirens, which are no longer part of the applicant's alert system, has been deleted from the plan. The applicant states, in its submittal of February 23, 1984, that the siren alert system consisting of a total of 31 sirens in Monroe and Wayne Counties has been installed and each siren has been silently tested and determined to be operational. Installation of the control panel at the Monroe City-County Joint Communications Center remains to be completed to make the system operational. Additionally, the applicant has committed in its submittal of February 23, 1984, to have the siren alert system operational prior to fuel load. Based on the information provided by the applicant and the applicant's commitment cited above, we find this issue has been resolved in an acceptable manner. We will confirm that the siren alert system is installed and operational prior to fuel load in a future supplement to the SER.

We requested the applicant to coordinate its planning efforts with offsite authorities to ensure that there will be administrative capability to alert the public and make prompt protective action decisions for rapidly developing emergency situations, especially during non-normal working hours.

As described in the applicant's emergency plan, the physical means in the form of the siren system will exist for alerting the public within the plume exposure pathway EPZ. Additionally, the applicant is continuing to coordinate its planning efforts with offsite authorities, as indicated in its letter dated February 23, 1984, to provide assurance that the administrative capability will exist for offsite authorities to promptly alert and notify the public, especially in a rapidly escalating situation. The applicant has provided references to the protective action decision-making process in the State and local emergency plans. The Michigan plan specifies that local government has the primary responsibility to activate the alert and notification system and to provide prompt instructions to the public. The revised Monroe County emergency plan (i.e., the draft dated December 1983) specifies that the Chairperson of the Monroe County Board of Commissioners has the authority and responsibility for activating the County emergency plan and implementing protective actions for the public. The County indicates in its plan that the Chairperson, in consultation with the Director of the Monroe City-County Office of Civil Preparedness (OCP), will consider the protective action recommendations made by the plant operator prior to the establishment of communications with the State. In the event of a General Emergency, the County indicates in its plan that the Chairperson and OCP Director will provide decision-making input regarding protective actions. The applicant states that the County has been requested to provide more explicit information in the County plan concerning the responsibility for prompt decision making and public notification during a rapidly moving event.

We find that the applicant has coordinated planning efforts with offsite authorities and while it appears that the capability exists for these authorities to make prompt protective action decision to protect the public, we conclude that this item is confirmatory pending further clarification from the applicant regarding the County's response procedures. We will address this item in a future supplement to the SER.

13.3.2.7 Public Information

We requested the applicant to submit draft public information brochures for review prior to fuel loading and to commit to distribute these brochures to the public before operation at power.

In response to this item, the applicant has provided us with a public information brochure. The information in the brochure includes: (1) educational material on radiation; (2) alerting and notification information; (3) a map of the plume exposure pathway EPZ showing evacuation routes; (4) the location of reception centers; (5) the point of contact for additional information; and (6) basic information on what to do in the event of an evacuation. We have reviewed this public information brochure and found it to be in conformance with the guidance of NUREG-0654. In its submittal dated of February 23, 1984, the applicant states that the information brochure was distributed to the general public prior to the FERMEX '82 exercise conducted in February 1982 and would be re-distributed prior to FERMEX '84 which was conducted in June 1984. Information obtained during the onsite appraisal in October 1983 verified that the brochure has been distributed to the population within the plume EPZ. Based on this information, we find that the applicant has provided an acceptable response to this item.

13.3.2.8 Emergency Facilities and Equipment

The applicant was requested to establish a backup emergency operations facility (EOF) in accordance with regulatory guidance or, alternatively, to justify the absence of this facility.

The applicant has described in Section H.1.3 of Revision 2 to the Fermi-2 emergency plan, an alternate EOF which is located at its Wayne-Monroe Division Headquarters, 22 miles northwest of the Fermi-2 site. Implementing procedures EP-304-1 and EP-304-2 describe the activation of the alternate EOF and the responsibilities of the assigned staff, respectively. In the event that the primary EOF is not available, the alternate EOF provides for the coordination of offsite emergency response actions, including: (1) radiological and environmental assessments; (2) protective action decision making; (3) coordination of public information; and (4) communication with government agencies.

We have reviewed the applicant's request to establish an alternate backup EOF 22 miles from the plant site. We recommended approval of this backup EOF in a letter to the Commission dated December 30, 1983 (SECY-83-524).

On January 20, 1984, we were informed that the commission did not object to the staff's recommended approval of the location of the backup EOF for Fermi-2. On this basis, we consider this item to be now resolved.

We requested the applicant to provide a commitment that the permanent emergency response facilities will be operational before fuel loading or that adequate interim facilities and equipment will be in place.

The applicant's primary response facilities (ERF's) consist of a Technical Support Center (TSC) located within the protected area on the ground floor of the office services building, an Operations Support Center located near the control room in the turbine building, and an Emergency Operations Facility (EOF) located in the Nuclear Operations Center about 6,000 feet southwest of the plant on property controlled by the applicant. In a letter to the NRC staff dated June 23, 1983, the applicant estimated that the ERF's would be operational by October 1983, and functional by September 1984. Operational is defined as available and capable of being staffed to respond to an emergency without the Emergency Response Information System (ERIS), an automated data acquisition system which includes the Safety Parameter Display System (SPDS). Functional is defined here to indicate that the ERIS is installed and operational and that the Fermi-2 operating personnel are trained on its use.

Based on the information in the Fermi-2 emergency plan and the findings of the emergency plan implementation appraisal conducted at the Fermi-2 site in October 1983, we conclude that, on an interim basis, the ERF's are adequate to support a response effort in the event of an emergency. As indicated in Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capability," dated December 17, 1982, we will conduct a post-implementation appraisal of the adequacy of the applicant's completed ERF's on a schedule to be developed between the applicant and the NRC. This item is considered to be resolved. However, we will confirm that the ERIS is functional in a future supplement to the SER.

13.3.2.9 Accident Assessment

We requested the applicant to provide a commitment to have the ERIS system operational before fuel loading, or upgrade the capability of the manual dose assessment model to account for all monitored gaseous release pathways and nonmonitored releases.

The applicant estimated in a letter to the NRC dated June 23, 1983, that the ERIS would be installed and operational in the TSC and EOF by September 1984. The adequacy of the ERIS will be evaluated as part of the post-implementation appraisal of the applicant's completed ERF's to be conducted by the staff in accordance with the requirements of Supplement 1 to NUREG-0737.

We requested the applicant to establish the methodology for performing an analysis of the radiological consequences if the instrumentation used for assessment is off-scale or inoperable.

The applicant's manual dose assessment methodology is presented in its emergency plan implementing procedures EP-540, EP-541, EP-542, and EP-543. These procedures address the calculation of doses resulting from airborne and waterborne releases. A review of the procedures indicates that the manual dose assessment models account for the design basis loss-of-coolant accident and other accidents with monitored releases through the standby gas treatment system (SGTS), and unmonitored releases through the SGTS and other building vent stacks utilizing grab sample information, fan flow rates, and other plant data to estimate a source term. Based on this review, the staff finds that the applicant's manual dose assessment capability has been upgraded and is adequate to evaluate the potential offsite consequence of a radiological emergency. When the ERIS system with its automated dose assessment capability is functional, the manual dose assessment procedures will provide a backup calculational capability. Based on these considerations, we find that the above two items have been resolved in an acceptable manner.

13.3.2.10 Protective Response

We requested the applicant to develop predetermined protective action recommendations in accordance with the guidance of Appendix 1 to NUREG-0654 and to incorporate these recommendations into the emergency plan and procedures.

In response, the applicant has developed predetermined protective action recommendations which are included in its emergency plan implementing procedure EP-545, "Protective Action Guidelines Recommendations." This procedure provides guidance for determining the appropriate protective action recommendations (PAR's) based on either radiation dose estimates or on plant conditions and status. We find that the PAR's are in accordance with the guidance of Appendix 1 to NUREG-0654. In addition to predetermined PAR's, EP-545 contains information which would be useful to a decision-maker in developing recommendations for offsite authorities. This information typically consists of: (1) shielding factors for various structures for both gamma cloud and surface deposited radionuclides; (2) contamination action levels; (3) evacuation time estimates; (4) population distributions; (5) a map of evacuation subareas; and (6) the location and the population of special facilities. A short overview of the protective actions for the public is given in Section J.4 of the Fermi-2 emergency plan. However, this discussion does not indicate that the PAR's may be based on plant status as well as on dose projections as specified in EP-545. In response to our concern on this matter, the applicant has committed in its letter dated February 23, 1984, to revise the plan.

Based on our review of EP-545, and on the applicant's commitment to revise the emergency plan to indicate that PAR's are based on plant conditions as well as dose estimates, the staff finds that this item has been resolved. We will confirm the applicant's implementation of this item in a future supplement to the SER.

We requested the applicant to revise its evacuation analysis to include: (1) a listing of the special facility population on an institution-by-institution basis; (2) an estimate of the reduction in road capacity which could be caused by adverse weather; and (3) coordination of this study with local authorities.

In its letter to the NRC dated January 10, 1983, the applicant submitted a listing of special population groups and information on the effect of adverse weather conditions on vehicular flow rates. The requested information is contained in two addenda to the original evacuation time estimate study developed for the applicant by PRC Voorhees in October 1980 and revised in March 1982. In its letter of January 10, 1983, the applicant states that the evacuation time estimates contained in the Monroe and Wayne County emergency plans were also developed by PRC Voorhees and are consistent with the evacuation time estimates in the Fermi-2 emergency plan. Implementing procedures EP-545, "Protective Action Guideline Recommendations," includes maps showing the locations of the special facilities within the plume exposure pathway EPZ and also includes a table containing the evacuation time estimates for various population groups for both normal and adverse weather conditions. Based on our review of the information in the applicant's letter of January 10, 1983, and in EP-545, we find that the applicant has provided an acceptable response to this item. However, we recommend that the special facility population information also be included in implementing procedure EP-545. We will confirm the applicant's implementation of this item in a future supplement to the SER.

13.3.2.15 Radiological Emergency Response Training

We requested the applicant to clearly define the Fermi-2 emergency plan training program categories to provide assurance that all personnel who will implement the plan and all functional areas of emergency activity are included. In addition, we requested the applicant in a letter dated July 22, 1983, to coordinate its planning efforts with the appropriate staff and local officials to provide assurance that training is provided to local emergency response personnel.

In Revision 2 of Table O-1 of the Fermi-2 emergency plan, the applicant presented a matrix of emergency plan training courses and emergency organization positions. Twenty training courses and 64 emergency functional positions are listed with the courses given to the individuals who fill each position shown in the matrix. Our review of this information indicated that appropriate training will be provided to members of the onsite emergency response organization. The applicant has also provided information in its submittal dated February 23, 1984, which establishes that the applicant has been actively involved in the development of a training program for local offsite emergency workers. The program has been developed in conjunction with the State of Michigan's Emergency Management Division and the counties within the plume exposure EPZ. General training is provided in such subject areas as basic nuclear physics, plant operations, biological effects of radiation, and emergency plans and procedures at the State and County level. Specialized training is given to certain groups of emergency workers in specific areas such as radiological monitoring and decontamination procedures. All offsite emergency workers will receive training in their response duties. The training program will be conducted on an annual basis and will include participation in drills and exercises. Training of offsite response personnel was initiated on March 15, 1984.

The applicant will also hold on an annual basis, a seminar for the offsite emergency response decision makers covering such matters as emergency action levels, dose assessment, meteorology, and protective actions. Information describing the training program for offsite emergency personnel is to be summarized in Revision 3 to the emergency plan. On this basis, we conclude that this item has been resolved. We will confirm the applicant's implementation of this item in a future supplement to the SER.

13.3.4 Federal Emergency Management Agency (FEMA) Findings on Offsite Emergency Plans and Preparedness

FEMA has been actively involved in the development and review of offsite emergency plans for the Fermi-2 facility. In its letter dated February 28, 1983, FEMA transmitted to the NRC an interim finding report on the State and local emergency plans and preparedness for the Fermi-2 facility. This report supplemented previous FEMA finding and status reports dated January 26, March 22, and April 30, 1982. Based on its review of the emergency plans for the State of Michigan and for Monroe and Wayne Counties, and on observations made during the full-scale exercise held in February 1982, FEMA reported that an adequate level of offsite emergency preparedness existed for the Fermi-2 facility. However, FEMA identified several offsite preparedness areas requiring either additional information or corrective action. In its letter dated July 18, 1983, FEMA provided additional information and clarification of their interim finding report of February 28, 1983. The FEMA areas of concern are discussed in the following sections.

13.3.4.1 State Emergency Classification System

FEMA noted that while the emergency classification system reflected in State and local emergency plans was consistent with NUREG-0654/FEMA-REP01, Revision 1, a 1977 State law embodied a different classification system terminology (i.e., Class A, Class B, and Class C). The emergency classification system in current State and local plans consists of the four standard emergency classes; i.e., Notification of Unusual Event, Alert, Site Area Emergency and General Emergency. This categorization of emergency classes is in agreement with the classification system used in the Fermi-2 onsite emergency plan and procedures. Our position is that the emergency classification system for Fermi-2 is acceptable provided the emergency classification system specified in State, local and applicant plans remains consistent and in conformance with the guidance contained in Revision 1 to NUREG-0654. We conclude that this issue is not an impediment to effective emergency planning and response for Fermi-2.

13.3.4.2 Radiological Emergency Response Training

The FEMA interim finding report identified the need for an integrated, comprehensive training program for offsite emergency response personnel. In its letter of July 18, 1983, FEMA informed the NRC that the Michigan Emergency Management Division has taken the lead in developing a training program for offsite emergency workers. Additional information on the program is contained in the letter from the applicant dated February 27, 1984, and discussed in Section 13.3.2.15 of this supplement. A key aspect of the program is the joint participation of the State, local agencies, and the applicant.

Based on our review of the information provided by FEMA and the applicant on the training program for offsite emergency personnel, we conclude that this issue has been resolved.

13.3.4.3 Notification Methods and Procedures

FEMA expressed a concern regarding the availability of the siren alerting system and the timeliness of notification of the public by offsite authorities. As discussed in Section 13.3.2.5 of this supplement, the applicant's siren alerting system will be installed and operational prior to fuel load. As further indicated in Section 13.3.2.5, State and local emergency plans contain provisions for prompt alerting and notification of the public by local officials, based on the recommendations of the utility, in the event of a rapidly escalating emergency. The applicant has committed to provide additional information on this subject pending clarification of the protective action decision making process in the revised Monroe County emergency plan. We consider that FEMA's concerns in this area have been satisfactorily addressed.

13.3.4.4 Additional Offsite Emergency Plan Review

We have requested FEMA's support in reviewing the revised radiological emergency plan for Monroe County and the plan developed for Brownstown Township in Wayne County, Michigan. Both counties are within the plume exposure pathway EPZ. As stated in a draft of the Monroe County plan dated December 1983, the plan has been substantially revised and expanded to reflect the specific needs of Monroe County and to define the use of the County's resources. Brownstown Township being a local government subdivision of over 10,000 population, has elected under Michigan law to develop on its own, an emergency plan which will be separate from that of Wayne County. FEMA's findings and determinations on the revised Monroe County plan and the Brownstown Township plan, will be provided in a future supplement to our SER.

13.3.5 Interim Conclusions

Based on the review of Revision 2 of the Fermi-2 Radiological Emergency Response Plan, and additional information provided by the applicant in a letter dated February 23, 1984, the staff finds that the items previously identified in Supplement No. 2 to the Fermi-2 SER as requiring resolution, have either been satisfactorily addressed or the applicant has committed to provide the required information. We conclude on an interim basis, that upon satisfactory completion of those items for which the applicant has made commitments, an adequate planning basis for an acceptable state of onsite emergency preparedness will exist in accordance with the requirements in Appendix E and Section 50.47 of 10 CFR Part 50.

After a review of the findings and determinations made by the Federal Emergency Management Agency on the adequacy of State and local emergency response plans, and upon confirmation that the applicant's commitments have been satisfied, a future supplement to the SER will provide the staff's overall final conclusions regarding whether the state of onsite and offsite emergency preparedness for the Fermi-2 facility provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency at the Fermi-2 facility.

14 INITIAL TEST PROGRAM

In Supplement No. 3 to the SER, we stated that the initial test program was acceptable, subject to modification of the Fermi 2 FSAR to reflect the deletion of the steam condensing mode of the residual heat removal (RHR) system. In Amendment No. 51 to its FSAR, the applicant proposed deleting this operating mode of the RHR system and stated that it had removed the associated piping and valves.

While there are certain operating modes of the RHR which are safety-related (e.g., low pressure coolant injection, containment spray and long-term shutdown cooling), the steam condensing mode is not safety-related. Additionally, the applicant does not use this RHR operating mode for mitigating either transients or accidents. On this basis, we conclude that the applicant's proposal to delete the steam condensing mode of the RHR system is acceptable. We find that this matter is now resolved.

22 TMI-2 REQUIREMENTS

22.2 TMI Action Plan Requirements for Applicants for Operating Licenses

I. Operational Safety

I.C Operating Procedures

I.G.1 Training During Low-Power Testing

In our SER, we stated that we would review the applicant's safety analysis and test procedures for a Simulated Loss of Onsite and Offsite Alternating Current Power Test which we designated as a station blackout (SBO) test. This test was to be performed subject to a safety analysis finding that the SBO test would not constitute either a risk to public safety or a risk of damage to equipment in the Fermi-2 facility. We advised all applicants for an operating license (OL) for a boiling water reactor (BWR) that a safety analysis for an SBO test at the Susquehanna Steam Electric Station had shown that the test would pose an unacceptable risk of damage to equipment in the drywell of that facility. Accordingly, we reassessed the value and risk of the SBO test and concluded that the test is not warranted at this time. (We will reconsider the need for an SBO test when Unresolved Safety Issue A-44, "Station Blackout," has been resolved.) We further advised OL applicants that compliance with the BWR Owners' Group recommendations regarding TMI Item I.G.1 constitutes an acceptable program for satisfying our requirements for this TMI item.

We requested Detroit Edison to determine if the Susquehanna findings apply to the Fermi-2 facility. The applicant confirmed in its letter dated October 5, 1983, that the Susquehanna findings are applicable to the Fermi-2 facility; i.e., the SBO test would cause high temperature and humidity in the drywell, possibly damaging non-safety grade equipment. In addition, the applicant reaffirmed its commitment to the BWR Owners' Group recommendations.

Based on the applicant's commitment to the BWR Owners' Group program on this matter and consistent with our position on this matter, we conclude that the applicant's commitment in its letter dated October 5, 1983, satisfies our requirements regarding TMI Item I.G.1 without the need to perform an SBO test. Accordingly, we find that this item is now resolved.

APPENDIX A

CONTINUATION OF CHRONOLOGY OF RADIOLOGICAL SAFETY REVIEW

November 9, 1982	Letter from applicant concerning submittal of reports for the Fermi 2 vacuum breakers.
November 9, 1982	Letter from applicant concerning additional information to verify the applicability of generic SRV test results to Fermi 2.
November 16, 1982	Letter from applicant concerning reactor pressure vessel water level sensing lines.
November 18, 1982	Representatives from NRC & DE meet in Bethesda, Md. to discuss plans for completing environmental qualification. (Summary issued December 13, 1982)
November 30, 1982	Letter from applicant transmitting Amendment No. 45 to the FSAR.
December 1, 1982	Representatives from NRC & DE meet in Bethesda, Md. to discuss and clarify an independent design verification program for Fermi 2. (Summary issued December 15, 1982)
December 1, 1982	Letter to applicant transmitting Amendment No. 2 to CPPR-87. This amendment merges two owners and changes their name. Detroit Edison maintains sole responsibility for operating plant as lead applicant.
December 3, 1982	Letter from applicant concerning channel box deflection.
December 3, 1982	Letter from applicant concerning reactor building base mat capacity reserved for Torus uplift.
December 6, 1982	Letter from applicant concerning information on emergency operations facility.
December 7, 1982	Letter from applicant concerning scope of an independent design verification program for Fermi 2.
December 9, 1982	Letter from applicant transmitting a certificate of service for Amendment 45 to the FSAR.
December 15, 1982	Letter from applicant concerning request for drawings to support review of Fermi 2 Inservice Inspection Program for pumps and valves.

December 27, 1982	Letter to applicant concerning acceptance of the proposed Fermi-2 Design Verification Program.
January 7, 1983	Letter from applicant concerning a request for exemption of Section IV.F.1 of Appendix E to 10 CFR Part 50 to conduct a full-scale emergency preparedness exercise within one year before issuance of a full power operating license for Fermi-2.
January 11, 1983	Letter from applicant concerning radiological emergency response plan Table D-1, Emergency Action Levels.
January 24, 1983	Letter to applicant transmitting two copies of the SER Supplement 3 to Fermi-2. The printed copies will be forwarded when they have returned from our printer-contractor.
January 26, 1983	Representatives from NRC and Detroit Edison meet in Bethesda, MD to discuss proposed changes in operator staffing and in the Nuclear Review and Audit Group. (Summary issued)
January 31, 1983	Letter from applicant transmitting Amendment No. 46 to the FSAR.
February 4, 1983	Letter to applicant transmitting 20 copies of Supplement No. 3 to NUREG-0798 (SER Supplement No. 3) for Fermi-2.
February 8, 1983	Letter from applicant transmitting a Certificate of Service for Amendment No. 46.
February 14, 1983	Letter from applicant concerning simulated loss of AC power special test.
March 16, 1983	Letter to applicant concerning unqualified electrical components in safety-related systems.
March 16, 1983	Letter from applicant concerning pump and valve inservice testing program.
March 16, 1983	Letter from applicant concerning Fermi 2 vacuum breakers.
March 31, 1983	Letter from applicant transmitting Amendment No. 47 to the amended and substituted application for licenses.
April 7, 1983	Letter from applicant concerning submittal of SQRT list update and confirmation of open items.
April 7, 1983	Letter from applicant transmitting a Certificate of Service for Amendment No. 47 to the FSAR.
April 11, 1983	Letter from applicant concerning induction heating stress improvement (IHSI) Program on Fermi 2.

May 3, 1983	Letter from applicant concerning final report - Independent Design Verification Program.
May 3, 1983	Letter from applicant transmitting a summary of post accident sampling analytical procedures.
May 13, 1983	Representatives from NRC & DECO meet in Bethesda, Md. to discuss the general procedures followed by the NRC staff in responding to a Section 2.206 petition. (Summary issued)
May 17 & 18, 1983	Representatives from DE & EG&G Idaho Falls meet at the site to discuss the inservice test program. (Summary issued August 18, 1983)
May 18, 1983	Letter from applicant concerning induction heating stress improvement (IHSI) program on Fermi 2.
May 26, 1983	Letter from applicant concerning further information on FERMI 2 Low-Low Set Design and Analysis.
May 27, 1983	Letter from applicant transmitting Amendment No. 48 to the FSAR.
June 7, 1983	Letter from applicant transmitting a Certificate of Service for Amendment No. 48 to the FSAR.
June 7, 8 & 9, 1983	Representatives from NRC & DECO meet at the Fermi-2 Site in Troy, Michigan for the NRC Caseload Forecast Panel to visit the Fermi-2 facility to estimate the construction completion date. (Summary issued August 2, 1983)
June 10, 1983	Letter from applicant submitting the plant unique analysis report for torus attached piping.
June 14, 1983	Letter to applicant concerning methodology for establishing setpoints for the Fermi-2 Technical Specifications.
June 14, 1983	Representatives from NRC & DECO meet in Bethesda, MD. to discuss three specific topics related to fire protection. (Summary issued)
June 15, 1983	Representatives from NRC, Cygna, & DECO meet in Bethesda, Md. for Cygna who performed the IDVP to present its findings including the basis for its conclusions. (Summary issued)
June 22, 1983	Letter from applicant advising that the new fuel load date is December 30, 1983.
June 22, 1983	Letter to applicant concerning staff position on the design criteria for Fermi-2 standby liquid control system.

June 29, 1983	Letter from applicant concerning Emergency Operations Facility.
June 29, 1983	Letter from applicant transmitting a revision to the physical security plan.
June 30, 1983	Letter from applicant transmitting Amendment No. 49 to the FSAR.
July 7, 1983	Representatives from NRC & DE meet in Bethesda, Md. to discuss the staff's positions regarding the design criteria for the instrumentation control and power cables of the standby liquid control system of Fermi-2. (Summary issued July 19, 1983).
July 7, 1983	Letter from applicant transmitting a Certificate of Service for Amendment No. 49 to the FSAR.
July 8, 1983	Letter from applicant concerning evaluation of safety relief valve piping and torus attached piping for the effects of site-specific earthquake.
July 15, 1983	Letter from applicant concerning methodology for establishing setpoints for Fermi 2 technical specifications.
July 21, 1983	Representatives from NRC & DECO meet in Bethesda, Md. to discuss a proposal regarding the Independent Safety Evaluation Group. (Summary issued)
July 22, 1983	Letter to applicant concerning emergency preparedness plans.
July 28, 1983	Letter to applicant concerning site verification visit for the environmental qualification of Electrical equipment.
July 28, 1983	Letter to applicant concerning evaluation of emergency response facilities for the Fermi-2 facility.
July 29, 1983	Letter from CYGNA transmitting a supplement to final report Independent Design Verification Program.
August 3, 1983	Letter to applicant requesting dates for submittal of previous requests of additional information.
August 17, 1983	Letter from applicant concerning standby liquid control system.
August 18, 1983	Letter from applicant concerning containment leakage testing, Type A.
August 31, 1983	Letter from applicant transmitting Amendment No. 50 to the FSAR.

September 8, 1983	Letter from applicant transmitting a Certificate of Service for Amendment No. 50 to the FSAR.
September 12, 1983	Letter to applicant concerning postulated reactor vessel level sensing line failures at Fermi-2.
September 13, 1983	Letter to applicant concerning a request for exemption from a full-scale emergency preparedness exercise.
September 23, 1983	Letter from applicant concerning Mark I Containment - Torus attached piping submittal of additional information.
September 23, 1983	Letter from applicant concerning postulated reactor vessel sensing line failures at Fermi-2.
October 4, 1983	Letter from applicant transmitting Revision 2 of Report on Inservice Testing of Pumps and Valves.
October 5, 1983	Letter from applicant concerning Commitment to BWR Owners' Group Position in Accordance with Generic Letter 83-24.
October 5, 1983	Letter from applicant concerning deferring the issuance of the Proof and Review version of the Fermi-2 Technical Specifications.
October 10, 1983	Letter from applicant changing the Fermi-2 Supervisor to Mr. O. Keener Earle.
October 14, 1983	Letter from applicant transmitting Amendment 51 to the Amended and Substituted Application for Licenses.
October 19, 1983	Representatives from NRC & DECO meet in Bethesda, Md. to discuss the environmental qualification of equipment. (Summary issued)
October 26, 1983	Letter from applicant transmitting a Certificate of Service for Amendment No. 51 to the FSAR.
November 1, 1983	Letter to applicant concerning control of heavy loads at FERMI-2 in accordance with NUREG-0612.
November 14, 1983	Letter from CYGNA transmitting Supplement to Final Report Independent Design Verification Program.
December 7, 1983	Letter to applicant concerning qualifications of the FERMI-2 Radiation Protection Manager.
December 12, 1983	Letter to applicant concerning Use of BN-TOP-1 for the FERMI-2 Facility Containment Leakage Testing Per Appendix J.
December 16, 1983	Letter from applicant concerning Nuclear Experience Survey.

December 16, 1983	Letter from applicant concerning Emergency Preparedness Exercise.
December 19, 1983	Letter from applicant concerning Fermi-2 Operator Requalification Training Program.
December 20, 1983	Letter from applicant transmitting an applicant to amend Construction Permit CPPR-87 to delete the ownership shares in paragraph 2F of Fermi-2 CP.
December 29, 1983	Letter from applicant transmitting a Certificate of Service for Amendment No. 52 to the application for licenses.
January 17, 1984	Letter to applicant concerning technical specifications for the Fermi-2 1SEG.
January 31, 1984	Letter to applicant concerning Emergency Operations Facilities at Fermi 2.
January 31, 1984	Letter from applicant transmitting two copies of Endorsement 104 of Policy Number NF-92 (Indemnity Insurance Endorsement).
February 6, 1984	Letter from applicant concerning resubmittal of the Draft Fermi 2 Offsite Dose Calculation Manual (ODCM).
February 7, 1984	Letter to applicant concerning deletion of home telephone numbers, unlisted utility numbers, etc. from emergency plans for the Fermi-2 Facility.
February 14, 1984	Letter from applicant concerning additional environmental qualification information requested by the NRC.
February 16, 1984	Letter from applicant concerning staffing levels of Radchem Section.
February 21, 1984	Letter to applicant transmitting Amendment No. 3 to CPPR-87 to delete the ownership share interests in the Construction Permit.
February 22, 1984	Letter from applicant transmitting Amendment 53 to the Amended and Substituted Application for licenses consisting of certain additions and modifications to the FSAR.
February 22, 1984	Letter to applicant requesting additional information for the Fermi-2 facility regarding a postulated failure of a HPCI steam line without isolation.
February 21, 1984	Letter from applicant concerning incorporate previous responses to the Mark I Containment Plant Unique Analysis Report.

February 23, 1984	Letter from applicant concerning response to items requiring resolution evaluation of Emergency Plan Supplement No. 3, NUREG-0798, January, 1983.
March 5, 1984	Letter from applicant concerning use of BN-TOP-1 for the Fermi-2 facility containment leakage testing per Appendix J.
March 9	Letter from applicant transmitting a Certificate of Service for Amendment 53 to the Amended and Substituted application for licenses.
March 14, 1984	Letter to applicant concerning qualifications of the Fermi-2 radiation protection manager.
March 14, 1984	Letter from applicant concerning issue of proof and review Technical Specifications.
March 14, 1984	Letter from applicant concerning clarification of TMI Action Plan Item II.K.3.31.
March 21, 1984	Letter from applicant concerning conformance to NUREG-0619.
March 26, 1984	Letter from applicant transmitting a Certificate of Service for Amendment 54 to the Amended and Substituted application for licenses.
March 27, 1984	Letter from applicant concerning leakage reduction program.
March 27, 1984	Letter to applicant concerning preliminary evaluation of the IDVP performed by Cygna Energy Services for Fermi-2.
March 30, 1984	Letter from applicant transmitting 2 copies of Endorsement 104 of Indemnity Policy No. NF-92 for Fermi-2.
March 30, 1984	Letter from applicant transmitting Amendment No. 55 to the Amended and Substituted application for licenses and additions and modifications to the FSAR concerning response to Commission staff questions.
April 3, 1984	Letter from applicant concerning control of heavy loads and responses to Generic Letter 83-42.
April 20, 1984	Letter from CYGNA Energy Services concerning independent design verification program - notice of scheduled meeting with NRC.
May 8, 1984	Letter from applicant concerning results of the short-term meteorological study conducted to determine the effect of Lake Erie on plume transport characteristics at the Fermi-2 site.

May 8, 1984	Letter to applicant concerning review of the Fermi-2 ODCM.
May 11, 1984	Representatives from NRC, DECO and CYGNA meet in Bethesda, M.d for CYGNA to present its program for responding to the NRC Staff's concerning regarding the Fermi-2 IDVP. (Summary Issued)
May 12, 1984	Letter to applicant concerning independent safety engineering activities.
May 24, 1984	Letter to applicant requesting additional information on JIO's for the environmental qualification of equipment important to safety in the Fermi-2 facility.
May 25, 1984	Letter from applicant transmitting an application for Amendment 4 to the Fermi-2 Construction Permit.
May 25, 1984	Letter from applicant concerning HPCI Steam Line Isolation Valve Integrity.
June 5, 1984	Letter to applicant concerning FEMA Supplemental Interim Finding on Onsite Radiological Emergency Planning for Fermi-2.
June 5, 1984	Representatives from NRC, Detroit Edison & Region III met in Bethesda, Md. to discuss the compliance of the as-built Fermi-2 facility with the requirements of Appendix R to 10 CFR Part 50. (Summary issued July 10, 1984)
June 5, 1984	Letter from applicant transmitting certificate of service for Amendments 55 and 56.
June 8, 1984	Letter from applicant transmitting Amendment 57 to the amended and substituted application for licenses and modifications to the FSAR.
June 13, 1984	Letter from applicant reporting failures of safety/relief valves.
June 15, 1984	Letter to applicant concerning protocol Governing the Independent Design Verification Program (IDVP) being conducted by CYGNA for the Fermi-2 Facility.
June 20, 1984	Letter from applicant transmitting certificate of service for Amendment 57 to the amended and substituted application for license.
June 22, 1984	Letter from applicant concerning standby liquid control system.

June 22, 1984	Letter from applicant submitting a SQRT list update and responses to open items.
June 25, 1984	Letter from applicant concerning license condition for control blade stress corrosion cracking.
June 26, 1984	Letter from applicant concerning secondary containment drawdown time.
July 3, 1984	Letter from applicant concerning Fermi-2 Contingency Plan.
July 10, 1984	Representatives from NRC & Detroit Edison meet in Bethesda, Md. concerning the requirements of Appendix R to 10 CFR Part 50. (Summary issued August 6, 1984)
July 11, 1984	Letter from applicant concerning code compliance verification of ASME Class I Flued Heads.
July 13, 1984	Letter from applicant concerning NRC Question to CYGNA on RHR Reservoir Freezing.
July 13, 1984	Letter from applicant concerning response to Generic Letter No. 84-11.
July 20, 1984	Letter from applicant concerning comments on NUREG-0798.
July 24, 1984	Letter from applicant concerning amended physical security plan.
July 25, 1984	Letter from applicant concerning control room design review open item status.
July 31, 1984	Letter from applicant transmitting Amendment 58 to the FSAR.
August 8, 1984	Letter from applicant transmitting a certificate of service for Amendment 58 to the FSAR.
August 16, 1984	Letter from applicant concerning procedure for estimating core damage.
August 22, 1984	Letter from applicant concerning Radiation Protection Manager (RPM) Qualifications.
August 24, 1984	Letter from applicant concerning radiological emergency response plan, revision 3A.

APPENDIX G

NRC STAFF CONTRIBUTORS AND CONSULTANTS

This supplement to the SER is a product of the NRC staff and its consultants. The NRC staff members listed below were principal contributors to this report. A list of consultants follows the list of staff members.

NRC STAFF

Name	Branch
R. Eberly	Chemical Engineering
R. Kirkwood	Mechanical Engineering
F. Kantor	Emergency Preparedness Licensing
A. Lee	Equipment Qualification
W. Lefave	Auxiliary Systems
W. Long	Procedures and Systems Review
G. Thomas	Reactor Systems
M. Virgilio	Instrumentation and Control Systems
S. L. Wu	Core Performance

CONSULTANTS

Name	Organization
T. L. Cook	EG&G Idaho, Inc
W. H. Hubble	EG&G Idaho, Inc

APPENDIX K

SAFETY EVALUATION REPORT

PUMP AND VALVE INSERVICE TESTING PROGRAM
ENRICO FERMI ATOMIC POWER PLANT, UNIT 2

The review contained in this Appendix was prepared with substantial assistance from the Idaho National Engineering Laboratory (EG&G Idaho, Inc.) under contract to the U.S. Nuclear Regulatory Commission.

CONTENTS

CONTENTS (Continued)

CONTENTS (Continued)

1. INTRODUCTION

Contained herein is a safety evaluation of the pump and valve inservice testing (IST) program submitted by the Detroit Edison Company for the Enrico Fermi Atomic Power Plant, Unit 2.

The working session with Detroit Edison and Enrico Fermi, Unit 2, representatives was conducted on May 16 and 17, 1983. The licensee's preliminary resubmittal dated September 19, 1983, was received by EG&G Idaho, Inc., on September 21, 1983, and was reviewed to verify compliance of proposed tests of Class 1, 2, and 3 pumps and valves whose function is important to safety with the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, 1980 Edition through the Winter of 1980 Addenda. In their resubmittal, Revision 2 of DET-16-0201, Detroit Edison has requested relief from the ASME Code testing requirements for specific pumps and valves and these requests have been evaluated individually to determine whether they have significant risk implications and whether the tests, as required, are indeed impractical.

The evaluations in this SER of the Enrico Fermi Atomic Power Plant, Unit 2, pump and valve inservice testing program and the associated relief requests are the recommendations of EG&G Idaho, Inc. The Mechanical Engineering Branch, Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission has concurred with these recommendations.

A summary of pump and valve Section XI testing requirements is provided in Appendix A.

Category A, B, and C valves that meet the requirements of the ASME Code, Section XI, and are not exercised quarterly are addressed in Attachment 1.

A listing of P&IDs used for this review is contained in Attachment 2.

2. PUMP TESTING PROGRAM

The Enrico Fermi, Unit 2, IST program submitted by the Detroit Edison Company was examined to verify that all Class 1, 2, and 3 pumps whose function is important to safety are included in the program and are subjected to the periodic tests required by the ASME Code, Section XI. Our review found that these pumps are tested in accordance with Section XI except for those pumps identified below for which specific relief from testing has been requested and as noted in Attachment 3. Each Detroit Edison Company basis for requesting relief from the pump testing requirements and the EG&G Idaho, Inc., evaluation of that request is summarized below.

2.1 All Pumps in the IST Program

2.1.1 Relief Request

The licensee has requested specific relief from the test requirement of measuring vibration amplitude for all pumps in the IST program in accordance with the requirements of Section XI and proposed measuring vibration velocity for these pumps.

2.1.1.1 Code Requirement

Refer to Appendix A.

2.1.1.2 Licensee's Basis for Requesting Relief

Enrico Fermi Atomic Power Plant proposes an alternate program which is believed to be more comprehensive than that required by Section XI. The proposed program is based on vibration readings measured in velocity units rather than vibration amplitude in mils displacement. This technique is more sensitive to small changes that are indicative of developing mechanical problems and hence more meaningful. Velocity measurements detect not only high amplitude vibrations that indicate a major mechanical problem, but also the equally harmful low amplitude, high frequency vibrations resulting from misalignment, imbalance, or bearing wear that usually go undetected by simple displacement measurements.

In conclusion, the foregoing reasons demonstrate that the proposed program of vibration measurements is a more practical method of testing which meets the intent of the ASME Code requirements.

Pump vibration measurements will be taken in vibration velocity (in/sec). The evaluation of the readings will be as follows:

Acceptable Range: less than .236 inches/sec

Alert Range: greater than or equal to .236 inches/sec
: less than .314 inches/sec

Required Action Range: greater than or equal to .314 inches/sec.

2.1.1.3 Evaluation

We agree with the licensee's basis and, therefore, feel that relief should be granted from the requirements of Section XI for measuring displacement vibration amplitude for all pumps in the IST program. The licensee has demonstrated that vibration velocity measurements are superior to displacement vibration amplitude measurements for monitoring pump degradation. Also, the "alert range" and "required action range" that the licensee has proposed utilizing for the evaluation of the readings meet the current NRC staff position for these ranges.

2.1.1.4 Conclusion

We conclude that the licensee's proposed alternate testing method of measuring vibration velocity for all pumps in the IST program should provide sufficient information to adequately monitor pump degradation and meet the intent of the Section XI requirements. Based on the considerations discussed above, we conclude that the alternate testing proposed will give reasonable assurance of pump operability intended by the Code and that the relief thus granted will not endanger life or property or the common defense and security of the public.

2.1.2 Relief Request

The licensee has requested specific relief from the test requirement of measuring pump bearing temperatures for all pumps in the IST program in accordance with the requirements of Section XI and proposes to utilize vibration velocity readings to detect bearing problems.

2.1.2.1 Code Requirement

Refer to Appendix A.

2.1.2.2 Licensee's Basis for Requesting Relief

Enrico Fermi Atomic Power Plant proposes an alternate program which is believed to be more comprehensive than that required by Section XI. The proposed program is based on vibration readings measured in velocity units. This technique is sensitive to small changes that are indicative of developing mechanical problems and hence more meaningful. In addition, these readings go far beyond the capabilities of a bearing temperature monitoring program. A bearing will be seriously degraded prior to the detection of increased heat at the bearing housing. Quarterly vibration velocity readings will achieve a much higher probability of detecting developing problems than annual bearing temperature readings.

Finally, IWP-3500 requires "three successive readings taken at ten minute intervals that do not vary more than three percent." Meeting this requirement for pumps having no recirculation test loop would be very difficult because the system water temperature, and consequently the lubricant temperature, are expected to drift more than three percent during 20 minutes. Also, the temperature of the lubricating fluid will vary with ambient conditions and make meaningful data trending impractical.

As described above, a program of bearing temperature trends and the evaluation of the results would in some cases be difficult to analyze. Improper interpretation of results could result in unnecessary pump maintenance. In addition, it is impractical to measure bearing temperatures on many of the pumps in the program. Some specific examples are as follows:

(1) Core Spray

The pump bearings are lubricated by emergency equipment cooling water flow. Changes in emergency equipment cooling water system temperature would seriously affect the accuracy of trends.

(2) Residual Heat Removal (RHR)

Same as (1) above.

(3) High Pressure Coolant Injection

This pump is driven by a steam turbine which exhausts steam into the pressure suppression chamber. Extended run times to stabilize bearing temperatures could heat the suppression pool water to a temperature exceeding the Technical Specification limit of 105°F (Technical Specification paragraph 3.6.2.1.a.2.a).

In conclusion, the foregoing reasons demonstrate that the proposed program of vibration measurements is a more practical method of testing which meets the intent of the ASME Code requirements.

Pump vibration measurements will be taken in vibration velocity units (in/sec). The evaluation of the readings will be per the ranges given in Relief Request PR-1. Temperature measurements will not be taken.

2.1.2.3 Evaluation

We agree with the licensee's basis and, therefore, feel that relief should be granted from the requirements of Section XI for measuring bearing temperatures for all pumps in the IST program. The licensee has demonstrated that annual bearing temperature measurements may not detect pending bearing failure as soon as the proposed alternate testing method of measuring vibration velocity and, therefore, we feel that deletion of this measurement will not affect the licensee's pump monitoring program.

2.1.2.4 Conclusion

We conclude that the licensee's proposal to monitor pump vibration velocity rather than bearing temperature to detect bearing degradation should be sufficient to monitor pump degradation. Based on the considerations discussed above, we conclude that the alternate testing proposed will give reasonable assurance of pump operability intended by the Code and that the relief thus granted will not endanger life or property or the common defense and security of the public.

2.2 Service Water Pumps

2.2.1 Relief Request

The licensee has requested specific relief from the test requirement of measuring inlet pressure for the residual heat removal service water pumps, emergency equipment service water pumps, and diesel generator service water pumps in accordance with the requirements of Section XI and proposed calculating inlet pressure for these pumps from RHR reservoir level.

2.2.1.1 Code Requirement

Refer to Appendix A.

2.2.1.2 Licensee's Basis for Requesting Relief

The pump impellers of the subject pumps are submerged in the RHR reservoir. The inlet pressure at the impeller is simply the hydrostatic head. Because there is no instrumentation at the pump inlet, the hydrostatic head will be computed from the reservoir level.

The inlet pressure measurement, by computation of hydrostatic head developed from the reservoir level, is a practical method of testing, given the placement of the impellers in the RHR reservoir.

The inlet pressure measurement will be based on reservoir level.

2.2.1.3 Evaluation

We agree with the licensee's basis and, therefore, feel that relief should be granted from measuring inlet pressure for these pumps in accordance with the requirements of Section XI. These pumps are submerged in the RHR reservoir and do not have installed inlet pressure measurement devices. As an alternate testing method, the licensee has proposed calculating inlet pressure from the level of the reservoir. We feel this proposed alternate testing method meets the intent of the Code.

2.2.1.4 Conclusion

We conclude that the licensee's proposed alternate testing method of calculating inlet pressure for these pumps from the RHR reservoir level should provide sufficient information to adequately monitor pump degradation and meet the intent of the Section XI requirements. Based on the considerations discussed above, we conclude that the alternate testing proposed will give reasonable assurance of pump operability intended by the Code and that the relief thus granted will not endanger life or property or the common defense and security of the public.

2.3 Standby Liquid Control Pumps

2.3.1 Relief Request

The licensee has requested specific relief from the test requirements of measuring inlet and differential pressure for the standby liquid control pumps in accordance with the requirements of Section XI and proposed measuring discharge pressure, flow rate, and vibration during the inservice testing of these pumps.

2.3.1.1 Code Requirement

Refer to Appendix A.

2.3.1.2 Licensee's Basis for Requesting Relief

No suction tap or inlet pressure instrumentation is provided for the Standby Liquid Control pumps. Suction pressure when testing is small compared to discharge pressure (less than 3 psig compared to 1190 psig). The pumps are positive displacement pumps and since the suction pressure is low, the differential pressure is essentially equal to discharge pressure (1187 psig vs. 1190 psig). The suction pressure is less than one percent of discharge pressure and can be considered insignificant.

Discharge pressure, flow rate and vibration will be measured during inservice testing. Check adequate suction head to ensure safe pump operation by determining liquid level in the storage tank.

2.3.1.3 Evaluation

We agree with the licensee's basis and, therefore, feel that relief should be granted from measuring inlet and differential pressure for these pumps in accordance with the requirements of Section XI. These pumps are positive displacement pumps, therefore, changes in the inlet pressure have no effect on the discharge pressure or the flow rate of the pumps. For this reason, we feel that calculating or measuring inlet pressure would not contribute meaningful data to utilize in monitoring pump degradation.

2.3.1.4 Conclusion

We conclude that the proposed alternate testing of measuring pump discharge pressure, flow rate, and vibration should provide sufficient information to adequately monitor pump degradation. Based on the considerations discussed above, we conclude that the alternate testing proposed will give reasonable assurance of pump operability intended by the Code and that the relief thus granted will not endanger life or property or the common defense and security of the public.

2.4 Diesel Fuel Oil Transfer Pumps

2.4.1 Relief Request

The licensee has requested specific relief from measuring inlet pressure, differential pressure, and flow rate for the diesel fuel oil transfer pumps in accordance with the requirements of Section XI.

2.4.1.1 Code Requirement

Refer to Appendix A.

2.4.1.2 Licensee's Basis for Requesting Relief

No flow control or instrumentation is provided for the diesel fuel oil transfer pumps. In addition, flow rate cannot be indirectly measured based on diesel fuel oil day tank level changes with time since the diesel fuel oil day tank cannot be drained back into the main storage tank. Therefore, flow rate cannot be measured quarterly during inservice testing. Because flow cannot be controlled and cannot be measured, repeatability and hydraulic ranges cannot be established for differential pressure. No instrumentation exists for measuring inlet pressure.

The diesel fuel oil day tank level will be monitored during the emergency diesel generator test. If the diesel fuel oil day tank is always full, the diesel fuel oil transfer pump flow rate is adequate. The emergency diesel generator's rate of fuel consumption is less than the flow rate of the diesel fuel oil transfer pumps.

2.4.1.3 Evaluation

The licensee has demonstrated that instrumentation is not installed to measure inlet pressure or flow rate for the diesel fuel oil transfer pumps. Also, due to the present system design, the licensee is unable to calculate the flow rate and differential pressure for these pumps. For these reasons, the licensee is unable to measure inlet pressure, differential pressure, and flow rate for the diesel fuel oil transfer pumps in accordance with the requirements of Section XI. However, we feel that without the measurement of these parameters, the licensee may not adequately monitor the hydraulic characteristics of these pumps and therefore detect possible pump degradation. We feel that relief should not be granted from measuring inlet pressure, differential pressure, and flow rate for the diesel fuel oil transfer pumps in accordance with the requirements of Section XI.

2.4.1.4 Conclusion

We conclude that the licensee should consider performing the necessary modifications to the diesel fuel oil transfer system to allow measuring inlet pressure, differential pressure, and flow rate for these pumps in accordance with the requirements of Section XI.

3. VALVE TESTING PROGRAM

The Enrico Fermi Atomic Power Plant, Unit 2, IST program submitted by the Detroit Edison Company was examined to verify that Class 1, 2, and 3 valves that perform a function important to safety were included in the program and that those valves are subjected to the periodic tests required by the ASME Code, Section XI, and the NRC positions and guidelines. Our review found that, except as noted in Attachment 3 or where specific relief from testing has been requested, these valves are tested to the Code requirements and the NRC positions and guidelines summarized in Appendix A and Section 3.1 of this report. Each Detroit Edison Company basis for requesting relief from the valve testing requirements and the EG&G Idaho, Inc., evaluation of that request is summarized below and grouped according to system and valve category.

3.1 <u>General Considerations</u>

3.1.1 Exercising of Check Valves

The NRC's position was stated to the licensee that check valves whose safety function is to open are expected to be full-stroke exercised. Since the disc position is not always observable, the NRC staff position is that verification of the maximum flow rate through the check valve identified in any of the plant's safety analyses would be an adequate demonstration of the full-stroke requirements. Any flow rate less than this will be considered partial-stroke exercising unless it can be shown that the check valve's disc position at the lower flow rate would permit maximum required flow through the valve. It is the NRC staff position that this reduced flow rate method of demonstrating full-stroke capability is the only test that requires measurement of the differential pressure across the valve.

3.1.2 Valves Identified for Cold Shutdown Exercising

The Code permits valves to be exercised during cold shutdowns where it is not practical to exercise them quarterly during plant operation. The licensee has specifically identified the applicable valves and these valves are full-stroke exercised during cold shutdowns; therefore, the licensee is meeting the requirements of the ASME Code. Since the licensee is meeting the requirements of the Code, it is not necessary to grant relief; however, during our review of the IST program, we have verified that it is not practical to exercise these valves during power operation and that we agree with the licensee's cold shutdown justifications.

It should be noted that the NRC differentiates, for valve testing purposes, between the cold shutdown mode and the refueling mode. That is, for valves identified for testing during cold shutdowns, it is expected that the tests will be performed both during cold shutdowns and each refueling outage. However, when relief is granted to perform tests on a refueling outage frequency, testing is expected only during each refueling outage. In addition, for extended refueling outages, tests being performed are expected to be maintained as closely as practical to the Code-specified frequencies.

3.1.3 Conditions for Valve Testing During Cold Shutdowns

Cold shutdown testing of valves identified by the licensee is acceptable when the following conditions are met:

1. The licensee is to commence testing as soon as the cold shutdown condition is achieved, but not later than 48 hours after shutdown, and continue until complete or the plant is ready to return to power.

2. Completion of all valve testing is not a prerequisite to return to power.

3. Any testing not completed during one cold shutdown should be performed during any subsequent cold shutdowns that may occur before refueling to as closely as possible meet the Code-specified testing frequency.

4. For planned cold shutdowns, where ample time is available for testing all the valves identified for the cold shutdown test frequency in the IST program, exceptions to the 48 hours may be taken.

3.1.4 Category A Valve Leak Test Requirements for Containment Isolation Valves (CIVs)

All containment isolation valves that are Appendix J, Type C, leak tested should be included in the IST program as Category A or A/C valves. The NRC staff has concluded that the applicable leak test procedures and requirements for containment isolation valves are determined by 10 CFR 50, Appendix J. Relief from Paragraphs IWV-3421 through -3425 for containment isolation valves presents no safety problem since the intent of IWV-3421 through -3425 is met by Appendix J requirements, however, the licensee shall comply with Paragraphs 'WV-3426 and -3427 unless specific relief is requested from these paragraphs. Based on the considerations discussed above the NRC staff has concluded that this alternate testing will give reasonable assurance of valve leak-tight integrity intended by the Code and that this testing will not endanger life or property or the common defense and security of the public.

3.1.5 Application of Appendix J Testing to the IST Program

The Appendix J review for this plant is completely separate from the IST program review. However, the determinations made by the review are directly applicable to the IST program. The licensee has agreed that, should the Appendix J program be amended, they will amend their IST program accordingly.

3.1.6 Valves Whose Function is Important to Safety

This review was limited to valves whose function is important to safety. Valves whose function is important to safety are defined as those valves that are needed to mitigate the consequences of an accident and/or to shut down the reactor and to maintain the reactor in a shutdown condition. Valves in this category would typically include certain ASME Code Class 1, 2, and 3 valves and could include some non-Code class valves. It should be noted that the licensee may have included valves whose function is not important to safety in their IST program as a decision on their part to expand the scope of their program.

3.1.7 Valves Which Perform a Pressure Boundary Isolation Function

Several safety systems connected to the reactor coolant pressure boundary have design pressures below the reactor coolant system operating pressure. Redundant isolation valves within the Class 1 boundary forming the interface between these high and low pressure systems protect the low pressure systems from pressures which exceed their design limit. In this role, the valves perform a pressure isolation function. The NRC staff considers the redundant isolation provided by these valves to be important and considers it necessary to assure that the condition of each of these valves is adequate to maintain this redundant isolation and system integrity.

The following is a list of valves that appear to perform a pressure isolation function.

Core Spray

E21-F005A, CS injection, Category A

E21-F005B, CS injection, Category A

E21-F006A, CS testable check, Category A/C

E21-F006B, CS testable check, Category A/C

High Pressure Coolant Injection

E41-F006, HPCI injection, Category A

E41-F007, HPCI injection, Category A

Reactor Core Isolation Cooling

E51-F012, RCIC injection, Category A

E51-F013, RCIC injection, Category A

Residual Heat Removal

E11-F008, RHR suction isolation, Category A

E11-F009, RHR suction isolation, Category A

E11-F015B, RHR injection, Category A

E11-F022, RHR reactor head injection, Category A

E11-F023, RHR reactor head injection, Category A

E11-F050B, RHR testable check, Category A/C

E11-F608, RHR suction isolation, Category A

E11-F015A, RHR injection, Category A

E11-F050A, RHR testable check, Category A/C

Detroit Edison has included these valves in the Enrico Fermi, Unit 2, IST program and categorized each valve A or A/C as appropriate and is leak testing the valves in accordance with Technical Specification requirements.

3.2 General Relief Requests

3.2.1 Relief Request

The licensee has requested specific relief from evaluating the stroke times of all active Category A and B solenoid operated valves included in the IST program in accordance with the requirements of Section XI and proposed to verify that the stroke time for these valves does not exceed five seconds.

3.2.1.1 Code Requirement

IWV-3413(b) states, "The stroke time of all power operated valves shall be measured to the nearest second, for stroke times 10 sec or less, or 10% of the specified limiting stroke time for full-stroke times longer than 10 sec whenever such a valve is full-stroke tested."

IWV-3417(a) states, "If, for power operated valves, an increase in stroke time of 25% or more from the previous test for valves with full-stroke times greater than 10 sec or 50% or more for valves with full-stroke times less than or equal to 10 sec is observed, test frequency shall be increased to once each month until corrective action is taken, at which time the original test frequency shall be resumed. In any case, any abnormality or erratic action shall be reported."

3.2.1.2 Licensee's Basis for Requesting Relief

It is impractical to apply the requirements of IWV-3413(b) to valves with very short stroke times (i.e., <5 seconds). Solenoid operated valves typically have full stroke times under one second. For these short stroke time valves, variances of 50 percent or more can occur in the measured times for reasons that are in no way related to valve performance, for example, operator reaction times. In these specific cases, verifying that the valve's stroke time does not exceed 5 seconds would be sufficient to evaluate valve performance.

For solenoid operated valves where position indication is provided, the measured stroke time shall not exceed 5 seconds.

3.2.1.3 Evaluation

We do not agree with the licensee's basis and, therefore, feel that relief should not be granted from the stroke time measurement requirements of Section XI for all active Category A and B solenoid operated valves included in the IST program. The licensee has identified these rapid-acting valves in the IST program and has assigned a maximum stroke time limit of five seconds to each valve, except for valves E11-F414 and E11-F415 in the RHR system for which no maximum stroke time limit has been assigned. It appears that this is a typographical error in the IST program since the two similar valves in the other RHR train (E11-F412 and E11-F413) have a maximum stroke time limit of five seconds, assigned. However, the NRC staff has determined that rapid-acting valves are defined as those valves with stroke times of two seconds or less and that valves with stroke times greater than two seconds should be tested in accordance with the requirements of Section XI.

3.2.1.4 Conclusion

We conclude that the licensee should apply the rapid-acting valve definition only to the active Category A and B solenoid operated valves with stroke times of two seconds or less. The valves that do not fit into this category should be stroke time tested in accordance with the requirements of Section XI.

3.2.2 Relief Request

The licensee has requested specific relief from exercising all instrumentation excess flow check valves (93 valves) included in the IST program in accordance with the requirements of Section XI and proposed verifying closure of these valves (their safety position) during refueling outages.

3.2.2.1 Code Requirement

Refer to Appendix A.

3.2.2.2 Licensee's Basis for Requesting Relief

Excess flow check valves cannot be exercised without isolating instrumentation downstream of the excess flow check valve. Isolating instruments during normal operation would produce erroneous instrument readings which could lead to a degraded or unsafe plant condition.

Excess flow check valves will be exercised in the closed direction at the end of each refueling outage. The exercise test and seat leakage test (AT-3) for these valves will be performed simultaneously.

3.2.2.3 Evaluation

We agree with the licensee's basis and, therefore, feel that relief should be granted from the exercising and leak testing requirements of Section XI for excess flow check valves (93 valves). The licensee has demonstrated that these valves cannot be exercised during power operation because various instrument sensing lines must be isolated thus removing multiple reactor instrumentation from service. Those instruments provide reactor protection and control signals and cannot be removed from service without a possible reactor trip. Additionally, these valves cannot be exercised during cold shutdown because removal of multiple instruments from service could prevent operation of systems required for decay heat removal. We also feel that leak testing excess flow check valves in accordance with Section XI is not required due to the design requirements of Regulatory Guide 1.11.

3.2.2.4 Conclusion

We conclude that verifying closure of these valves (their safety position) during the performance of modified leak rate testing at refueling outages should demonstrate proper valve operability. Based on the considerations discussed above, we conclude that the alternate testing proposed will give reasonable assurance of valve operability intended by the Code and that the relief thus granted will not endanger life or property or the common defense and security of the public.

3.2.3 Relief Request

The licensee has requested specific relief from comparing the measured leakage to a specific maximum leakage for each valve for the following groups of Category A and A/C valves.

1. Containment Isolation Valves that receive a Type C air leak test per 10 CFR 50, Appendix J

2. Containment Isolation Valves that receive a Type C water leak test per 10 CFR 50, Appendix J

3. Main Steam Isolation Valves

4. Valves that are leak tested to demonstrate a bypass leakage isolation function

5. Valves subject to the Purge and Vent Valve leakage test.

3.2.3.1 Code Requirement

IWV-3426 states, "Leakage rate measurements shall be compared with previous measurements and with the permissible leakage rates specified by the plant Owner for a specific valve. If leakage rates are not specified by the Owner, the following rates shall be permissible:

(a) for water, at function pressure differential, 30D ml/hr;

(b) for air, at function pressure differential, 7.5D standard cu ft/day.

D is the nominal valve size, in."

3.2.3.2 Licensee's Basis for Requesting Relief

1. Containment Isolation Valves that receive a Type C air leak test per 10 CFR 50, Appendix J:

   A specific maximum leakage per valve is not applicable to containment isolation valve leakage testing. As long as the sum of the 10 CFR 50, Appendix J, Type B and C leakage is less than 0.6La, the requirements of 10 CFR 50, Appendix J will be satisfied.

The sum of the 10 CFR 50, Appendix J, Type B and C leakage shall be less than 0.6La.

2. Containment Isolation Valves that receive a Type C water leak test per 10 CFR 50, Appendix J:

   A specific maximum leakage per valve is not applicable to containment isolation valve water leakage testing. As long as the sum of the water leakage is less than 5 gpm, the requirements of Technical Specification Paragraph 3.6.1.2.d. will be satisfied.

   The sum of the water leakage from containment isolation valves shall be less than 5 gpm.

3. Main Steam Isolation Valves:

   A specific maximum leakage per valve is not applicable for the MSIV seat leakage (AT-9) tests. As long as the combined leakage from the four main steam lines is less than 100 scfh, then the requirements of Technical Specification Paragraph 3.6.1.2.c. are satisfied.

   The combined leakage from all MSIV's shall be less than the maximum allowable combined leakage of 100 scfh.

4. Valves that are leak tested to demonstrate a bypass leakage isolation function:

   A specific maximum leakage per valve is not applicable to bypass leakage isolation valve testing. As long as the sum of the leakage from all bypass leakage paths is less than 0.04La, the requirements of Branch Technical Position CSB 6-3 will be satisfied. This position has been reviewed and approved by the NRC staff in the EF2 Safety Evaluation Report (NUREG-0798), Supplement 2, pg. 6-1. See Technical Specification Paragraph 3.6.1.2.a for the definition of La.

   Bypass leakage valves subject to Type C tests (AT-1) need not have an additional bypass leakage test (AT-4) performed. The results from the Type C test can be used to determine a particular valve's contribution to the total bypass leakage maximum of 0.04La.

   The sum of the leakage from all bypass leakage paths shall be less than 0.04La.

5. Valves subject to the Purge and Vent Valve leakage test:

   A specific maximum leakage per valve is not applicable to purge and vent valve leakage testing. The leakage criteria for purge and vent valve leakage testing specifies that seat leakage shall not be greater than 1.2 times the previous 10 CFR 50, Appendix J, Type C (AT-1) containment isolation valve leak test result. Since the containment isolation valve leakage test does not have a specific maximum leakage per valve (Relief Request No. VR-11), the specific maximum leakage for an AT-8 test cannot be established until after the Type C (AT-1) test has been performed.

This position has been reviewed and approved by the NRC in Supplement 1 of Enrico Fermi's Safety Evaluation Report (NUREG-0798), pg. 6-3.

Purge and vent valve seat leakage shall not be greater than 1.2 times the previous 10 CFR 50, Appendix J, Type C containment isolation valve leakage test result. The test shall be conducted every 92 days per Technical Specification, Paragraph 4.6.1.8.1.

3.2.3.3 Evaluation

We agree with the licensee's basis and, therefore, feel that relief should be granted from the Section XI requirement of comparing the measured leakage to a specific maximum leakage for each of these valves. We feel that the plant Technical Specifications have adequately established acceptable leakage limits for all of these valves. Additionally, the licensee's Technical Specifications have been previously reviewed and approved by the NRC staff.

3.2.3.4 Conclusion

We conclude that using plant Technical Specifications to establish acceptable leakage limits for these valves should ensure valve leak-tight integrity. Based on the considerations discussed above, we conclude that the alternate testing proposed will give reasonable assurance of valve leak-tight integrity intended by the Code and that the relief thus granted will not endanger life or property or the common defense and security of the public.

3.3 Traversing In-Core Probe

3.3.1 Category A/C Valves

3.3.1.1 Relief Request

The licensee has requested specific relief from exercising valve C51-J009, containment isolation check valve in the nitrogen supply to the TIP system, in accordance with the requirements of Section XI and proposed verifying closure of this valve, its safety position, at a refueling outage interval when it is leak tested.

3.3.1.1.1 Code Requirement

Refer to Appendix A.

3.3.1.1.2 Licensee's Basis for Requesting Relief

During normal operation the purge system supplies nitrogen to the TIP system to prevent corrosion. The only method of exercising this valve closed is by leak testing which can only be performed during reactor refueling. This valve will be exercise tested during reactor refueling.

3.3.1.1.3 Evaluation

We agree with the licensee's basis and, therefore, feel that relief should be granted from the exercising requirements of Section XI for valve C51-J009. The licensee has demonstrated that, due to plant design, the only method available to verify valve closure, the safety position, is leak testing. This valve is not equipped with valve position indication and some of the required test connections are located inside containment.

3.3.1.1.4 Conclusion

We conclude that the proposed alternate testing of verifying valve closure during the performance of leak testing at a refueling outage interval should demonstrate proper valve operability. Based on the considerations discussed above, we conclude that the alternate testing proposed will give reasonable assurance of valve operability intended by the Code and that the relief thus granted will not endanger life or property or the common defense and security of the public.

3.4 Feedwater

3.4.1 Category A/C Valves

3.4.1.1 Relief Request

The licensee has requested specific relief from exercising valves B21-F076A and B21-F076B, feedwater header check valves, in accordance with the requirements of Section XI and proposed verifying these valves open quarterly with feedwater flow and verifying closure of these valves during refueling outages when they are leak tested.

3.4.1.1.1 Code Requirement

Refer to Appendix A.

3.4.1.1.2 Licensee's Basis for Requesting Relief

These check valves cannot be tested for operability to the closed position during reactor operation because the feedwater system is needed to maintain primary coolant inventory. If a feedwater isolation valve was closed during operation, the feedwater nozzle and spargers would undergo a severe thermal shock when feedwater was restored. This thermal shock could cause cracking and possible failure of the spargers and nozzles. When normal feedwater flow is established, these valves are confirmed open. The only means of verifying valve closure is the AT-1 leak test that is performed during refueling outages. Valve closure for these spring-to-close check valves will be verified during refueling outages when the AT-1 leak test is performed.

3.4.1.1.3 Evaluation

We agree with the licensee's basis and, therefore, feel that relief should be granted from the exercising requirements of Section XI for valves B21-F076A and B21-F076B. The licensee has demonstrated that these valves cannot be exercised shut during power operation since this would interrupt feedwater flow to the reactor, which could result in a plant trip. Due to the design of these valves, the only method available for verifying valve closure is by leak testing. These valves are leak tested during refueling outages in accordance with the requirements of Appendix J. Performing this leak testing during cold shutdowns could result in delaying plant startup from the cold shutdown condition.

3.4.1.1.4 Conclusion

We conclude that verifying these valves open quarterly with feedwater flow and verifying these valves closed during refueling outages when they are leak tested in accordance with the requirements of Appendix J should demonstrate proper valve operability. Based on the consideration discussed above, we conclude that the alternate testing proposed will give reasonable assurance of valve operability intended by the Code and that the relief thus granted will not endanger life or property or the common defense and security of the public.

3.5 Core Spray

3.5.1 Category C Valves

3.5.1.1 Relief Request

The licensee has requested specific relief from verifying closure individually for valves E21-F029A, E21-F029B, E21-F030A, and E21-F030B, check valves in the keep fill system lines to the core spray system, in accordance with the requirements of Section XI and proposed verifying closure of each pair of valves (E21-F029A and E21-F030A; E21-F029B and E21-F030B) quarterly.

3.5.1.1.1 Code Requirement

Refer to Appendix A.

3.5.1.1.2 Licensee's Basis for Requesting Relief

The E21-F029A and E21-F030A valves are placed in series as are the E21-F029B and E21-F030B valves. There are no taps between these valves and no manual lifting levers to indicate disc position. The only way to verify valve closure is to check for reverse flow leakage, which will confirm that one out of two valves closed. The valves are exercised to the open position during the vent and fill portion of the core spray system operability tests. One of the two valves will be confirmed closed by the absence of reverse flow leakage.

3.5.1.1.3 Evaluation

We agree with the licensee's basis and, therefore, feel that relief should be granted from the requirements of Section XI for verifying closure of valves E21-F029A, E21-F029B, E21-F030A, and E21-F030B individually. The licensee has demonstrated, that due to the present system design, no means exist to verify closure of each valve. Since these valves are two pairs of series valves, verifying closure of each pair of valves, quarterly, will demonstrate that the intended safety function of preventing reverse flow from the core spray system to the keep fill system is being met.

3.5.1.1.4 Conclusion

We conclude that verifying each valve open quarterly and verifying closure of each pair of series valves (E21-F029A and E21-F030A; E21-F029B and E21-F030B) quarterly should demonstrate proper valve operability. Based on the considerations discussed above, we conclude that the alternate testing proposed will give reasonable assurance of valve operability intended by the Code and that the relief thus granted will not endanger life or property or the common defense and security of the public.

3.6 <u>High Pressure Coolant Injection</u>

3.6.1 Category C Valves

3.6.1.1 Relief Request

The licensee has requested specific relief from exercising valve E41-F045, check valve in the HPCI suction line from the suppression pool, in accordance with the requirements of Section XI and proposed disassembling this valve during each refueling outage to demonstrate proper valve operability until sufficient data is accumulated to justify an inspection interval between tests longer than each refueling outage.

3.6.1.1.1 Code Requirement

Refer to Appendix A.

3.6.1.1.2 Licensee's Basis for Requesting Relief

There is no convenient method for verifying the ability of this valve to swing to the full open position. The system test circuits utilize the condensate storage tank for pump suction rather than the suppression pool. Taking suction from the suppression pool during testing is undesirable because torus water would be transferred to the condensate storage tank. Since torus water is not demineralized, the entire condensate storage tank inventory would have to be processed after the test.

In lieu of the Code required full stroke test, valve operability will be demonstrated by disassembling the valve and verifying that the valve disc swings freely to the full open position. Since this valve has no function during normal operation, no wear-induced degradation of the valve internals is expected.

Valve disassembly and inspection will occur at every refueling outage until sufficient data can be accumulated to adequately monitor valve degradation. The maximum inspection interval will be determined based on the results of that data.

3.6.1.1.3 Evaluation

We agree with the licensee's basis and, therefore, feel that relief should be granted from the exercising requirements of Section XI for valve E41-F045. The licensee has demonstrated that this valve cannot be exercised with flow since this would result in transferring non-demineralized water from the torus to the condensate storage tank, which would require processing the entire condensate storage tank inventory. As an alternate means of full-stroke exercising this valve, the licensee has proposed disassembling this valve to verify freedom of disc movement during each refueling outage until sufficient data is accumulated to determine an acceptable maximum inspection interval. We feel that this proposed alternate testing method will demonstrate proper valve operability, however, the licensee will need to provide the NRC staff with the results of the inspections before any inspection interval between tests longer than each refueling outage can be accepted.

3.6.1.1.4 Conclusion

We conclude that the proposed alternate testing of disassembling this valve during each refueling outage to verify freedom of disc movement should demonstrate proper valve operability. Based on the considerations discussed above, we conclude that the alternate testing proposed will give reasonable assurance of valve operability intended by the Code and that the relief thus granted will not endanger life or property or the common defense and security of the public.

3.6.1.2 Relief Request

The licensee has requested specific relief from individually verifying closure of valves E41-F076 and E41-F077, HPCI exhaust line vacuum breaker valves, and proposed verify closure of the pair of valves quarterly (i.e., the licensee will confirm that at least one of the two valves has closed properly).

3.6.1.2.1 Code Requirement

Refer to Appendix A.

3.6.1.2.2 Licensee's Basis for Requestig Relief

These valves will be exercised during the HPCI pump/turbine performance test. Based on the present valve configuration, the following indirect means of position verification is being used. The absence of a vacuum condition in the turbine exhaust line after the turbine has been tripped will confirm that both of these valves have stroked open. The absence of steam in the torus air space, as indicated by the containment monitoring system, during the HPCI pump/turbine performance test will confirm that one of the two valves has closed properly.

Both valves will be confirmed open by the absence of a vacuum in the turbine exhaust line and one of the two valves will be confirmed closed by the absence of steam in the torus air space.

3.6.1.2.3 Evaluation

We agree with the licensee's basis and, therefore, feel that relief should be granted from the Section XI requirement of verifying closure of each of these valves individually. Due to the present system design, there is no available means of verifying closure individually for these valves. These valves are simple check valves that do not have operators or valve position indication. We feel that the licensee's proposed alternate testing method of verifying at least one of these two valves closed by observing an absence of steam in the torus air space should demonstrate the intended safety function in the closed position for these valves.

3.6.1.2.4 Conclusion

We conclude that the proposed alternate testing of verifying closure of the pair of valves (i.e., at least one of the two valves has closed) by observing the absence of steam in the torus air space quarterly after the HPCI pump/turbine performance test and verifying that these valves open quarterly by observing an absence of a vacuum in the turbine exhaust line after the HPCI pump/turbine performance test should demonstrate the intended safety function of these valves. Based on the considerations discussed above, we conclude that the alternate testing proposed will give reasonable assurance of valve operability intended by the Code and that the relief thus granted will not endanger life or property or the common defense and security of the public.

3.7 Control Rod Drive Hydraulic

3.7.1 Category B and C Valves

3.7.1.1 Relief Request

The licensee has requested specific relief from exercising valves C11-126, C11-127, and C11-114, inlet and outlet scram valves for each of the 185 hydraulic control units (C11-126 and C11-127) and scram discharge header check valves for each of the 185 hydraulic control units (C11-114), in accordance with the requirements of Section XI and proposed full-stroke exercising these valves during the individual control rod scram insertion testing that is performed in accordance with Technical Specification requirements.

3.7.1.1.1 Code Requirement

Refer to Appendix A.

3.7.1.1.2 Licensee's Basis for Requesting Relief

The proper operation of each of these valves is demonstrated during scram testing. During scram testing, each drive's scram insertion time is measured and a fail-safe actuator test is performed. The Technical Specifications provide a limit for individual CRD scram insertion times to specific values (Technical Specification paragraphs 3.1.3.3 and 3.1.3.4). If a particular CRD's scram insertion time is less than the specified limit, the above mentioned valves are functioning properly.

The frequency of individual scram insertion tests is: 1) 100% of control rod drives following core alternations or after a reactor shutdown greater than 120 days with reactor power equal to or less than 40% and 2) 10% of control rods at least once every 120 days of operations, per Technical Specification paragraph 4.1.3.2.

3.7.1.1.3 Evaluation

We agree with the licensee's basis and, therefore, feel that relief should be granted from the exercising requirements of Section XI for valves C11-126, C11-127, and C11-114. Exercising these valves would require that each control unit be individually scrammed. Individual control rod scram testing is conducted in accordance with Technical Specification requirements and results in all control rods being tested at least once every operating cycle. These valves must operate properly to ensure that the associated control rod meets the scram insertion time limit defined in the Technical Specifications.

3.7.1.1.4 Conclusions

We conclude that the proposed alternate testing of verifying proper control rod scram insertion times during the performance of control rod scram testing in accordance with Technical Specifications is an acceptable method for monitoring valve degradation and demonstrating proper valve operability. Based on the considerations discussed above, we conclude that the alternate testing proposed will give reasonable assurance of valve operability intended by the Code and that the relief thus granted will not endanger life or property or the common defense and security of the public.

3.7.1.2 Relief Request

The licensee has requested specific relief from exercising valves C11-115 and C11-138, control rod drive charging water header and cooling water header check valves for each of the 185 hydraulic control units, in accordance with the requirements of Section XI and proposed verifying closure of these valves (their safety position) during the individual control rod scram insertion testing that is performed in accordance with Technical Specification requirements.

3.7.1.2.1 Code Requirement

Refer to Appendix A.

3.7.1.2.2 Licensee's Basis for Requesting Relief

The proper operation of each of these valves is demonstrated during scram testing. During scram testing, each drive's scram insertion time is measured and a fail-safe actuator test is performed. The Technical Specifications provide a limit for individual CRD scram insertion times to specific values (Technical Specification paragraphs 3.1.3.3 and 3.1.3.4). If a particular CRD's scram insertion time is less than the specified limit, the above mentioned valves are functioning properly.

The frequency of individual scram insertion tests is: 1) 100% of control rod drives following core alterations or after a reactor shutdown greater that 120 days with reactor power equal to or less than 40% and 2) 10% of control rods at least once every 120 days of operations, per Technical Specification paragraph 4.1.3.2.

3.7.1.2.3 Evaluation

We do not agree with the licensee's basis and, therefore, feel that relief should not be granted from the exercising requirements of Section XI for valves C11-115 and C11-138. We do not feel that the license can be assured of proper valve closure unless the control rod drive charging header and cooling water header are depressurized. These headers are not depressurized during the individual control rod scram testing.

3.7.1.2.4 Conclusion

We conclude that the licensee should either exercise these valves in accordance with the requirements of Section XI or provide the NRC staff with additional technical information which demonstrates that closure of these valves is being positively verified during the performance of the individual control rod scram insertion testing.

3.8 Residual Heat Removal

3.8.1 Category C Valves

3.8.1.1 Relief Request

The licensee has requested specific relief from individually verifying closure of valves E11-F089, E11-F090, E11-F184, and E11-F185, check valves in the keep fill system lines to the residual heat removal system, in accordance with the requirements of Section XI and proposed verifying closure of each pair of valves (E11-F089 and E11-F090; E11-F184 and E11-F185) quarterly.

3.8.1.1.1 Code Requirement

Refer to Appendix A.

3.8.1.1.2 Licensee's Basis for Requesting Relief

The E11-F089 and E11-F090 valves are placed in series as are the E11-F184 and E11-F185 valves. There are no test taps between these valves and no manual lifting levers to indicate disc position. The only way to verify valve closure is to check for reverse flow leakage, which will confirm that one out of two valves closed. The valves are exercised to the open position during the vent and fill portion of the residual heat removal system operability tests. One of the two valves will be confirmed closed by the absence of reverse flow leakage.

3.8.1.1.3 Evaluation

We agree with the licensee's basis and, therefore, feel that relief should be granted from the requirements of Section XI for verifying closure of valves E11-F089, E11-F090, E11-F184, and E11-F185 individually. The licensee has demonstrated, that due to the present system design, no means exist to verify closure of each valve. Since these valves are two pairs of series valves, verifying closure of each pair of valves, quarterly, will demonstrate tht the intended safety function of preventing reverse flow from the residual heat removal system to the keep fill system is being met.

3.8.1.1.4 Conclusion

We conclude that verifying each valve open quarterly and verifying closure of each pair of series valves (E11-F089 and E11-F090; E11-F184 and E11-F185) quarterly should demonstrate proper valve operability. Based on the considerations discussed above, we conclude that the alternate testing proposed will give reasonable assurance of valve operability intended by the Code and that the relief thus granted will not endanger life or property or the common defense and security of the public.

3.9 Combustible Gas Control

3.9.1 Category B Valves

3.9.1.1 Relief Request

The licensee has requested specific relief from exercising valves T48-F001A and T48-F001B, isolation valves for the RHR water supply to the water-spray cooler in the combustible gas control system, in accordance with the requirements of Section XI and proposed full-stroke exercising these valves every six months during the combustible gas control system operability tests.

3.9.1.1.1 Code Requirement

Refer to Appendix A.

3.9.1.1.2 Licensee's Basis for Requesting Relief

These valves automatically open upon initiation of the combustible gas control system. There is no manual means of stroking these valves. Therefore, these valves will be exercised and timed during the combustible gas control system operability tests, which are performed every six (6) months (Technical Specification paragraph 4.6.6.1).

3.9.1.1.3 Evaluation

The licensee has demonstrated that the only feasible means of exercising valves T48-F001A and T48-F001B is by initiation of the combustible gas control system. However, the licensee has not supplied any technical justification for not performing the combustible gas control system operability tests quarterly to demonstrate proper operability of these valves; therefore, we feel that relief should not be granted from the exercising requirements of Section XI for these valves.

3.9.1.1.4 Conclusion

We conclude that the licensee should either full-stroke exercise these valves quarterly in accordance with the requirements of Section XI or provide the NRC staff with additional technical information indicating why these valves cannot be full-stroke exercised quarterly.

3.10 Nuclear Boiler

3.10.1 Category B/C Valves

3.10.1.1 Relief Request

The licensee has requested specific relief from exercising and measuring stroke times for valves B21-F013E, B21-F013H, B21-F013J, B21-F013P, and B21-F013R, primary system safety-relief valve (S/RV) ADS valves, in accordance with the requirements of Section XI and proposed exercising these valves once every 18 months by observing changes in steam flow and/or turbine bypass valve position to insure that these valves have stroked in less than or equal to five seconds.

3.10.1.1.1 Code Requirement

Refer to Appendix A.

3.10.1.1.2 Licensee's Basis for Requesting Relief

Relief is requested from the Section XI required testing frequency of once every three months. These valves will be exercised once every eighteen (18) months as specified in Technical Specification paragraph 4.5.1.d.2.

In addition, relief is requested from the stroke timing requirements of Section XI. It is impractical to measure stroke times for a S/RV since the stroke times are on the order of 100 mS. Steam flow measurements and/or turbine bypass valve position will verify that the S/RVs have performed their function in less than or equal to 5 seconds. Time "zero" for this stroke time measurement corresponds to the instant the S/RV hand switch is aligned to the "open" position.

These valves will be exercised at least once per 18 months when the reactor is operating at sufficient power to bypass a quantity of steam through the turbine bypass valve(s) equal to or greater than the capacity of a S/RV. Since the turbine bypass valves respond automatically to RPV dome pressure, the actuation of a S/RV will result in rapid closure of the turbine bypass valves. Conversely, closing the S/RV will be accommodated by rapid opening of the turbine bypass valves. A change in turbine bypass valve position can be directly associated with a certain steam flow rate. The flow rate would be equal to the quantity of steam discharged by the S/RV.

No stroke time measurements will be performed. An abrupt change in turbine bypass valve position or steam line flow (per Tech. Spec. 4.5.1.d.2) within 5 seconds will be adequate to demonstrate valve operability.

3.10.1.1.3 Evaluation

Although the NRC staff does not agree with the applicants' basis for requesting relief from testing of the valves quarterly in accordance with the code specified frequency, there are other safety related reasons for exercising the valves only at 18 month intervals. If the valves were to fail to reclose after testing, the plant would be placed in a LOCA condition. In addition, a recent study (BWR Owners Group Evaluation of NUREG-0737, II.K.3.16 Reduction of Challenges and Failures of Relief Valves) recommends that the number of SRV valve openings be reduced as much as possible.

3.10.1.1.4 Conclusion *

Based on these considerations we conclude that relief should be granted to exercise these valves once every eighteen months as specified in Technical Specification paragraph 4.5.1.d.2.

3.10.2 Category C Valves

3.10.2.1 Relief Request

The licensee has requested specific relief from exercising valves B21-F024A, B21-F024B, B21-F024C, B21-F024D, B21-F029A, B21-F029B, B21-F029C, B21-F029D, B21-F036A, B21-F036B, B21-F036C, B21-F036E, and B21-F036G, check valves in the air or nitrogen supply lines to the MSIV accumulators and the S/RV ADS accumulators, in accordance with the requirements of Section XI and proposed full-stroke exercising these valves during refueling outages.

3.10.2.1.1 Code Requirement

Refer to Appendix A.

3.10.2.1.2 Licensee's Basis for Requesting Relief

The position of these simple check valves cannot be verified during normal operation or cold shutdown since special testing will be required. In addition, access to these valves is limited since they are inside the drywell. These valves will be exercised during refueling.

3.10.2.1.3 Evaluation

We agree with the licensee's basis and, therefore, feel that relief should be granted from the exercising requirements of Section XI for these valves. The licensee has demonstrated that these valves cannot be exercised during power operation since a drywell entry would be required to perform the testing. These valves are simple check valves that do not have an external operator or valve position indication. Also, during cold shutdowns, exercising these valves would require de-inerting the drywell and could result in delaying startup from the cold shutdown condition due to the special testing required to full-stroke exercise these valves.

3.10.2.1.4 Conclusion

We conclude that full-stroke exercising these valves during each refueling outage should demonstrate proper valve operability. Based on the considerations discussed above, we conclude that the alternate testing proposed will give reasonable assurance of valve operability intended by the Code and that the relief thus granted will not endanger life or property or the common defense and security of the public.

3.10.2.2 Relief Request

The licensee has requested specific relief from exercising valves B21-F037A, B21-F037B, B21-F037C, B21-F037D, B21-F037E, B21-F037F, B21-F037G, B21-F037H, B21-F037J, B21-F037K, B21-F037L, B21-F037M, B21-F037N, B21-F037P, and B21-F037R, S/RV discharge line vacuum breakers, in accordance with the requirements of Section XI and proposed full-stroke exercising these valves during cold shutdowns when the drywell is de-inerted.

3.10.2.2.1 Code Requirement

Refer to Appendix A.

3.10.2.2.2 Licensee's Basis for Requesting Relief

These check valves have no external means of actuation for exercising. The only practical method for exercising these valves open and closed is by manual pushing the disc from its seat. Since this requires access to the valves, which are located in the drywell, the test must be deferred to cold shutdowns if the primary containment is de-inerted.

These check valves will be verified to freely swing to their full open and closed positions during cold shutdowns when the drywell is de-inerted.

3.10.2.2.3 Evaluation

We agree with the licensee's basis and, therefore, feel that relief should be granted from the exercising requirement of Section XI for these valves. The licensee has demonstrated that the only practical method for exercising these valves would require an entry into the drywell. The drywell is required to be inerted during power operation and may not be routinely de-inerted during cold shutdowns.

3.10.2.2.4 Conclusion

We conclude that full-stroke exercising these valves during cold shutdowns, if the drywell is de-inerted, and during refueling outages should demonstrate proper valve operability. Based on the considerations discussed above, we conclude that the alternate testing proposed will give reasonable assurance of valve operability intended by the Code and that the relief thus granted will not endanger life or property or the common defense and security of the public.

3.11 Emergency Equipment Cooling Water

3.11.1 Category A/C Valves

3.11.1.1 Relief Request

The licensee has requested specific relief from exercising valves P44-F282A and P44-F282B, inside containment isolation check valves for the EECW lines, in accordance with the requirements of Section XI and proposed verifying closure of these valves (their safety position) during refueling outages when they are leak tested.

3.11.1.1.1 Code Requirement

Refer to Appendix A.

3.11.1.1.2 Licensee's Basis for Requesting Relief

During power operation, the reactor building closed cooling water system supplies cooling water to components inside the drywell, including the reactor recirculating pumps and motors. Closing the subject valves would interrupt cooling water flow to the reactor recirculating pump and motor bearings. These valves will not be exercised during normal operation because interruption of flow may cause damage to the pump and motor.

These valves will be verified to close during the AT-1 test every reactor refueling cycle.

3.11.1.1.3 Evaluation

We agree with the licensee's basis and, therefore, feel that relief should be granted from the exercising requirements of Section XI for valves P44-F282A and P44-F282B. Due to plant design, the only method available to verify valve closure (the safety position) is leak testing. These valves are not equipped with valve position indication and some of the required test connections are located inside containment.

3.11.1.1.4 Conclusion

We conclude that the proposed alternate testing of verifying valve closure during the performance of leak testing at refueling outages should demonstrate proper valve operability. Based on the considerations discussed above, we conclude that the alternate testing proposed will give reasonable assurance of valve operability intended by the Code and that the relief thus granted will not endanger life or property or the common defense and security of the public.

3.11.2 Category C Valves

3.11.2.1 Relief Request

The licensee has requested specific relief from exercising valves P44-F246 and P44-F274, check valves in the return lines from the penetration cooling jackets and the drywell sump heat exchanger, in accordance with the requirements of Section XI and proposed verifying closure of these valves (their safety position) by leak testing during refueling outages.

3.11.2.1.1 Code Requirement

Refer to Appendix A.

3.11.2.1.2 Licensee's Basis for Requesting Relief

Because these valves are non-testable check valves inside primary containment, they can only be verified closed by a leak test. This test can only be performed during reactor refueling.

These valves will be leak tested during reactor refueling to confirm their close position.

3.11.2.1.3 Evaluation

We agree with the licensee's basis and, therefore, feel that relief should be granted from the exercising requirements of Section XI for valves P44-F246 and P44-F274. The licensee has demonstrated that, due to plant design, the only method available to verify valve closure (the safety position) is leak testing. These valves are not equipped with valve position indication and some of the required test connections are located inside containment.

3.11.2.1.4 Conclusion

We conclude that the proposed alternate testing of verifying valve closure during the performance of leak testing at refueling outages should demonstrate proper valve operability. Based on the considerations discussed above, we conclude that the alternate testing proposed will give reasonable assurance of valve operability intended by the Code and that the relief thus granted will not endanger life or property or the common defense and security of the public.

APPENDIX A-1

1. CODE REQUIREMENTS--VALVES

Subsection IWV-3411 of the Section XI Code requires that Code Category A and B valves be exercised once every three months, with the exceptions as defined in IWV-3412(a). IWV-3521 requires that Code Category C valves be exercised once every three months, with the exceptions as defined in IWV-3522. IWV-3700 contains test requirements for active and passive valves. The limiting value of full stroke time for each power operated valve shall be identified by the owner and tested in accordance with IWV-3413(a), (b), and IWV-3417(a), (b). In the above exceptions, the Code permits the valves to be tested at cold shutdown where:

1. It is not practical to exercise the valves to the position required to fulfill their function or to the partial position during power operation.

2. It is not practical to observe the operation of the valves (with fail-safe actuators) upon loss of actuator power.

Subsection IWV-3413(b) requires all Category A and B powered-operated valves to be stroke-time tested to the nearest second or 10% of the maximum allowable owner-specified time.

2. CODE REQUIREMENTS--PUMPS

An inservice test shall be conducted on all pumps whose function is important to safety, nominally once each quarter during normal plant operation. Each inservice test shall include the measurement, observation, and recording of all quantities in Table IWP-3100-1, except bearing temperature, which shall be measured during at least one inservice test each year.

ATTACHMENT 1

The following are Category A, B, and C valves that meet the exercising requirements of the ASME Code, Section XI, and are not full-stroke exercised every three months during plant operation. These valves are specifically identified by the owner and are full-stroke exercised during cold shutdowns and refueling outages. EG&G Idaho, Inc., has reviewed all valves in this attachment and agrees with the licensee that testing these valves during power operation is not practical, due to the valve type and location or system design. We feel that these valves cannot or should not be exercised during power operation. These valves are listed below and grouped according to the system in which they are located.

1. MAIN AND REHEAT STEAM

1.1 Category B Valves

B21-F041A, B21-F041B, B21-F041C, and B21-F041D, blocking valves for the MSIV leakage control system, cannot be exercised during power operation since closure of any of these valves could result in primary system pressure spikes, reactor power fluctuations, and increased flow in the unisolated steam lines. This unstable operation could lead to a reactor scram. In addition, pressure transients resulting from exercising these valves would increase the chances of actuating primary system safety/relief valves. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

2. FEEDWATER

2.1 Category A/C Valves

B21-F010A and B21-F010B, feedwater header check valves, cannot be exercised shut during power operation because the feedwater system is needed to maintain primary coolant inventory. Also, if these valves were closed during power operation, the feedwater nozzles and spargers would undergo a severe thermal shock when feedwater was restored. Finally, the air operators on these testable check valves cannot close the valves against feedwater flow. These valves are verified open quarterly with feedwater flow and will be full-stroke exercised closed with the air operators during cold shutdowns and refueling outages.

3. SUMP PUMP RADWASTE

3.1 Category A Valves

G11-F018 and G11-F600, inside containment isolation valves for the sump pump radwaste headers, cannot be exercised during power operation since failure of these valves in the closed position during exercising would require a plant shutdown to correct the problem since these valves are located inside containment. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

4. CORE SPRAY

4.1 Category A and A/C Valves

E21-F005A, E21-F005B, E21-F006A, and E21-F006B, core spray injection header isolation valves, cannot be exercised during power operation since these valves have interlocks which require the primary system pressure to be below the core spray system design pressure prior to opening. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

5. HIGH PRESSURE COOLANT INJECTION

5.1 Category A Valves

E41-F002, inside containment isolation valve in the steam supply line to the HPCI turbine, cannot be exercised during power operation since failure of this valve in the closed position during exercising would require a plant shutdown to correct the problem since this valve is located inside containment. This valve will be full-stroke exercised during cold shutdowns and refueling outages.

E41-F006, HPCI injection line isolation valve, cannot be exercised during power operation since opening this valve with the HPCI pump secured could result in overpressurizing the suction side of the HPCI pump and opening this valve with the HPCI pump running would result in thermal shock to the HPCI injection nozzle. This valve will be full-stroke exercised during cold shutdowns and refueling outages.

6. REACTOR CORE ISOLATION COOLING

6.1 Category A Valves

E51-F007, inside containment isolation valve in the steam supply line to the RCIC turbine, cannot be exercised during power operation since failure of this valve in the closed position during exercising would require a plant shutdown to correct the problem since this valve is located inside containment. This valve will be full-stroke exercised during cold shutdowns and refueling outages.

E51-F013, RCIC injection line isolation valve, cannot be exercised during power operation since opening this valve with the RCIC pump secured could result in overpressurizing the suction side of the RCIC pump and opening this valve with the RCIC pump running would result in thermal shock to the RCIC injection nozzle. This valve will be full-stroke exercised during cold shutdowns and refueling outages.

7. REACTOR WATER CLEAN-UP

7.1 Category A Valves

G33-F001, reactor water clean-up system inside containment isolation valve, cannot be exercised during power operation since failure of this valve in the closed position during exercising would require a plant shutdown to correct the problem since this valve is located inside containment. This valve will be full-stroke exercised during cold shutdowns and refueling outages.

8. STANDBY LIQUID CONTROL

8.1 Category A/C Valves

C41-F006 and C41-F007, standby liquid control injection line containment isolation valves, cannot be exercised during power operation. The air operator on valve C41-F007 cannot move the valve disc to the full open position with the reactor coolant system at 1000 psig. For valve C41-F006, the licensee has no method for determining the differential pressure across the disc. If a large differential pressure existed across the disc of this valve, damage to the air operator could result if this valve was exercised. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

9. RESIDUAL HEAT REMOVAL

9.1 Category A Valves

E11-F008, E11-F009, and E11-F608, isolation valves from the reactor recirculation loop to the RHR pump suctions, cannot be exercised during power operation since these valves have interlocks which require the primary system pressure to be below the residual heat removal system design pressure prior to opening. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

E11-F015A, E11-F015B, E11-F022, and E11-F023, isolation valves on the RHR injection lines to the reactor recirculation loops and the reactor vessel head, cannot be exercised during power operation since these valves have interlocks which require the primary system pressure to be below the residual heat removal system design pressure prior to opening. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

E11-F412, E11-F413, E11-F414, and E11-F415, primary containment isolation system instrumentation valves, cannot be exercised during power operation since closure of these valves could result in a reactor scram and ECCS initiation due to isolation of safety-related instrumentation. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

9.2 Category A/C Valves

E11-F050A and E11-F050B, isolation valves on the RHR injection lines to the reactor recirculation loops, cannot be exercised during power operation since these valves have interlocks which require the primary system pressure to be below the residual heat removal system design pressure prior to opening. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

10. NUCLEAR BOILER

10.1 Category A Valves

B21-F022A, B21-F022B, B21-F022C, B21-F022D, B21-F028A, B21-F028B, B21-F028C, and B21-F028D, main steam isolation valves, cannot be full-stroke exercised during power operation since closure of any of these valves would result in primary system pressure spikes, reactor power fluctuations, and increased flow in the unisolated steam lines which could cause a reactor scram. These valves will be partial-stroke exercised quarterly and full-stroke exercised during cold shutdowns and refueling outages.

10.2 Category B Valves

B21-F003 and B21-F004, reactor vessel head vent valves, cannot be exercised during power operation since opening one of these valves would result in the other valve being the only barrier between the reactor vessel and the drywell sump. If the closed valve was not leak tight, the drywell could be pressurized. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

11. REACTOR RECIRCULATION

11.1 Category A Valves

B31-F014A, B31-F014B, B31-F016A, and B31-F016B, containment isolation valves on the CRD seal water injection lines to the reactor recirculation pumps, cannot be exercised during power operation since this would require isolating the seal water flow to the reactor recirculation pumps, which could result in damaging these pumps. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

11.2 Category B Valves

B31-F031A and B31-F031B, reactor recirculation pumps discharge valves, cannot be exercised during power operation since failure of either valve in the closed position would require a shutdown to correct the problem. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

12. EMERGENCY EQUIPMENT COOLING WATER

12.1 Category A Valves

P44-F606A, P44-F606B, P44-F607A, P44-F607B, P44-F615, and P44-F616, primary containment isolation valves for the four EECW lines, cannot be exercised during power operation since closure of these valves would interrupt cooling water to the reactor recirculating pump and motor bearings, which could result in damage to the pumps and motors. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

12.2 Category B Valves

P44-F601A, P44-F601B, P44-F603A, and P44-F603B, isolation valves between the reactor building closed cooling water system and the emergency equipment cooling water system, cannot be exercised during power operation since this would interrupt cooling water flow to the reactor recirculating pump and motor bearings, which could result in damage to the pumps and motors. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

P44-F608 and P44-F614, EECW drywell sump cooling and EECW penetration jacket isolation valves, cannot be exercised during power operation since failure of these valves in the closed position during exercising would require a plant shutdown to correct the problem since these valves are located inside containment. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

P44-F604, isolation valve in the cooling water supply line to the CRD pumps, cannot be exercised during power operation since closure of this valve would interrupt cooling water flow to the CRD pumps, which could result in damage to these pumps. This valve will be full-stroke exercised during cold shutdowns and refueling outages.

12.3 Category C Valves

P44-F182, check valve in the cooling water return line from the CRD pumps, cannot be exercised during power operation since closure of this valve would interrupt cooling water flow to the CRD pumps, which could result in damage to these pumps. This valve will be full-stroke exercised during cold shutdowns and refueling outages.

ATTACHMENT 2

The P&IDs listed below were used during the course of this review.

System	P&ID	Revision
Traversing In-Core Probe System	6I721-2145-66	A
Post Accident Sampling	6I721-2400-10	O
Primary Containment Monitoring System	6I721-2679-1	G
Main and Reheat Steam Systems	6M721-2002	O
Station and Control Air System	6M721-2015	N
Feedwater System	6M721-2023	N
Sump Pump - Radwaste System	6M721-2032	R
Core Spray System	6M721-2034	N
High Pressure Coolant Injection System	6M721-2035	N
High Pressure Coolant Injection System (Barometric Condenser)	6M721-2043	H
Reactor Core Isolation Cooling System	6M721-2044	M
Reactor Core Isolation Cooling (Barometric Condenser)	6M721-2045	J
Reactor Water Clean-Up	6M721-2046	M
Fuel Pool Cooling & Clean-Up System	6M721-2048	G
Control Rod Drive Hydraulic System	6M721-2081	K
Stand-By Liquid Control System	6M721-2082	M
Residual Heat Removal - Division II	6M721-2083	M
Residual Heat Removal - Division I	6M721-2084	P
Station Air Risers	6M721-2085	E
Combustible Gas Control System	6M721-2087	E
Nuclear Boiler System	6M721-2089	I

ATTACHMENT 2 (cont'd)

System	P&ID	Revision
Nuclear Boiler System (Instrumentation)	6M721-2090	F
Demineralized Service Water Risers	6M721-2678	0
Reactor Building & Auxiliary Building Ventilation System	6M721-2707	D
Reactor Recirculation System	6M721-2833	H
Main Steam Isolation Valve Leakage Control System	6M721-3045	E
Nitrogen Inerting System	6M721-3445	H
Torus Water Management System	6M721-4100	H
Water Side Control Center A/C	6M721-4325	I
Interruptible and Non-Interruptible Control Air	6M721-4615	B
Primary Containment Pneumatic Supply System	6M721-5007	B
Emergency Equipment Cooling Water - Division II	6M721-5357	A
Emergency Equipment Cooling Water - Division I	6M721-5444	A
Control Rod Drive Hydraulic System - Part 2	6M721-5449	0
Diesel Generator System - Division I - R.H.R. Complex	6M721N-2046	M
Diesel Generator System - Division II - R.H.R. Complex	6M721N-2047	N
Diesel Fuel Oil System & Lube Oil System - Division I - R.H.R. Complex	6M721N-2048	S
Diesel Fuel Oil System & Lube Oil System - Division II - R.H.R. Complex	6M721N-2049	T
R.H.R. - Service Water System - Division I - R.H.R. Complex	6M721N-2052	0
R.H.R. - Service Water System - Division II - R.H.R. Complex	6M721N-2053	R

ATTACHMENT 2 (cont'd)

System	P&ID	Revision
Service Water, Make-Up, Decant, & Overflow Systems	6M721N-2054	K
Standby Gas Treatment and Primary Containment Purge System	7M721-2709	F

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Docket No. 50-341

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Supplement No. 4 to the Safety Evaluation Report related to the operation of the Enrico Fermi Atomic Power Plant, Unit 2, provides the staff's evaluation of additional information submitted by the application regarding outstanding review issues identified in Supplement No. 3 to the Safety Evaluation Report, dated January 1983.

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