NUREG-0798
Supplement No. 5

Safety Evaluation Report
related to the operation of
Fermi-2

Docket No. 50-341

Detroit Edison Company, et al.

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

March 1985

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ABSTRACT

Supplement No. 5 to the Safety Evaluation Report (SER) related to the operation of the Fermi-2 facility, provides the NRC staff's evaluation of additional information submitted by the applicant regarding outstanding review issues identified in Supplement No. 4 to the SER dated September 1984. This supplement contains the staff's conclusion that there are no outstanding issues which must be resolved prior to issuance of a low-power operating license (i.e., less than five percent of full rated power) for the Fermi-2 facility. Supplement No. 5 to the SER also summarizes the conditions which are placed in the Fermi-2 operating license.

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APPENDICES

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, 3 and 4 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 59 through 60 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 4 to the SER.

This supplement (Supplement 5 to the SER) provides the staff's evaluation of additional information provided by the applicant in FSAR amendments through Amendment 60 and by the 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 by 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 organizations which provided consultants to the staff are listed below. The individual consultants are also listed in Appendix G.

Brookhaven National Laboratory
Franklin Research Center
Idaho National Engineering Laboratory (EG&G Idaho, Inc.)

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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 4 to the SER is described in appropriate sections of this supplement. There are no outstanding issues remaining in our review which must be resolved prior to issuance of the low-power Fermi-2 operating license. There is only one issue which must be resolved prior to issuance of the Fermi-2 full power license. This is a confirmatory item by FEMA related to the final version of the Monroe County Emergency Plan. (Refer to Section 13.3.5 of this supplement.)

1.8.2 License Conditions

In our review of outstanding issues identified in the SER and in Supplements No. 1 through 5, we have resolved three of the issues which were identified in Supplement 4 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 Supplements 1, 2, 3, 4 or 5 to the SER in which we discuss the license condition.

(1) Safety/relief valve in-plant testing (SER Supplement 5, Section 3.8.1).

(2) Suppression pool temperature measurement (SER Supplement 5, Section 3.8.1).

(3) Environmental qualification of equipment (SER Supplement 2, Section 3.11; SER Supplement 5, Section 3.11).

(4) Control room habitability (SER Supplement 5, Section 6.4.1).

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

(6) Modifications for fire protection (SER Supplement 2, Section 9.5.1; SER Supplement 5, Section 9.5.1).

(7) Emergency diesel lubricating oil surveillance program (SER Supplement 5, Section 9.5.7).

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

(9) Liquid radwaste treatment system (SER Supplement 5, Section 11.2.1

(10) Retention of persons with BWR operating experience on shift until 100 percent power is achieved (SER Section 13.1; SER Supplement 1, Section 18; SER Supplement 5, Section 13.1).

(11) Inservice inspection program.

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

(13) Initial test program (SER Section 14).

(14) Final procedure for post-accident sampling (SER Supplements 2 and 5, Section 22, Item II.B.3).

(15) Emergency planning.

(16) Emergency response capability.

(17) Actions based on the generic implications of the Salem ATWS events.

1.9 Nuclear Waste Policy Act of 1982

Section 302(b) of the Nuclear Waste Policy Act of 1982 states that the NRC shall not issue or renew a license for a nuclear power plant unless the utility receiving the license has signed a contract with the Department of Energy for disposal services. The Detroit Edison Company has signed a contractual agreement with the Department of Energy dated June 27, 1983.

2 SITE CHARACTERISTICS

2.4 Hydrology and Hydrologic Engineering

2.4.2 Floods and Flood Protection

2.4.2.5 Flooding Protection Considerations

A. Basis For Reevaluation

In the SER we issued in July 1981, we provided our evaluation of the protection provided for the Fermi-2 safety-related structures including the reactor building, the auxiliary building and the residual heat removal (RHR) complex, from the effects of the probable maximum surge (PMS) plus wind-generated waves. In Section 2.4.2.3 of the SER, we found that the maximum calculated stillwater level resulting from the PMS was 586.9 feet (New York Mean Tide Datum, 1935) which is 3.9 feet above the plant grade of 583.0 feet. This latter elevation is also the top elevation of the breakwater along the eastern boundary of the Fermi-2 site (i.e., the western shoreline of Lake Erie). This breakwater (also identified as the shore barrier) was proposed by the applicant to attenuate the effects of wind-generated waves occurring in conjunction with the PMS. The shore barrier is a rubble mound structure using an armor cover of stone. Based on our review, we found that the proposed design and design criteria of the shore barrier was conservative and acceptable to us. The applicant's proposal for periodic inspections and surveys were incorporated into the proposed Fermi-2 Technical Specifications.

The applicant has previously committed to conduct an initial survey of the shore barrier prior to issuance of an operating license to assure that the barrier was built as designed. The applicant completed this survey but did not indicate that there was any deviation of the "as-built" barrier from the design we had approved. However, we determined in a visual inspection of this structure in mid-1983 that the surface configuration of the barrier was apparently outside the design tolerances for this structure. (Refer to Open Item 81-10-01 of Inspection Report No. 54-341/83-10.) The applicant issued Deviation Disposition Report (DDR) No. C-12154 in August 1983 to address these apparent discrepancies. This DDR was subsequently dispositioned by the applicant as "use as is." The applicant attached survey data taken during construction to its disposition and also provided a report from its consultant, R. Noble, to justify its decision.

In July 1984, the Duke Power Company issued its report containing its assessment of the construction of certain portions of the Fermi-2 facility. Duke Power recommended in its report that an engineering evaluation of the shore barrier be performed to determine the significance of the cited deviations with respect to the design function of this structure.

We also conducted additional inspections of the shore barrier in June and July of 1984. (Refer to Inspection Report No. 50-341/84-30.) In these inspections, we concluded that the applicant's disposition cited above, was an item of noncompliance in that it constituted a violation of Criterion XVI of Appendix B to 10 CFR Part 50. In response to this inspection report, the applicant stated its position in its letter dated September 10, 1984, that the as-built shore barrier was acceptable since it would meet its design function of protecting the site fill.

We found this response unacceptable since it did not address the root cause of the deviations of the as-built structure from its original design. Moreover, the applicant did not evaluate whether there were any potentially adverse consequences on the Fermi-2 safety-related structures which could arise from these deviations. The applicant later submitted a report dated November 7, 1984 from another of its consultants, Sargent & Lundy, addressing our concerns on this matter.

We again inspected the shore barrier in late November 1984 (refer to Inspection Report No. 50-341/84-64) and made the following findings:

(a) The as-built structure deviates significantly from the original design both in internal construction of the rock layers and in its external profile.

(b) While QC inspectors reported deficiencies in the construction of the shore barrier, no corrective action was taken. In addition, there is some question regarding the qualifications of these QC inspectors.

(c) Soft clay, defined as unsuitable in the construction specifications, may be present in the shore barrier foundation.

Based on the preceeding considerations, we concluded that we should reevaluate our prior findings on the acceptability of the Fermi-2 shore barrier.

To resolve this issue, we met with the applicant and its consultants (Dames and Moore; Sargent & Lundy; and Ron Noble and Associates) on December 13, 1984. At this meeting, the apparent deficiencies in the as-built shore barrier were discussed. The applicant agreed at this meeting to the following course of action:

(1) Establish additional monitoring stations to better define the as-built shore barrier profile.

(2) Submit the locations and elevations of the new and existing monitoring stations.

(3) Determine the alignment and top elevation of the sheet pile wall at the toe of the shore barrier, if possible.

(4) Submit a filter analysis of the underlying native clay and the first filter layer of the barrier.

B. Design Function of the Shore Barrier

The design function of the shore barrier is to protect the plant from the effects of a PMS on Lake Erie. We do not require that the structure be designed to withstand a simultaneous safe shutdown earthquake (SSE) and a PMS plus wind-generated waves since it is highly improbable that a PMS and an SSE will occur simultaneously or even occur close together in time. However, we do require that the shore barrier be able to perform its design function during a standard project surge on Lake Erie coincident with an operating basis earthquake (OBE). (Refer to Section 2.5.5 of this supplement for our evaluation of the seismic stability of the shore barrier.)

C. Review of the Significant Deficiencies

We reviewed the additional information submitted by the applicant in its letter dated January 16, 1985, and have also reviewed: (1) the revisions to the construction specifications; (2) the recorded deviations from the shore barrier specifications; (3) construction photographs; (4) the as-built survey of the shore barrier; and (5) our findings on our field inspection. Based on this review, we find that there are four potentially significant deficiencies in the construction of the Fermi-2 shore barrier. We discuss our conclusions on each of these four deficiencies below.

(1) At the shore barrier's southerly end from station N6792 to about N6810 and east of about E5665 (i.e., the bottom of the 1 on 2 slope to the sheet pile wall at the toe), the capstone was placed directly on the crushed stone layer. The "A" and "B" stone layers were not placed under the capstone as required by the design drawings.

We find that this portion of the barrier is a transition zone at the end of the structure and is not critical for the protection of the plant fill and for limiting wave heights at safety-related structures during a PMS. Additionally, there are two jetties of land which extend into the lake at this location that will cause larger waves to break lakeward of the shore barriers during major storms thereby reducing wave forces on the shore barrier at this location. Though this portion of the barrier would likely degrade to some extent during a major storm, this degradation would probably be in the form of minor settlements due to erosion of the underlying material. However, we find that this type of structure can readjust (i.e., settle) without resulting in a major structural failure.

(2) At several locations, some of the layer thicknesses are less than the design value. While the stone layers are not entirely missing except as noted in Item (1) above, the "B" stone layer is missing in a portion of the shore barrier at Station N7800.

Although this situation is probably a result of poor construction practices, it is our conclusion that the integrity of the fill material behind the structure will not be adversely affected nor will this deficiency prevent the shore barrier from performing its design function. The "B" stone layer which is missing is in a transition zone at the northern end of the shore barrier; this part of the shore barrier may partially fail during a major storm. A partial failure would probably occur due to a loss of the underlying material and the subsequent settlement and readjustment of the rock layers above. We conclude that a partial failure of this sort at the northern end should have no significant impact on the main structure of the shore barrier.

(3) By visual observation, the quality of some of the capstone rocks appears to be quite poor. Some of the stones seem to be composed of weakly cemented material which is already showing signs of fracturing due to weathering. This poor quality stone appears to be placed randomly throughout the shore barrier and is not concentrated in any one location. About five to ten percent of the capstone rocks fall into this category.

The poor quality rock cited above will eventually succumb to weathering and will have to be replaced. However, we conclude that this is a maintenance problem and should not affect the integrity of the structure since the shore barrier will be properly maintained in accordance with the applicable requirements of the Fermi-2 Technical Specifications.

(4) The profile (i.e., the cross-section of the shore barrier) deviates from the design configuration.

The shore barrier design profile requires a 35-foot crest width at elevation 583.0, then a 1 on 2 slope from 583.0 to 570.0, then a 20-foot section at elevation 572.0. Figures 2-1 and 2.-2 show examples of the as-built profiles from the December 1984 survey. Figure 2-1 shows the best as-built section while Figure 2-2 shows the most deviation of the as-built structure from the design profile. We note that for this type of structure and considering the rock sizes used, it is difficult to construct the entire length of the structure to close tolerances. We further note that a tolerance of 6 to 12 inches is a normal deviation for a structure of this type. The Fermi-2 construction specifications required a tolerance of ± 6 inches. The applicant's position on this matter is that is was difficult to change from a one-capstone thickness between stations E5925 and E5960 to a two-capstone thickness between stations E5960 and E5980. While we cannot conclude whether the observed deviations are due to poor construction or minor settlement, our review of the construction photographs and the as-built surveys indicate that the shore barrier was probably constructed in its present configuration and that the observed deviations did not occur after construction. We conclude that the as-built shape of the shore barrier is satisfactory and will not impair its intended function. (Refer to Section 2.5.5 of this supplement for our evaluation of the filtering capability of the shore barrier.)

D. Evaluation of the As-Built Shore Barrier

The shore barrier was originally intended to protect an on-site cooling pond which was to be the ultimate heat sink. This pond was to be located just west of the existing Lake Erie shoreline. The ultimate heat sink is now contained within the reactor heat removal (RHR) complex which is about 1000 feet from the shoreline. The auxiliary building is about 600 feet from the shoreline and is the nearest safety-related structure. As noted above, the maximum depth of water on the plant grade during a PMS is 3.9 feet. This depth of water can support a 3.0 foot reformed wave which is the the design wave for the Fermi-2 safety-related structures. A major storm on the lake would have to erode above 500 feet of plant fill before waves greater than the design basis wave height could be generated. The historical evidence for the Great Lakes region indicates that an erosion of this magnitude does not occur on unprotected shores. Accordingly, we conclude that the shore barrier at the Fermi-2 site could sustain serious degradation without creating a safety hazard for the safety-related structures. We conclude that the shore barrier is still necessary to provide long-term control of waves and erosional forces. In addition, we find that we can accept the cited deficiencies in the as-built shore barrier since we conclude that the safety margins for the safety-related structures are unchanged (i.e., the maximum reformed wave height on the plant grade is unchanged).

E. Summary of Conclusions

We have evaluated the as-built condition of the shore barrier and conclude that the observed deficiencies in the thicknesses of the layers and the poor quality control on rock gradation would probably lead to some loss of the underlying foundation material during severe storms and that this could result in some settlement of the overlying rock layers. However, we conclude that a settlement of this nature would not preclude the shore barrier from performing its design function. Moreover, the inspection and maintenance program required by the applicable portion of the Fermi-2 Technical Specifications will provide reasonable assurance that the shore barrier will not be allowed to deteriorate significantly from its present as-built configuration. On this basis, we find that the as-built Fermi-2 shore barrier meets the requirements of General Design Criterion 2 of Appendix A to 10 CFR Part 50 and is, therefore, acceptable.

2.5 Geology and Seismology

2.5.5 Slope Stability

2.5.5.1 Basis For Reevaluation of the Shore Barrier

In the SER we issued in July 1981, we stated that we had insufficient information to evaluate the stability of the shore barrier (i.e., the breakwater) under static and seismic loading conditions. Accordingly, we requested additional information on this matter from the applicant and indicated we would provide our evaluation of the stability of the shore barrier in a future supplement to the SER.

Subsequently, the applicant submitted additional information in its letter dated July 13, 1981, regarding its analysis of the slope stability of the Fermi-2 shore barrier. In Supplement No. 1 to the SER, we provided a brief description of this shore barrier and also provided our evaluation of the applicant's proposed design. Based on our review of this information from the applicant, we concluded that the Fermi-2 shore barrier met the requirements of 10 CFR Part 100 for a seismically qualified structure. On this basis, we found the shore barrier to be acceptable.

However, based on new information regarding the design and construction of the Fermi-2 shore barrier, we have reevaluated our prior conclusions and findings with regard to this structure. (Refer to Section 2.4.2.5 of this supplement.) Our concern in this matter is whether the shore barrier can continue to fulfill its intended purpose as stated in Section 2.5.5 of Supplement No. 1. Namely, the shore barrier will: (1) preserve the foundation integrity of the plant site fill, placed to an elevation of 583 feet; and (2) protect the safety-related structures and components of the Fermi-2 facility against waves which would occur up to the probable maximum surge (PMS) condition.

As discussed in Section 2.4.2.5 of this supplement, soft and unsuitable soils may exist in the shore barrier foundation and there may have been deficiencies in the construction of the Fermi-2 shore barrier. Following a meeting on December 13, 1984, between the applicant and its consultants and members of the NRC staff on this matter, the applicant submitted a letter dated January 16, 1985, in which it provided additional information on the shore barrier. Based on this new information, we conclude that the profile of the shore barrier deviates from that of the original design. Additionally, one of the applicant's consultants on this matter, Sargent & Lundy, identified numerous changes to material properties and construction procedures for the shore barrier as listed below:

(a) Soft clay, defined as unsuitable by the original construction specifications, may exist under the clay fill seal.

(b) The foundation preparation requirements were relaxed in areas where excavations of unsuitable material were required.

(c) Changes in compaction were made for the crushed stone bedding layer where the compactive effort was decreased from two passes with a vibratory compactor to two passes with an unspecified bulldozer.

(d) In some parts of the shore barrier, the thickness of the "B" stone and crushed stone layer were significantly thinner than called for by the construction specifications.

(e) Several zones of infilling of finer materials were observed in the capstone layer.

2.5.5.2 Reevaluation of the Shore Barrier

Sargent & Lundy had expressed its concern about the adequacy of the shore barrier foundation and questioned the reliability of the field testing, the in-situ plate bearing test and the unconfined compressive strength tests. Nevertheless, Sargent & Lundy concluded that the shore barrier foundation was adequate stating that "the results of the testing were sufficiently large to eliminate the possibility of the results changing [due to] the methods of construction." (Sargent & Lundy assumed other changes in the construction of the shore barrier were evaluated and approved by the designer.) Accordingly, Sargent & Lundy concluded that the shore barrier is "geotechnically stable." It also noted that some localized damage to the Fermi-2 shore barrier and localized erosion of the soil fill may occur in those areas where the stone layer thickness were appreciably less than the original design thicknesses. However, Sargent & Lundy did not discuss whether those changes from the original design of the shore barrier cited above, would adversely affect the performance of the shore barrier. Specifically, Sargent & Lundy provided no conclusion as to whether the current shore barrier profile was deformed due to the presence of unsuitable soils and the compaction effort during construction which was less than specified or whether the shore barrier was built to its present profile. Neither the applicant nor Sargent & Lundy offered any conclusion regarding the stability of the shore barrier under seismic loading.

Although the applicant did not provide specific information addressing the concerns cited above, we can make the following observations regarding the Fermi-2 shore barrier as built:

(1) The latest survey data submitted by the applicant in its letter of January 16, 1985, indicates that the present shore barrier slope is flatter than the original design. This would normally indicate that the "as built" shore barrier has a larger margin of safety against sliding than that of the original design.

(2) The settlement monitoring program, consisting of 12 monitoring points for the past four years, does not indicate any significant lateral or vertical movements of the shore barrier

(3) The sheet pile wall at the toe of the shore barrier, which would increase the stability of the barrier, was not included in the applicant's original stability analysis.

Based on the considerations cited above, we conclude that the Fermi-2 shore barrier is statically stable. However, the applicant has not demonstrated that the shore barrier is seismically stable. Moreover, the margin of safety of the shore barrier against sliding under seismic loading was relatively low and the changes to the design of the shore barrier are rather significant. (Refer to Section 2.5.5 of Supplement No. 1 in which we noted a maximum factor of safety against a slip-circle failure of 1.25 for a 0.15g SSE.)

However, we do not require that the applicant demonstrate stability for the shore barrier under seismic loading. The basis for this position is that safety-related structures and conduits will not be threatened by waves during a seismic event since the wide buffer zone of plant fill between the shore barrier and the safety-related structures and conduits would not be eroded at a rate sufficient to threaten them before the Fermi-2 shore barrier could be restored.

To address this matter, the applicant has proposed a revision in the Fermi-2 Technical Specifications regarding the shore barrier. We find that these revised Technical Specifications provide reasonable assurance that the protective function of the shore barrier will be properly monitored and adequately maintained.

2.5.5.3 Filter Analysis

Based on our review of the applicant's letter dated January 16, 1985, we find that the applicant's filter analysis is not acceptable since it did not use: (1) the gradation of the stone fills as built; and (2) the proper gradation curves for the clay seal placed under the shore barrier.

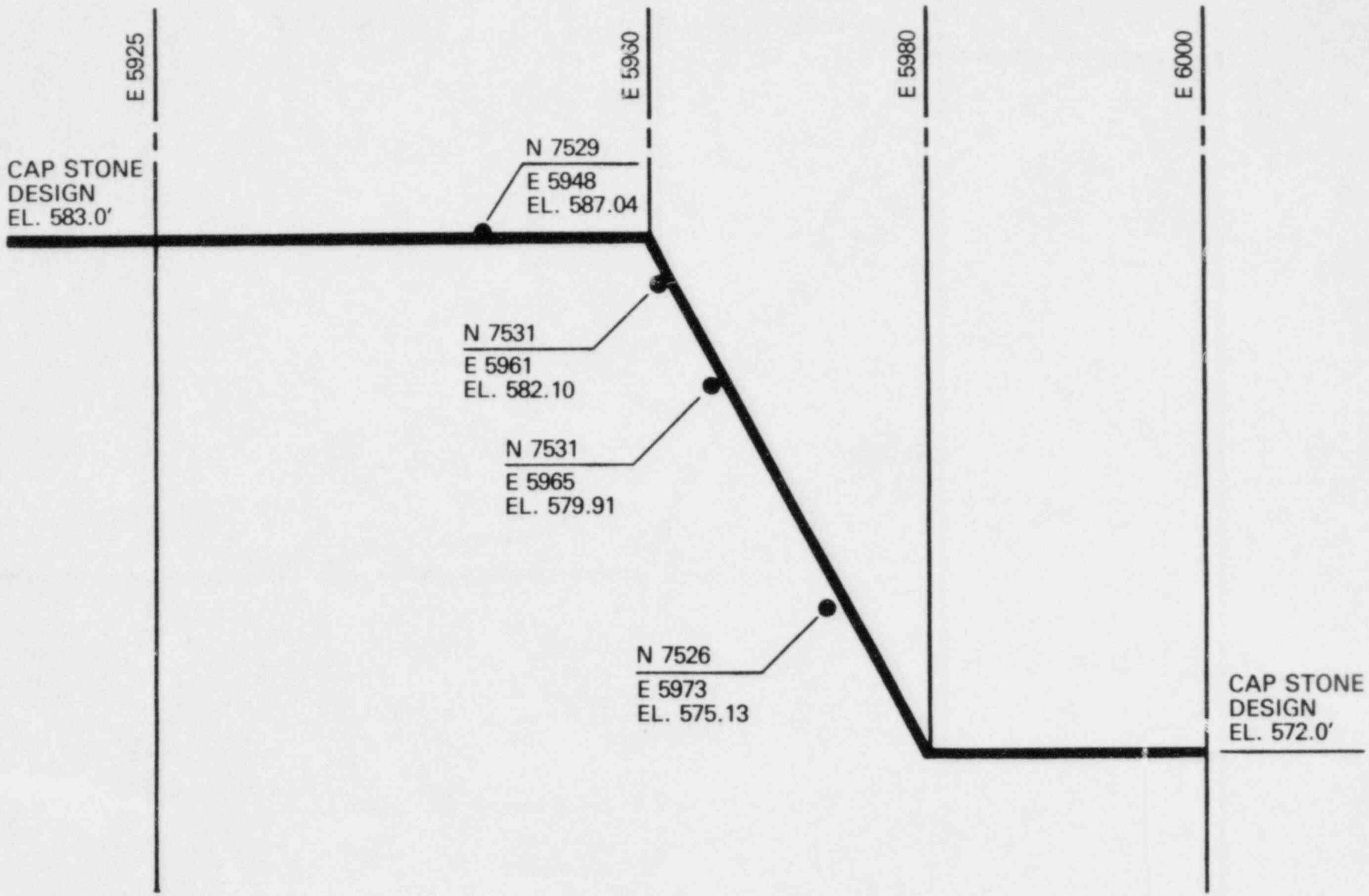
Furthermore, based on this additional information, we conclude that scouring and localized erosion of the Fermi-2 shore barrier could take place under severe storm conditions. However, this scouring and erosion will be localized and limited to the proximity of the barrier. We conclude that these effects should be easily detected by the revised monitoring program in the Technical Specifications.

2.5.5.4 Conclusion

Based on our review of the additional information provided by the applicant, we conclude that the shore barrier is stable under static condition but with a potential for some localized erosion to take place under severe storm conditions. We also conclude that the applicant has not demonstrated the seismic stability of the shore barrier. However, we find that the seismic stability of the shore barrier is not essential since safety-related structures and conduits will not be threatened before repairs to the shore barrier could be made. In order to assure the protective function of the shore barrier, a revised monitoring and maintenance program has been proposed by the applicant which requires that the shore barrier be inspected on an annual basis and after major storms and seismic events. Since the revised Technical Specifications require the applicant to promptly restore the shore barrier to its present configuration in the event of any significant damage, we conclude that the Fermi-2 shore barrier will fulfill its intended protective function. On this basis, we find that the "as built" shore barrier is acceptable.

MONITORING POINT PROFILE FOR THE FERMI-2 SHORE BARRIER

GRID LINE N 7500 (±30.0')



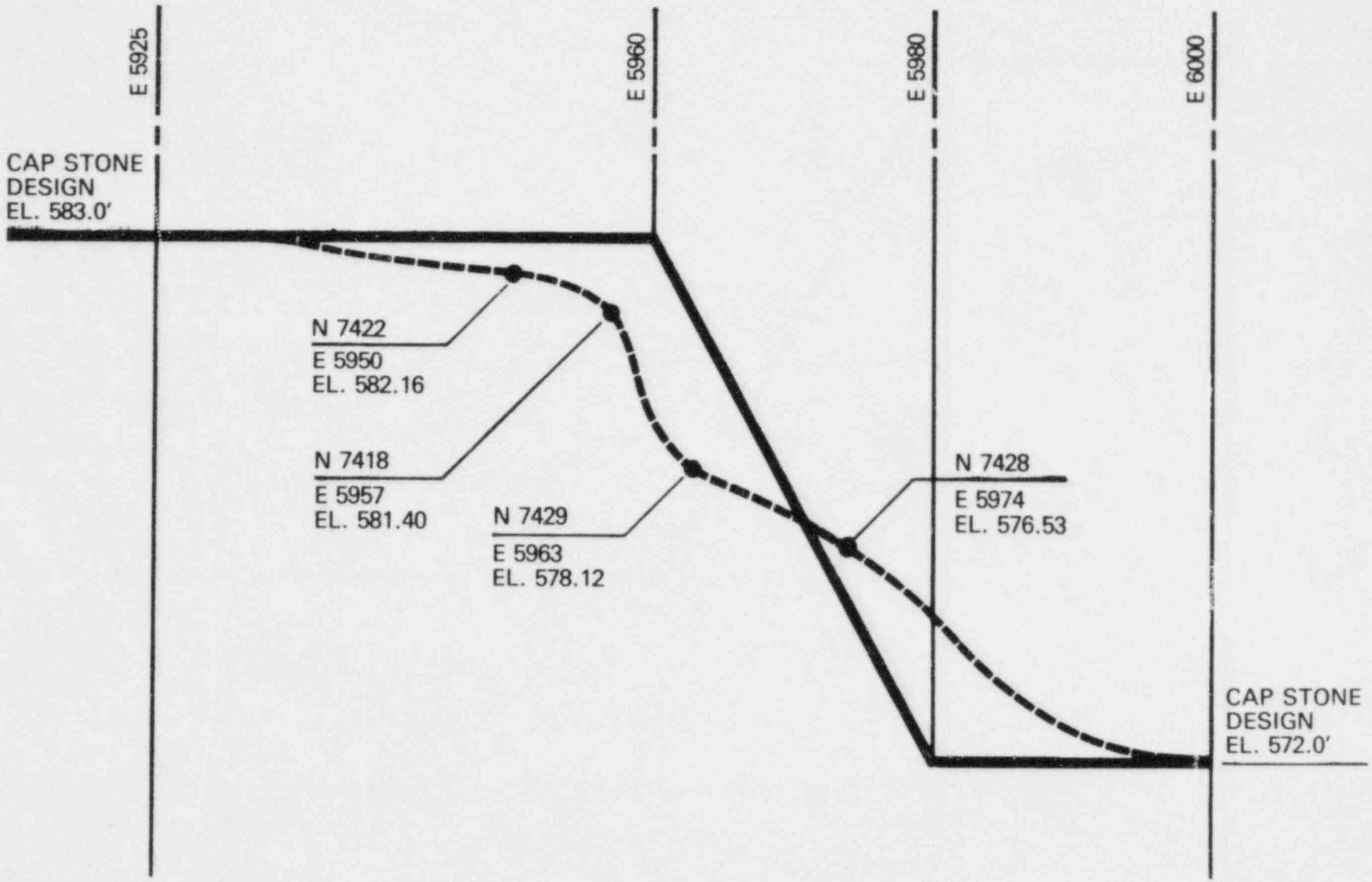
NOTE:
● = MONITORING POINT

Figure 2-1

12/21/84

MONITORING POINT PROFILE FOR THE FERMI-2 SHORE BARRIER

GRID LINE N 7400 (±30.0′)



NOTE:
● = MONITORING POINT

Figure 2-2

12/21/84

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS AND COMPONENTS

3.8 Design of Seismic Category I Structures

3.8.1 Steel Containment

3.8.1.1 Torus and Related Structures

In Supplement No. 3 to the SER, we stated that we and our consultants would review the applicant's analysis of the torus attached piping (TAP) portion of the Fermi-2 Plant Unique Analysis Report (PUAR) when it was submitted. The applicant subsequentally submitted its TAP analysis with its letter dated June 10, 1983. This section of Supplement No. 5 provides our evaluation of the applicant's TAP analysis and the open issues identified in Supplement No. 3 related to the major Fermi-2 containment modifications which the applicant instituted following the criteria established in the BWR Mark I containment Long-Term Program. (Refer to Supplement No. 3 for a discussion of this program.) The following sections follow the format of Supplement No. 3.

<u>Evaluation of Load Audit Review</u>

In this section of Supplement No. 3, we evaluated eight items in Table 1 of Appendix I to that supplement. We found the proposed resolutions of all but Item 8 to be acceptable. Subsequently, we have identified an additional concern with respect to Item 4 of Table 1 in Appendix I. This section of Supplement No. 5 contains our favorable evaluation of Items 4 and 8.

- <u>Item 4</u> (Exceedance Probability For Multiple Downcomer Chugs)

This item is discussed in the following section of this supplement titled "Evaluation of Additional Issues Applicable to Fermi-2." We find that this matter remains resolved.

- <u>Item 8</u> (Local Suppression Pool Temperature Model)

In Supplement No. 3 to the SER, we identified an open issue regarding the applicant's "local pool temperature model" which will be used to estimate the local suppression pool temperature during SRV transients. We found this to be an exception to our acceptance criteria in NUREG-0661 for which the applicant had not submitted sufficient justification.

Subsequently, the applicant has submitted the results of certain plant-unique analyses used to obtain the suppression pool temperature responses to transients involving SRV actuations as required by our acceptance criteria. The results from these analyses indicate that the Fermi-2 facility would be able to operate within the temperature limits specified in NUREG-0783. The applicant's analyses were developed by using a comprehensive computational methodology developed by GE. A key element of this overall methodology is a computer code known as TPOOL which computes the local suppression pool temperatures as a function of the performances of the nuclear steam supply system, the SRV and the reactor heat removal system. A description of TPOOL and the procedures used in its development and qualification was presented to us in a series of meetings, the last of which was held on August 25, 1983. Based on the information presented at these meetings, we and our consultants conclude that the total methodology which includes TPOOL, provides a conservative approach to demonstrate compliance with our requirement in NUREG-0783 for computing the suppression pool temperature transients.

Based on our evaluation of the applicant's analyses, we conclude that the assumptions used by the applicant in developing its local pool temperature model, are reasonably conservative and in agreement with our recommendations set forth in NUREG-0783 and, therefore, are acceptable. On this basis, we find this matter resolved.

Evaluation of Additional Issues Applicable to Fermi-2

(1) Potential Vacuum Breaker Failures From Chugging and Condensation Oscillation Loads

In Supplement No. 3 to the SER, we stated that we would require resolution of the issue of potential vacuum breaker failures due to chugging and condensation oscillation loads during a blowdown to the torus in the event of a loss-of-coolant accident (LOCA), prior to fuel loading. Subsequently, we issued Generic Letter 83-08, "Modification of Vacuum Breakers on Mark I Containments," dated February 2, 1983, in which we required all Mark I owners to submit the results of their plant unique analyses which either formed the bases for modifications to the vacuum breakers or to provide the justification for their as-built acceptability.

We have now completed our review of the generic models proposed by the BWR Mark I Owners Group which were contained in two reports prepared by Continuum Dynamics Inc. (CDI) for the General Electric Company (GE) and the Mark I Owners Group. In our letter to GE, dated December 24, 1984, we stated our conclusion that the proposed dynamic model conservatively predicts the opening and closing velocities for the vacuum breakers and, therefore, is acceptable for use in the analyses and/or qualification of Mark I wetwell-to-drywell vacuum breakers. We did, however, make our acceptance contingent on three restrictions. These are: (1) plant unique loads are to be computed using that drywell model which results in the most conservative prediction (there are two drywell models); (2) the results of this analysis should be submitted by each Mark I Owner, including the values of all plant unique parameters used in the analysis; and (3) each Mark I owner should identify any plant-unique deviations from the methodology and/or assumptions which we approved in our generic review. We will require the applicant to submit, prior to issuance of the full power license, its analyses using the approved generic model subject to the restrictions cited above.

In the course of our generic review of this matter, CDI stated that the damage sustained by the vacuum breaker in the full-scale test facility (FSTF) during Test M1 would not be expected to occur in a domestic Mark I containment since the vacuum breaker response in the FSTF was not prototypical and is very conservative. This is based on the fact that the ratio of the drywell volume to the total vent area in the FSTF is much smaller than in any domestic Mark I containment. Furthermore, the generic model we have approved indicates that no opening impacts of the vacuum breakers are anticipated in a domestic Mark I facility in the event of a LOCA. Based on these considerations, we find there is reasonable assurance that the vacuum breakers in the Fermi-2 facility will not sustain any damage in the event of a LOCA.

On this basis, we find this matter need not be resolved prior to issuance of the full power operating license. We will report our evaluation of the applicant's plant-unique analyses of the effects of chugging and condensation oscillation loads in a future supplement to the SER.

(2) Low-Low Setpoint Logic

In Supplement No. 3 to the SER, we stated that we required the applicant to submit the applicable electrical instrumentation and control systems drawings containing the low-low setpoint logic proposed by the applicant. The applicant has submitted the required additional information on this matter which we have reviewed and found acceptable. (Refer to Section 7.3.2 of this supplement.) On this basis, we find this matter is now resolved.

(3) Determination of Local-To-Bulk Suppression Pool Temperature

This issue has been found resolved in our preceding discussion in Item 8 (Local Suppression Pool Temperature Model).

(4) Cyclic Fatigue Analysis

In Supplement 3 to the SER, we stated that we are in general agreement with the generic approach to the cyclic fatigue analysis proposed by GE and the BWR Mark I Owners Group. However, this approval was subject to our subsequent approval of a generic report to be submitted by GE. On November 30, 1982, GE submitted a report developed by MPR Associates (MPR) on behalf of the BWR Mark I Owners Group. This report presents: (1) a fatigue evaluation method which can be used to analyze the ASME Class 2/3 piping design; (2) the rationale of the approach; (3) the results of the analyses for the Mark I piping systems which were analyzed; and (4) an assessment of the usage factor for the Mark I operating plants.

The method developed by MPR for the evaluation of ASME Class 2/3 piping systems follows the ASME rules for Class 2/3 piping design but augments it to include both mechanical and thermal cyclic stresses in the applicable evaluation equations. It uses the A.R.C. Markl stress range-cycle relationship developed in 1952 which is well known in the industry and was developed using piping fatigue data. Part of the ASME Class 2/3 piping design rule is basically a stepped approximation of the curve. The Markl curve compares conservatively with the ASME Class 1 fatigue design curve below 10,000 cycles. The majority of the loadings for this application are under 10,000 cycles and there are none above 20,000 cycles. A similar approach for low frequency vibrations was adopted by E. C. Rodabaugh and G. T. Yahr.

Using the loadings and loading combinations recommended in NUREG-0661, ASME Class 2 analyses were performed on torus attached piping systems in all BWR

Mark I plants. Limiting piping systems were selected from the results of these analyses. MPR then determined the loading cycles and loading cycle combinations during the life of a typical Mark I plant for these limiting piping systems. As a conservative assumption, an absolute summation was used to combine all dynamic responses. Appropriate loading cycle combinations were applied to representative limiting piping systems to determine the usage factors. A total of 11 SRV discharge piping systems and 25 other torus attached piping systems, were analyzed. The results of this analysis are:

<u>SRV Discharge Piping</u>

Less than 0.3 fatigue usage	73 percent
Less than 0.5 fatigue usage	100 percent

<u>Other Torus Attached Piping</u>

Less than 0.3 fatigue usage	92 percent
Less than 0.5 fatigue usage	100 percent

The fatigue usages are less than 0.5 for all systems. On this basis, the BWR Mark I Owners Group requested that they not be required to perform plant-unique fatigue analyses for individual plants as originally required in the Mark I criteria.

Based on our review, we find that the proposed Class 2/3 piping fatigue evaluation methodology utilized the Markl fatigue relationship which is more conservative than the ASME Class 1 method below 10,000 cycles. As an additional conservatism, fatigue analyses of limiting ASME Class 2 piping systems were conducted using an absolute summation to combine all dynamic responses. Since most representative BWR Mark I limiting systems have fatigue usages below 0.3 and none has a fatigue usage above 0.5 and since the ASME Section III Code allowable usage factor is 1.0, we find that the proposed generic approach is acceptable and that plant-unique fatigue analyses are not required. On this basis, we find this matter is now resolved.

(5) <u>Single Vent Lateral Chugging Load Magnitude</u>

In Supplement No. 3 to the SER, we stated our finding that when phasing between vents during a pool chug is taken into account, the probability that a group of two vents will exceed the design basis loads specified in the applicant's PUAR once per LOCA will be comparable to our acceptance criteria in NUREG-0661 of 10^{-4} per LOCA. Subsequentally, during the post-implementation phase of another Mark I PUAR, we developed a concern regarding the validity of the Fermi-2 downcomer chugging lateral load definition which we had favorably evaluated in Supplement No. 3 of the SER. The design basis load in question is based on the highest load recorded in the Mark I Full-Scale Test Facility (FSTF) during a series of conservative prototypical blowdowns. Since the total number of chugs in the FSTS data base was significantly less than the number of individual downcomer chugs which could be expected in a postulated LOCA in the Fermi-2 facility, we and our consultant, BNL, felt that using the highest observed FSTS lateral load as a design basis load should be reviewed with regard to its statistical validity. Additionally, we concluded that the relationship between the location of the strain gages which were used for load definition and the location on the downcomer at which the critical stresses occur, needed to be investigated further.

At a meeting on December 17, 1982, the General Electric Co. (GE) and the BWR Mark I Owners presented their analyses to demonstrate that the downcomer chugging lateral load definition which we had previously accepted, was still valid. However, we and our consultant, BNL, did not accept these analyses as sufficient justification to revalidate the design basis lateral chugging load. Accordingly, we indicated to GE and the BWR Mark I Owners that any margin in the current definition of this load due to the measurement location on the downcomer, needed to be quantified and that the conservatism of this load needed to be established on a statistically valid basis.

On March 11, 1983, EDS Nuclear and GE on behalf of the BWR Mark I Owners, presented to us and our consultant, BNL, their assessment of the conservatism in the Mark I downcomer chugging lateral load definition in accordance with our suggestions at the previous meeting on December 17, 1982. (This presentation was later documented in a report by EDS Nuclear and GE.) The presentation showed that the resultant static equivalent load (RSEL) derived from the data obtained by the strain gages located on the FSTF downcomers and used to determine the design lateral chugging load specification, is sufficiently conservative to alleviate our previous concerns about the size of the data base. This data demonstrated that the critical stresses caused by the lateral chugging load occur on the downcomer-vent head intersections. However, the lateral load definition was based on data from bending bridge strain gages located on the downcomer itself, more than two feet away from the intersection, rather than on gages located near the intersection. Since the downcomer itself is much stiffer than the downcomer-vent header intersection, the bending bridge strains are more amplified by the higher-mode dynamic response than the strains measured on the gages near the intersection.

On this basis, EDS and GE demonstrated that an RSEL obtained from the bending bridge data which was used for this lateral chugging load definition, is larger than an RSEL obtained from the intersection strain gage data. Since it is the intersection stresses which are critical, we find that using a load definition based on the the bending bridge RSEL is a conservative approach. The size of this conservatism was estimated in several ways. First, the responses at the two locations on a finite element model of the FSTS downcomer/vent header structure were compared. Secondly, the actual strain gage data at the two locations were examined. The factor of conservatism (i.e., the ratio of RSEL) varied for different chugs; this value averaged above a factor of two. The largest chugging loads had an RSEL ratio of 1.7 to 1.8.

A statistical analysis was then presented by GE to define an acceptable exceedance limit and to estimate the exceedance probability for the chugging lateral downcomer load when the margins discussed above are included. As a design goal, one occurrence per LOCA of a load equal to or above the design load was presented as an acceptable criteria. Based on estimates of a typical number of downcomers, the chugging frequency and its duration, this implies that the corresponding exceedance probability limit on a per downcomer chug basis, should not exceed about 10^{-4}. When a log-normal distribution is fit to the load data and the effects of random magnitude and direction of the load are considered, the exceedance probability of the original design load is found to be 5.8×10^{-4}. However, as discussed above, the design basis lateral chugging load is conservative by a factor of at least 1.7 to 1.8. The conservatism in this load is equivalent to an overestimation of the exceedance probability.

Therefore, the load margin can be converted to a reduced probability of exceedance. If a load is conservative by a factor of 1.3, the exceedance probability per downcomer chug is reduced to 7.6×10^{-5}; i.e., below the required level. A load which is conservative by a factor of 2.0 is equivalent to reducing the exceedance probability to 1.4×10^{-6}; i.e., much below the required bound.

We stated previously in Supplement No. 3 that one exceedance of the design basis lateral chugging load on any downcomer during a postulated LOCA, is an acceptable criterion and in agreement with exceedance limits established for other BWR designs (i.e., the BWR Mark II plants). We find that the 10^{-4} probability limit quoted for this design goal may possibly be non-conservative by about 30 percent. However, the margins of conservatism obtained from the RSEL comparisons were closer to 2.0 than to 1.3. This would more than offset any non-conservatism in the 10^{-4} exceedance probability. While the probability exceedance limit corresponding to one exceedance of the design basis load per LOCA may be as small as 8×10^{-5}, the probability of exceeding the current design basis load is closer to 1×10^{-6} when the demonstrated margins are accounted for. On the basis that EDS and GE confirmed these margins, we find that the BWR Mark I design basis chugging lateral load is adequately conservative for the Fermi-2 facility and is, therefore, acceptable. On this basis, we find this matter is now resolved

3.8.1.2 Torus Attached Piping

In Supplement No. 3 to the SER, we stated that we would provide our evaluation of the applicant's report on its design calculations for supports of piping attached to the torus, in a future supplement to the SER. Subsequently, the applicant provided this additional information in its submittal dated June 10, 1983. This section contains our review and evaluation of the applicant's submittal on the torus attached piping (TAP) portion of the Plant Unique Analysis Report (PUAR).

We have used two contractors to review the Fermi-2 PUAR. Brookhaven National laboratory (BNL) performed the load audit review and Franklin Research Center (FRC) performed the structural audit review. The audit review procedures are described in Supplement No. 3 to the SER.

Torus Attached Piping Analysis - Load Audit

BNL assisted us in evaluating the hydrodynamic loading methodology used by the applicant for the TAP portion of the Fermi-2 PUAR with respect to the acceptance criteria in NUREG-0661. During this review, two issues were identified as exceptions to these acceptance criteria. Based on the applicant's submittal of additional information, both these issues have now been resolved. However, resolution of the issue pertaining to the reduction factor which was used by the applicant to estimate the safety relief valve (SRV) water jet impingement load and the air bubble drag load, is subject to confirmation in the Fermi-2 SRV in-plant tests. We will condition the Fermi-2 operating license to require these confirmatory SRV in-plant tests to be performed prior to operation at full rated power. We will report our evaluation of these in-plant tests in a future supplement to the SER. Appendix M of this supplement contains our detailed evaluation of the applicant's hydrodynamic loading methodology.

Torus Attached Piping Analysis - Structural Audit

FRC performed an audit of the Fermi-2 PUAR with respect to the structural analysis of the torus attached piping, the suppression chamber penetrations, and the associated equipment and components. Based on this audit, we conclude that the applicant has satisfied our requirements in NUREG-0661. However, this approval is also predicated on the confirmation of the conservatism in the assumed load reduction used for the SRV water jet impingement load and the air bubble drag load as discussed in the section above regarding the load audit.

Conclusions

Based on our review and that of our consultants, BNL and FRC, of the the applicant's TAP portion of its PUAR, we have reached the following conclusions:

(1) The applicant's TAP submittal is acceptable and satisfies our requirements in NUREG-0661.

(2) The applicant's proposed deviations to the acceptance criteria contained in NUREG-0661, are acceptable. However, this approval is based on the confirmation by in-plant tests of the conservatism in the load reduction factor used for the SRV water jet impingment load and the air bubble drag load and our approval of the methodology used to derive this reduction factor.

(3) We will condition the license to require the applicant to submit the results of these Fermi-2 in-plant tests and their analysis.

3.8.1.3 Additional Mark I Issues

Recent Mark III Containment Design Concerns

In Supplement No. 3 to the SER, we stated our preliminary evaluation that no significant design deficiencies has been identified in the BWR Mark I pressure suppression containments as a result of our review of the concerns of Mr. John Humphrey. We also stated that we did not believe there was any erosion of existing safety margins in the Mark I designs. We did state in Supplement No. 3 that we would require resolution of any issues related to the Fermi-2 facility prior to fuel loading, which might arise from our continuing review of the Humphrey concerns.

Our position on this matter remains unchanged; i.e., the design differences between the Fermi-2 Mark I containment and the Mark III containment make many of the issues raised by Mr. Humphrey moot. Furthermore, the applicant stated in its letter dated October 26, 1982, that it has reviewed the General Electric (GE) letter on this matter, "Mark I Containment Program, Humphrey Containment Concerns", MFN 138-82, dated September 24, 1982, and has determined that the responses to these issues provided in GE's letter, are applicable to the Fermi-2 facility.

On the basis of our preliminary assessment of the 23 major items (each with related subtopics) identified by Mr. Humphrey, we find that all of these issues either were previously considered in some way by the applicant or do not represent major safety concerns. However, we will continue our review of this matter to clarify and confirm a few points relative to the Humphrey concerns. It is our position that in light of the applicant's response and our preliminary evaluation, the Humphrey concerns need not delay the issuance of a full power license for the Fermi-2 facility. If necessary, we will report any safety-related issue which may arise as a result of this continuing review. The basis for our position on this matter is:

(1) Based on our review of the Humphrey issues, we conclude that they were, for the most part, considered by the applicant in the design of the Fermi-2 containment. Furthermore, we have not, to date, uncovered any deficiency in the Fermi-2 containment design.

(2) The design differences between Fermi-2's Mark I containment and the Mark III containment make many of the issues not pertinent to Mark I containments.

(3) Our preliminary evaluation of the applicant's responses in the letter cited above, confirms the applicant's statement that it has not identified any design deficiency arising from the Humphrey concerns.

While this matter is not closed, we find that resolution is not required prior to issuance of the Fermi-2 full-power operating license.

Wetwell-to-Drywell Vacuum-Breaker Performance During Pool Swell

In Supplement No. 3 to the SER, we stated that our concern regarding suppression pool bypass during pool swell is not applicable to BWR Mark I containments. We did state, however, that we were still reviewing the effect of chugging and condensation oscillation loads on the performance of these vacuum breakers. We have partially completed our review of this latter phenomenon and found that it need not be resolved prior to issuance of the Fermi-2 full power operating license. (Refer to Section 3.8.1.1 of this supplement.)

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

In Supplement No. 4 to the SER, we stated that the applicant's seismic and dynamic qualification of safety-related equipment has been satisfactorily implemented with the exception that two confirmatory items needed further action by the applicant. These are: (1) that the values of acceleration used in qualifying valves are consistent with those calculated in the "as-built" piping analyses (Item (2)(c) in Supplement No. 4); and (2) that the applicant submit the remaining seismic qualification review team (SQRT) forms for equipment qualified after our SQRT audit. (Refer to Item (2)(d) in Supplement No. 4 to the SER.) This SQRT audit was conducted on July 27 through July 31, 1981, and is discussed in Section 3.10 of Supplement No. 1 to the SER. Subsequently, the applicant provided additional information on these two items in its submittals of June 22 and November 20, 1984.

We have reviewed this additional information and found that the results of the applicant's review of valve accelerations can be placed in one of the following three categories:

(a) The capability of the valve to withstand the accelerations calculated in the as-built piping analysis is less than the generic values of acceleration contained in the purchase specification.

(b) The calculated acceleration of the valve is greater than the generic values in the purchase specifications but an analysis by the applicant verified that the valve's capability to withstand acceleration was greater than, or equal to, the accelerations calculated in the as-built piping analysis.

(c) A modification was initiated for the one valve (V11-2006) whose acceleration calculated in the as-built piping analysis exceeds the valve's capability to withstand that acceleration. This modification has been completed and the as-built acceleration capability has been reduced to a value less than the valve's acceleration capability.

We have also reviewed the additional SQRT forms submitted in the applicant's letter dated November 20, 1984, for equipment installed and qualified after our SQRT audit. The applicant confirmed that all safety-related equipment is seismically qualified.

On the basis of our findings in the SQRT audit as well as our review of the subsequent submittals by the applicant, we conclude that an appropriate seismic and dynamic qualification program has been defined and implemented. We also conclude that this program provides reasonable assurance that such equipment will function properly during and after excitation under the combined effects of the seismic load and the hydrodynamic loads associated with the suppression pool. Accordingly, we find the Fermi-2 seismic and dynamic qualification program to be acceptable since it meets the applicable requirements of General Design Criteria 1, 2, 4, 14 and 30 of Appendix A to 10 CFR Part 50, Appendix B to 10 CFR Part 50, and Appendix A to 10 CFR Part 100.

3.11 Environmental Qualification of Electrical Equipment Important to Safety and Safety-Related Mechanical Equipment

3.11.1 Introduction

In the SER issued in July 1981, we stated that we would report the results of our review of the applicant's environmental qualification program in a supplement to the SER. In Supplement No. 1 to the SER, we concluded that additional information was required to complete our review and we presented the applicant's schedule for submitting additional information regarding its environmental qualification program. In Supplement No. 2 to the SER, we found on the basis of the additional information submitted by the applicant that its environmental qualification program for safety-related electrical equipment was acceptable subject to conformance by the applicant to certain requirements contained in Sections 3 and 4 of Appendix F to Supplement No. 2. In Supplement No. 3 to the SER, we found that the applicant had addressed some but not all of our requirements in Appendix F to Supplement No. 2. We have now completed our review of all outstanding items related to the applicant's environmental qualification program for safety-related electrical equipment. Our evaluation is contained in the following sections of this supplement to the SER.

3.11.2 Design Criteria

Equipment which is required to perform a safety function in a nuclear power plant must be demonstrated by the applicant to be capable of functional operability during its installed life under all postulated service conditions, including design basis accidents, for the time it is required to operate. This requirement which is embodied in General Design Criteria (GDC) 1 and 4 of Appendix A and Sections III, XI, and XVII of Appendix B to 10 CFR Part 50, is applicable to safety-related equipment located inside as well as outside containment. More detailed requirements and guidance relating to the methods and and procedures for demonstrating this capability for electrical equipment are presented in: (1) Section 50.49 of 10 CFR Part 50; (2) NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," which supplements the Institute of Electrical and Electronics Engineers (IEEE) Standard 323; and (3) various NRC regulatory guide and industry standards.

3.11.3 Development of the NRC Staff's Bases for Evaluation

NUREG-0588 was issued in December 1979 to promote a more orderly and systematic implementation of equipment qualification programs by industry and to provide guidance to the NRC staff for its use in ongoing licensing reviews. The positions contained in this report provide guidance on: (1) how to establish environmental service conditions; (2) how to select methods which are considered appropriate for qualifying equipment in different areas of a nuclear power plant; and (3) other areas such as margins, aging, and documentation. In February 1980, the NRC asked certain near-term applicants for an operating license (NTOL) to review and evaluate the environmental qualification documentation for each item of safety-related electrical equipment and to identify the degree to which their qualification programs were in compliance with the staff positions discussed in NUREG-0588.

IE Bulletin 79-01B, "Environmental Qualification of Class IE Equipment," which was issued by the NRC Office of Inspection and Enforcement (IE) on January 14, 1980, and its supplements dated February 29, September 30, and October 24, 1980. established environmental qualification requirements for operating reactors. This bulletin and its supplements were also provided to NTOL applicants for consideration in their applications.

A final rule on environmental qualification of electrical equipment important to safety for nuclear power plants became effective on February 22, 1983. This rule, Section 50.49 of 10 CFR Part 50, specifies the requirements to be met for demonstrating the environmental qualification of electrical equipment important to safety located in a harsh environment. In conformance with Section 50.49 of 10 CFR Part 50, electrical equipment for the Fermi-2 facility may be qualified according to the criteria specified in Category II of NUREG-0588.

The qualification requirements for mechanical equipment are principally contained in Appendices A and B of 10 CFR Part 50. The qualification methods defined in NUREG-0588 can also be applied to mechanical equipment.

3.11.4 NRC Staff Review and Evaluation of the Fermi-2 Environmental Qualification Program

Our review included onsite examinations of equipment, audits of qualification documentation, and a review of the applicant's submittals for completeness and acceptability of systems and components, qualification methods, and accident environments. The criteria described in: (1) Section 3.11 of the NRC Standard Review Plan (SRP), NUREG-0800, Revision 2; (2) NUREG-0588, Category II; and (3) the requirements in Section 50.49 of 10 CFR Part 50 form the bases for our evaluation. The scope of our review includes an evaluation of the applicant's conformance with the requirements of Section 50.49 of 10 CFR Part 50, the qualification of safety-related mechanical equipment, and the resolution of the open items identified in Supplements Nos. 2 and 3 of the SER.

To document the degree to which its environmental qualification program complies with the NRC environmental qualification requirements and criteria, and in response to a NRC staff request for additional information, the applicant provided equipment qualification information in its letters dated November 1, 1983 and January 16, February 14, March 14, July 16, July 23, and September 7, 1984, to supplement the information in Section 3.11 of the Fermi-2 FSAR.

We also performed a final audit of the applicant's qualification documentation and its installed electrical equipment on July 16, 17, and 18, 1984. This audit consisted of a review of 11 files containing information regarding equipment qualification. Our findings from the audit are discussed in Section 3.11.5.1 of this supplement. The following sections summarize our detailed review.

3.11.4.1 Compliance With Section 50.49 of 10 CFR Part 50

Section 50.49 of 10 CFR Part 50 identifies three categories of electrical equipment which must be qualified in accordance with the provisions of the rule.

(1) safety-related electrical equipment (equipment relied on to remain functional during and following design-basis events)

(2) nonsafety-related electrical equipment whose failure under the postulated environmental conditions could prevent satisfactory accomplishment of the safety functions by the safety-related equipment

(3) certain post-accident monitoring equipment (Regulatory Guide 1.97, Category 1 and 2 post-accident monitoring equipment).

Conformance with the requirements of Section 50.49(b)(1) of 10 CFR Part 50 for safety-related electrical equipment located in a harsh environment has been addressed by the applicant and has previously been reviewed and found acceptable by the staff.

To address its conformance with Section 50.49(b)(2) of 10 CFR Part 50, the applicant referred to its responses to IE Information Notice 79-22, "Qualification of Control Systems," and its conformance with both IEEE Std 279-1971 and IEEE Std 308-1971 to meet the intent of Regulatory Guide 1.75 as reported in Section 7.1.2 of our SER. In addition, we have reviewed the separation criteria and concluded that these criteria are acceptable and are equivalent to those in IEEE Std 384-194 with possible additional measures as reported in Section 8.3.4 of the SER. We have reviewed and evaluated the applicant's conformance with the above documents and found it acceptable as it relates to equipment qualification. Based on the foregoing discussion, we conclude that the applicant's conformance to Section 50.49(b)(2) of 10 CFR Part 50 is acceptable.

Section 50.49(b)(3) of 10 CFR Part 50 requires that all instrumentation located in a harsh environment which is installed in conformance with the guidelines of Regulatory Guide 1.97, Categories 1 and 2, be included in the equipment qualification program unless adequate justification for not doing so is provided. The applicant has indicated that all such equipment is included in its qualification program. Accordingly, we conclude that the applicant has complied with the requirements of Section 50.49(b)(3).

3.11.4.2 Qualification Methods

Detailed procedures for qualifying safety-related electrical equipment in a harsh environment are defined in NUREG-0588. The criteria in NUREG-0588 are also applicable to the other equipment important to safety defined in Section 50.49 of 10 CFR Part 50. We reviewed the methods used by the applicant to demonstrate qualification to assure that they are in compliance with NUREG-0588, Category II.

Although there are no detailed requirements for mechanical equipment, General Design Criteria 1 and 4 of Appendix A and Sections III and XVII of Appendix B to 10 CFR Part 50 contain the following requirements related to equipment qualifications:

(a) Components shall be designed to be compatible with the postulated environmental conditions, including those associated with loss-of-coolant accidents (LOCAs).

(b) Measures shall be established for the selection and review for suitability of application of materials, parts, and equipment which are essential to safety-related functions.

(c) Design control measures shall be established for verifying the adequacy of design.

(d) Equipment qualification records shall be maintained and shall include the results of tests and materials analyses.

The applicant has submitted the results of its safety-related mechanical equipment qualification program and qualification documentation of three items of safety-related mechanical equipment. Our review has verified that the requirements for environmental qualification of safety-related mechanical equipment have been addressed in an acceptable manner.

3.11.4.3 Equipment Not Completely Qualified

For equipment items not having complete qualification documentation, the applicant has provided a commitment for corrective action and a schedule for its completion. For items which will not have full qualification before an operating license is granted, the applicant performed analyses in accordance with paragraph (i) of Section 50.49 of 10 CFR Part 50 to ensure that the plant can be operated safely pending completion of environmental qualification. We have concluded that reasonable assurance has been provided that the Fermi-2 facility can be operated safely pending completion of the applicant's environmental qualification program.

3.11.5 Evaluation of the Equipment Qualification Program

As a result of our review of the applicant's submittal, our audit of the documentation contained in the applicant's qualification files, and previous staff evaluations of equipment in other plants, we conclude that the Fermi-2 environmental qualification program meets the requirements for environmental qualification of equipment and adequately resolves all open items identified in Supplements 1, 2 and 3 of our SER, including the submittal of all information requested in Appendix F of Supplement 2. The following discussion provides some details of our review and assessment.

The applicant's revised environmental qualification submitted in its letter dated July 19, 1983, adequately addresses equipment aging requirements and plant maintenance and surveillance program. In Supplement 2 to the SER, we required that surveillance and maintenance program procedures be implemented before operation at full-power. In its revised program submittal, the applicant states that individual sets of procedures will be complete and the program in place prior to fuel load. Subject to confirmation by Region III, we find that this item is resolved acceptably.

In Supplements 1 and 2 of the SER, we requested additional information for the environmental qualification of equipment required to function in the event of a break in the scram discharge system. Our evaluation of this item is presented in Section 6.3.4 of Supplement No. 4 to the SER. Based on this information, we find that this item is acceptably resolved as it relates to environmental qualification.

3.11.5.1 Environmental Qualification Audit

With assistance from EG&G Idaho, Inc., we conducted an audit of the applicant's qualification files on July 16, 17, and 18, 1984. The purpose of this audit was to verify the bases of the information submitted by the applicant. Eleven equipment qualification files, representing about 10 percent of the equipment items in the equipment qualification program, were selected for our detailed review during the audit.

The equipment items selected for audit were:

1. Pressure Switch, Pressure Controls Inc., A17-1P
2. Temperature Switch, Love 56-838
3. Thermocouple, Thermoelectric Type-K
4. Electric Cable, BIW, Bostrad 7E
5. Pressure Transmitter, Rosemount 1152
6. Thermocouple, Conax 2SK-2909-05
7. Motor, Westinghose 444US-TBDP
8. Radiation Monitor, General Atomics RD-23
9. Terminal Block, Kulka MAI-60
10. Pressure Differential Switch, Barton 288A
11. Electric Cable, Anaconda EP

These files were reviewed to determine if environmental qualification had been demonstrated based on the documents contained in the files. With some except ions, we determined that there was an acceptable demonstration that the applicant had established qualification. We made the following observations during the audit:

(a) In four of the files reviewed, instrument accuracy requirements had not been specified. The applicant responded that it has been involved in this generic issue related to instrument accuracy as a member of the utility group which is addressing this matter. The applicant has also submitted a letter regarding its approach and commitments.

We have previously stated that we consider the Technical Specifications for instrument channel setpoint allowable values issued for GE/BWR plants licensed to operate (and those specifications in preparation for NTOL plants) based on the GE set-point methodology developed to date, to be sufficiently conservative to permit continued licensing of NTOL plants until this issue is resolved. We have also requested that each utility, including the applicant, perform a confirmatory review of their Technical Specifications, predicated on the revised GE methodology. This review is to include a plant-specific assessment of the environmental effects on instrument bias and uncertainty during normal and off-normal conditions for the respective plant designs.

The applicant must confirm that the demonstrated instrument accuracy resulting from a harsh environment envelops the plant requirements specific to Fermi-2. Since this issue is part of an ongoing generic staff review, we consider the applicant's position to be acceptable pending final resolution.

(b) In one file, the applicant had used the first five hours of exposure time at high temperature during a LOCA test as the basis to calculate the qualified life of the equipment. In those SERs issued for operating reactors, we have affirmed a Franklin Research Center conclusion that, "extrapolation of test data from a saturated steam testing using an Arrhenius technique to establish a qualified life is not tecnically justified." Moreover, we find that five hours of thermal aging time is not technically adequate as reflected in the requirement of IEEE Std 323-1974 that a thermal aging time of 100 hours is a minimum acceptable time.

To address our concern on this matter, the applicant stated in its letter dated February 4, 1985, that in place of the Arrhenius extrapolation a material aging evaluation was added to its analysis. This analysis meets the guidance outlined in the aging section of NUREG-0588 for Category II equipment. On this basis, we find this matter resolved.

(c) We observed during the audit that a plant specific analysis to establish the acceptability of equipment performance during one of the qualification tests was not documented in the qualification files. To correct this, the applicant has committed to review all files and revise the files before fuel load to provide this supplemental information as necessary.

To address this specific concern, the applicant stated in its letter dated February 4, 1985, that it had performed a review of all of its qualification central files and that it determined that 28 of the 46 existing central file packages needed to be revised. The applicant further stated in this letter that it had completed these revisions. On this basis, we find that this issue is now resolved.

(d) In order to address radiation dose rate effects and other age-related degradation, the applicant has provided information to demonstrate its commitment to an acceptable surveillance/maintenance program for cables inside containment.

The applicant stated in its letter dated September 7, 1984, that it plans to participate in a generic study to investigate material aging characteristics of safety-related electrical cables located inside containment. On this basis, we find that this matter is resolved.

We find that the resolutions and actions proposed by the applicant during the audit and documented in subsequent letters cited above, are acceptable.

As part of our audit, the equipment as actually installed was inspected during a plant walkdown. The purpose of this walkdown was to verify that the manufacturer, the model number, the location and the installation of equipment was consistent with the qualification documents. We discovered no discrepancies, and we verified that equipment interfaces were adequately addressed in the qualification program. Our audit constitutes the confirmatory visit discussed in Supplement 3 to the SER.

The applicant has identified a limited number of safety-related equipment items which will not be qualified by March 31, 1985, in its letter dated March 7, 1985. Most of these are presently scheduled to be completed by August 1985. We will condition the license to require that these items be qualified no later than November 30, 1985. However, we have recommended to the applicant that this qualification be completed as early as is feasible.

3.11.6 Summary and Conclusions

We have reviewed the Fermi-2 program for the environmental qualification of electrical equipment important to safety and safety-related mechanical equipment. The purpose of our review was to determine the adequancy of the program, including the scope of the qualification program, and the methods used to demonstrate qualification.

We have determined that the following condition should be included in the Fermi-2 operating license:

> No later than November 30, 1985, DECO shall environmentally qualify all electrical equipment according to the provisions of 10 CFR 50.49.

Based on our review, we conclude that the applicant has demonstrated conformance with the requirements for environmental qualification contained in Section 50.49 of 10 CFR Part 50, the relevant parts of General Design Criteria 1 and 4, and Sections II, XI, and XVII of Appendix B to 10 CFR Part 50, and with the criteria specified in NUREG-0588.

4 REACTOR

4.4 Thermal and Hydraulic Design

4.4.1 Evaluation

In the SER we issued in July 1981, we stated that we would impose a condition on the Fermi-2 operating license to require the applicant to submit an additional hydrodynamic stability analysis for the Fermi-2 facility which must be approved prior to operating beyond the first refueling outage.

Subsequently, the applicant has proposed Technical Specifications for the Fermi-2 facility which provide procedures for detecting and suppressing power oscillations that might be induced by a thermal-hydraulic instability. We have reviewed these modified Fermi-2 Technical Specifications and find that they, together with the original Fermi-2 stability analysis, provide reasonable assurance that General Design Criterion 12 of Appendix A to 10 CFR Part 50 is satisfied. Accordingly, we do not require the applicant to submit any additional stability analysis for Cycle 2 and beyond and we will not impose a license condition on this matter. We find that this mattter is now resolved.

5 REACTOR COOLANT PRESSURE BOUNDARY

5.2 Integrity of the Reactor Coolant Pressure Boundary

5.2.1 Compliance with Appendices G and H, 10 CFR Part 50

In the SER we issued in July 1981, we stated that the applicant would submit additional information to justify exemptions to the requirements of Appendices G and H to 10 CFR Part 50. In Supplement No. 1 to the SER, we evaluated the additional information submitted by the applicant on these matters. Section 5.2.1 in Supplement No. 1 replaced and superseded Section 5.2.1 of the SER. In this new section, we concluded that exemptions to Sections I.B, III.A, III.B.1, III.B.3, III.B.4, III.C.1, III.C.2, IV.A.2.c, IV.A.3 and IV.B of Appendix G to 10 CFR Part 50 and Sections II.B and II.C.1 of Appendix H to 10 CFR Part 50 were required and that the applicant had submitted sufficient information to justify these exemptions. Subsequently, the Commission revised Appendices G and H. The revised Appendices G and H were published in the Federal Register on May 27, 1983, and became effective on July 26, 1983. Accordingly, the exemptions to Appendices G and H which we stated were required for the reactor coolant pressure boundary, are no longer required. The basis for our conclusions on this matter is provided in the following two sections.

5.2.1.1 Compliance with Appendix G, 10 CFR Part 50

Previous versions of Appendix G to 10 CFR Part 50 had specific requirements for the preparation and testing of all reactor coolant pressure boundary materials. In lieu of these specific requirements, the present version of Appendix G cited above requires that for a reactor vessel which was constructed in conformance with an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition, the fracture toughness data and data analyses must be supplemented in a manner approved by the Director, Office of Nuclear Reactor Regulation, to demonstrate equivalence with the present fracture requirements of Appendix G. As discussed in Supplement No. 1, the Fermi-2 reactor vessel was constructed in compliance with an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition. We stated in Section 5.2.1.3 of Supplement No. 1 that the alternative methods proposed by the applicant to demonstrate compliance with Appendix G had been reviewed, evaluated, and found to provide the safety margin required by Appendix G. Accordingly, the applicant has supplied sufficient information to demonstrate equivalency with the fracture toughness requirements of the present version of Appendix G to 10 CFR Part 50. On this basis, we find that the exemptions to Appendix G cited in Supplement No. 1 to the SER are no longer required.

5.2.1.2 Compliance with Appendix H, 10 CFR Part 50

Previous versions of Appendix H required that the surveillance program conducted prior to the first capsule withdrawal comply with the 1973 edition of ASTM E 185. The present version of Appendix H requires that the surveillance program conducted prior to the first capsule withdrawal comply with the requirements of the edition of ASTM E 185 that was current with respect to the

ASME Code to which the rector vessel was purchased. The applicant has indicated that the Fermi-2 surveillance program complies with this requirement. Accordingly, the Fermi-2 surveillance program complies with the revised requirements of Appendix H to 10 CFR 50. On this basis, we find that the exemptions to Appendix H cited in Supplement No. 1 to the SER are no longer required.

5.2.3 Materials

5.2.3.1 Material Specifications and Compatibility with Reactor Coolant

In the SER we issued in July 1981, we found that the materials used for the construction of the Fermi-2 facility which could come in contact with the reactor coolant, met applicable standards such as Section III of the ASME Boiler and Pressure Vessel Code and ANSI B.31.7 and conformed to our guidelines in Regulatory Guides 1.36 and 1.56. On this basis, we found that the applicant had proposed acceptable materials of construction.

However, during the assessment of construction performed by the Duke Power Company in July 1984 at the Fermi-2 facility (Refer to Inspection Report 50-341/84-21), the acceptability of the primary containment coatings was reported as an unresolved item (50-341/84-21-03). The concerns were related to: (1) repair and touch-up of damaged coatings; (2) the quantity of unqualified coatings; and (3) an assessment of uncoated surfaces. The primary concern is that debris generated inside containment under design basis accident conditions could adversely affect the performance of post-accident safety-related fluid systems. Specifically, the long-term cooling mode of the RHR system might be degraded if a sufficient quantity of debris were to be deposited into the pressure suppression pool and be drawn from there into the suction lines of the ECCS.

When the Fermi-2 application for an operating license was docketed in April 5, 1975, we had no specific requirements for qualified coatings. Subsequently, we issued guidance for qualified coatings in Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants." We also recommended adoption of ANSI 101.2, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities" in Section 6.1.2, "Organic Materials", of the Standard Review Plan (NUREG-75/087) on November 24, 1975.

While the Fermi-2 coatings were applied prior to the issuance of our guidance on this matter, we are concerned that unqualified coatings inside containment are capable of generating debris which could under design basis accident conditions, adversely affect the performance of fluid systems required for long-term removal of the reactor decay heat.

In response to our concern on this matter, the applicant provided additional information in its letters dated August 28 and October 11, 1984 and January 10 and January 24, 1985. In these letters, the applicant estimated that 99 cubic feet of debris could be generated following a DBA assuming all unqualified coatings inside containment failed. Assuming a 50 percent packing fraction, we estimate that there is a potential of about 197 cubic feet of debris being generated. This debris is divided into three types:

(a) Inorganic zinc: 147 cubic feet of debris; less than 20 microns particle size at a density of 217 lbs/ft^3 density.

(b) Organic paints: 0.3 cubic feet of debris; 100-1,000 microns particle size at a density of 90 to 150 lbs/ft^3.

(c) Mill scale and varnish: 50 cubic feet of debris; 4 to 60 microns particle size at a density of 350 lbs/ft^3.

The applicant assumed that all of the unqualified materials cited above will fail and separate from their surfaces under design basis accident (DBA) conditions and will be transported into the suppression pool (i.e., the torus). The applicant also assumed that this debris in the torus would be evenly distributed and suspended within the suppression pool water volume. These conservative assumptions result in a volumetric debris concentration in the suppression pool of 0.17 percent. Assuming that all the zinc in the unqualified coating is converted to zinc oxide, this concentration would be 0.22 percent.

The debris cited above could cause the following adverse effects: (1) blockage of the ECCS suction strainer; (2) blockage of the containment spray and the reactor pressure vessel core spray and feedwater spargers; (3) blockage of the ECCS valves; (4) generation of hydrogen inside the containment; (5) fouling of heat transfer surfaces; (6) degradation of ECCS performance by ingestion of fine particles of paint debris into the piping system pumps, valves and heat exchangers; and (7) deposition of debris in reactor pressure vessel. Each of this items of concern are discussed below.

Item 1. The ECCS suction strainers in the RHR, core spray and HPCI lines will allow passage of particles less than 0.125 inch in diameter. Since the debris particles are not expected to exceed 1000 microns (0.039 inch), we conclude that ECCS suction strainer blockage will not occur.

Item 2. The containment spray nozzles also permit passage of 0.125 inch diameter particles. The reactor pressure vessel core spray system has no intervening obstructions in its flow path. The core spray and feedwater spargers have a minimum flow passage of at least 0.5 inch. Since the debris particle size is much smaller than that of the spray nozzles and spargers, we conclude that flow blockage or performance degradation will not occur.

Item 3. We do not conclude that the ECCS electric motor-operated valves with seating surfaces perpendicular to the flow stream will be crud traps since the coating debris should not settle out on perpendicular surfaces. Further, as the valves close, the flow velocity of the ECCS water increases as the flow area across the seat decreases, thereby flushing off any deposited material. Coating debris should not affect valve operability since similar valves in industry perform acceptably with particulate concentrations much higher than the highly conservative estimates of the maximum debris concentrations which might occur in the Fermi-2 facility.

Item 4. The inorganic zinc coatings upon exposure to water at DBA containment temperatures, could oxidize to form zinc oxide and thereby release hydrogen gas into the containment. The DBA hydrogen generation analysis for the corrosion of aluminum and galvanized steel as well as the inorganic zinc coatings, has been addressed in Section 6.2.5.3.1 of the Fermi-2 FSAR. The additional hydrogen generated from radiolysis of unqualified organic coatings is negligible compared to that from the galvanized steel and inorganic zinc coatings. Accordingly, our conclusion in the SER on this matter remains the same. Moreover, we concluded in our SER that the combustible gas control systems conform to Regulatory Guide 1.7, equal or exceed the requirements of General Design Criteria 42 and 43 of Appendix A to 10 CFR Part 50 and Section 50.44 of 10 CFR Part 50 and are, therefore, acceptable.

Item 5. The RHR heat exchangers are vertical units with flow velocities between 6 to 10 ft/sec. at rated flow. The RHR primary fluid is taken from the water in the torus and is pumped through the shell side with no opportunity for any debris to cause a blockage in the heat exchanger. Due to the vertical orientation of the heat transfer surfaces and since the flow velocities exceed settling velocities, we conclude that particulate deposition and fouling should be minimal and that the heat transfer capacity of the heat exchangers should not be affected.

Item 6. As discussed above, the maximum debris volumetric concentration in the suppression pool water should not exceed 0.22 percent. In NUREG-0897, "Containment Emergency Sump Performance" and in NUREG/CR-2792, "An Assessment of Residual Heat Removal and Containment Spray Pump Performance Under Air and Debris Ingesting Conditions," we stated that a debris volumetric concentration of about one percent should not impair pump performance. The centrifugal pumps in the Fermi-2 facility are similar to those used in fossil power plant which have demonstrated long-term operability when pumping water containing silt and corrosion particles. Since the Fermi-2 facility uses single-stage pumps, concerns about the effects or particulates and debris on the inter-stage bushings are not applicable. Moreover, the cyclone separators provided for these pumps will reduce the probability of a pump bearing failure. Accordingly, we conclude that there is reasonable assurance that ECCS performance should not deteriorate due to ingestion of fine debris particles.

Item 7. As we stated above, the maximum volume of debris (i.e., total unqualified paints, mill scale and varnish) which could be deposited in the reactor pressure vessel is 197 cubic feet. A large fraction of this debris will not be transported to the torus. Further, an additional fraction of that debris which enters the torus, would settle out before it could be drawn into the ECCS suction line. It is, therefore, reasonable to state that most of this debris will probably not reach the ECCS suction strainer where it can be pumped into the reactor vessel. Moreover, even if all this debris were to settle on the bottom of the reactor vessel (i.e., the lower plenum), the fuel channels would not be blocked.

Moreover, debris which is entrained in the reactor coolant should flow up through the fuel channels without causing core flow blockage since the smallest restriction in the core is 0.040 inch in the spacer grid and the largest debris size is about 0.039 inch.

In the very unlikely event of a complete blockage of the ECCS suction strainers, the emergency operating procedures (EOPs) instruct the reactor operator to use alternate systems other than the ECCS for water makeup in the reactor pressure vessel (RPV). Specifically, the Fermi-2 EOPs for the RPV level control provides the operator with the following options for core cooling and level control:

(a) Normal feedwater using the electric-driven condensate and heater feed pumps taking suction from the condenser hotwell and injecting into a depressurized reactor vessel.

(b) Continued operation of the control rod drive water pump, injecting condensate into the RPV via the control rod drives.

(c) Aligning the standby liquid control system (SLCS) in the test mode thereby injecting condensate from the SLCS test tank into the RPV.

(d) Aligning the core spray system to take suction from the condensate storage tank and injecting it into the RPV.

(e) Utilizing the RHR to RHR service cross-tie and injecting water from the RHR reservoir into the reactor vessel via the RHR service water pump. (Note that this is a last resort since this water is not demineralized or otherwise cleaned.)

In addition to the coolant injection mechanisms cited above, there are two additional options which are available for coolant injection:

(f) Activate the standby feedwater system, using the dedicated electric-driven pumps and injecting directly into the RPV.

(g) Align the fuel pool cooling (FPC) system pumps with the RHR pumps and heat exchangers. In this mode, the FPC pumps take suction from the fuel pool skimmer tanks (with makeup provided via fire hose stations) and return water via the RHR pumps and heat exchangers into the depressurized RPV or into the containment as desired.

As can be seen from the preceding discussion, the operators have great latitude in utilizing all available water sources onsite, including the hotwell, the condensate storage tanks, the RHR reservoir and the fuel pool.

The Fermi-2 torus water management system (TWMS) is designed to continuously process water through a demineralizer system to maintain water quality and to prevent any gradual build-up of particles, contamination and sludge. During the post-LOCA long-term cooling period, this system can be used to clean the water in the torus in the event that debris is deposited there.

The applicant has committed in its Report No. DECO-12-2191 which was submitted in October 1984, to a visual inspection of containment coatings during refueling outages. Damaged areas (e.g., corrosion, blistering or peeling, and discoloration of coatings) will be repaired. These areas will be cleaned using the manufacturer's recommended surface preparations. New coatings will be applied in accordance with acceptable application criteria.

Based on our evaluation, we find that there is reasonable assurance that any debris generated by a failure of coatings inside of containment in the event of a design basis accident, (DBA), will not adversely affect the performance of the safety-related fluid systems required to achieve a safe shutdown following the DBA. Accordingly, we find that the matter of unqualified coatings inside containment is resolved.

6 ENGINEERED SAFETY

6.2 Containment Systems

6.2.3 Secondary Containment

In the SER we issued in July 1981, we found the applicant's analysis of the length of time to reduce the pressure in the secondary containment to a negative value of 0.25 inches water gauge (w.g.), to be acceptable. This time was calculated as six minutes. In accordance with the proposed Fermi-2 Technical Specifications, this value was is to be verified periodically by test. The applicant has now modified this drawdown time; this supplement evaluates this change.

Negative Pressure Differential

The applicant has committed to maintain the secondary containment at a negative pressure of one-quarter inch w.g. plus or minus one-eight inch w.g. Paragraph 4.6.5.1.a of the Fermi-2 Technical Specifications requires that the vacuum within the secondary containment with respect to the environment, be greater than or equal to a negative 0.125 inches w.g. In the event of a loss-of-coolant accident (LOCA), the pressure in the secondary containment could increase before returning to a vacuum. In its letter dated February 12, 1985, the applicant provided the results of its analysis in which it calculated this pressure transient to determine the time necessary to reduce the secondary containment internal pressure to minus one-quarter inch w.g. The principal assumptions used by the applicant in the analysis include:

(a) No credit was taken for exfiltration from the secondary containment.

(b) Infiltration to the secondary containment was included as a function of the pressure differential with a maximum infiltration rate of 3000 standard cubic feet per minute (scfm) at a secondary pressure of minus 0.25 inches w.g.

(c) It was assumed that there is no heat-transfer to the external environment.

(d) Heat transfer to interior secondary containment walls, floors and ceilings was included in the analysis.

(e) Heat transfer from the torus room to the secondary containment was based on the heat flow through the pressure relieving doors in the corner room basement walls.

(f) Only one standby gas treatment system (SGTS) filter train was assumed to function with a minimum volumetric flow rate of 3800 scfm.

(g) Off-site power was assumed lost during the postulated LOCA.

(h) It was assumed that activation of the SGTS was delayed by 33 seconds and activation of the safety-related emergency area coolers was delayed by 38 seconds in accordance with the loading sequence for the emergency diesel-generators.

(i) The pump room in the reactor heat removal (RHR) complex and the core spray and RCIC pump rooms in the reactor building sub-basement were treated separately from the main secondary containment volume. These rooms have their own engineered safety feature emergency coolers to handle emergency equipment and lighting heat loads.

(j) Heat loads generated prior to the startup of the emergency room coolers at 38 seconds, were neglected since the RHR pump actuation does not occur until 13 seconds following the start of the postulated LOCA, leaving only 25 seconds of time without the room coolers functioning.

(k) A bounding external temperature of -10°F was used in the analysis.

(l) The initial secondary containment pressure was assumed to be a negative 0.125 inches w.g.

The applicant's analysis indicates that, based on these assumptions, a negative pressure of one-quarter inch w.g. in the secondary containment will be re-established in ten minutes. The Fermi-2 Technical Specifications include a requirement to conduct periodic test to confirm both the infliltration rate and the drawdown time.

We have reviewed the applicant's method of analysis and its results including the assumptions cited above and conclude that they are in conformance with Branch Technical Postion CSB 6-3 and, therefore, the methodology, assumptions and results are acceptable. (Refer to Sections 6.4.1 and 15.2.3 of this supplement.)

6.2.7 Containment Leakage Testing

In the SER we issued in July 1981, we stated that we did not have sufficient information to evaluate the compliance of the Fermi-2 facility with the requirements of Appendix J to 10 CFR Part 50. On this basis, we concluded that the applicant's reactor containment leakage testing program was not acceptable. In Supplement No. 1 to the SER, we provided a detailed evaluation of the applicant's containment leakage testing program including seven categories of potential leakage paths. We stated in Supplement No. 1 that with the exception of two open items, we found that the applicant's proposed leakage testing program met the requirements of Appendix J. Subsequently, the applicant submitted a revised bypass leakage program; our favorable evaluation of this program was provided in Supplement No. 2 to the SER. We stated our finding in this supplement that the applicant's bypass leakage program was acceptable.

The applicant has recently requested two exemptions from the requirements of the Appendix J. The first of these is related to the testing of the main steam isolation valves (MSIVs) while the second is related to testing of the air lock. This supplement contains our evaluation of these two exemption requests.

Main Steam Isolation Valves

We require in Paragraph II.H.4 of Appendix J to 10 CFR Part 50 that the main steam isolation valves in boiling water reactors be leak tested at the peak calculated containment pressure associated with the design basis accident (refer to Paragraph III.C.2 of Appendix J). Furthermore we require that the measured leak rates be included in the summation of the local leak rate tests (refer to Paragraph III.C.3 of Appendix J).

The applicant has submitted additional information describing its proposed methods for complying with the requirements of Appendix J to 10 CFR Part 50 with respect to its proposed method of leak testing the Fermi-2 MSIVs. The applicant has also requested in its letter dated October 22, 1984, an exemption from the requirements of Appendix J so as to leak test the MSIVs by pressurizing the volume between each pair of isolation valves at a reduced pressure and to exclude the measured leakage of the post-accident leakage control system from the summation of the measured local leak rate tests. Based on our review of this exemption request, we have determined that an exemption from the full pressure test requirement of Appendix J is both required and justified. Our bases for our conclusion on this matter are discussed below.

Each of the four main steam lines is provided with two quick-acting main steam isolation valves (MSIVs) which are designed to provide sealing in the direction of post-accident containment leakage to the environment (i.e., the condition where the pressure inside the reactor pressure vessel is higher than the external atmospheric pressure). In the event of a loss-of-coolant accident (LOCA), the main steam line leakage control system will maintain a positive air pressure between these two main steam line isolation valves. Leakage through the inboard isolation valve will be discharged into the containment. An infiltration analysis for this potential source of leakage into the primary containment was performed based on an assumed leakage rate of 100 standard cubic feet per hour for all four main steam lines. The results of this analysis indicate a small increase in the post-accident primary containment pressure over a 30-day period (i.e., an increased pressure less than 5 psig).

The design of the main steam isolation valves is such that testing in the reverse direction tends to unseat the valve. This will significantly increase the leakage past the inboard valve since testing of the two MSIVs simultaneously by pressurizing between these valves at 1.1 Pa as specified in Appendix J, would completely lift the disc of the inboard valve and would result in a meaningless test. The test proposed by the applicant calls for a test pressure of 25.0 pounds per square inch gauge (psig) to avoid lifting the disc of the inboard isolation valve. A test pressure of this magnitude across the isolation valves is much greater and thus more conservative than the 2 to 6 psig which will exist during postulated post-accident conditions when the leakage control system is operating.

Furthermore, the leakage caused by operation of the main steam leakage control system is into the containment. Accordingly, we find it acceptable to exclude the main steam isolation valve leakage from the summation of the local leak rate tests since there will be no significant leakage of radioactive fission products to the environment caused by operation of this system.

In addition to the two innermost, quick-acting isolation valves, a third block valve is located in each of the four main steam lines to provide a redundant leakage control volume for the back-up pressurization system. Paragraph 3.6.1.4 of the Fermi-2 Technical Specifications requires that functional operability tests of the leakage control system be performed every 18 months to verify that the leak tightness integrity of this third isolation valve and to ensure that this leakage control system has the capability to maintain the control volume at a pressure greater than that in the containment.

We conclude that leak testing of the main steam isolation valves in the manner described above is an acceptable alternative to the specific requirements of Appendix J. Accordingly, we hereby grant an exemption from the specific requirements of Appendix J as described above.

Air Lock Testing

In its letter dated January 26, 1985, the applicant has requested an exemption from the full-pressure air lock testing required by Paragraph III.D.2(b)(ii) of Appendix J to 10 CFR Part 50. Specifically, this paragraph states:

> "Air locks open during periods when containment integrity is not required by the plant's Technical Specifications shall be tested at the end of such periods at not less than Pa."

This provision requires that if an air lock is opened during either Operational Condition 4 or 5, it must be tested satisfactorily via an overall air lock door test prior to plant entry into Operational Condition 3.

The applicant states in its January 26th letter that performing this test would impose a significant hardship on it in that an average time of between 11 to 13 hours would be required to perform the overall test as specified in Appendix J. This time interval is required since the following steps need to be performed to complete the overall test: (1) install 14 tie downs to the interior air lock door; (2) pressurize the air lock to Pa; (3) wait for the air lock volume to stabilize and perform the leakage measurement; and (4) depressurize the air lock, remove and store the tie downs.

The applicant proposes in its January 26th letter, an alternative method of verifying the air lock integrity in lieu of the method specified in Paragraph III.D.2(b)(ii) of Appendix J. The applicant's proposed method consists of leak testing the seals of the inner and outer doors at Pa rather than testing the entire door, provided no maintenance has been performed on the air lock since the last successful test. The proposed approach represents a nearly equivalent test since the seals are the most significant leakage source. The acceptance criteria for the seal test is proposed to be a leak rate less than or equal to 5 standard cubic feet per hour at a pressure, Pa, equal to 56.5 psig. If an air lock maintenance will have been performed since the last successful test, an overall air lock leakage test at Pa shall be performed prior to entering Operational Condition 3. It is our conclusion that the proposed air lock seal test performed as described above, will provide reasonable assurance that the leaktightness integrity of the air lock will be maintained.

On this basis, we find that the applicant's request for an exemption from the requirements of Paragraph III.D.2(b)(ii) of Appendix J to 10 CFR Part 50, is acceptable. Accordingly, we have incorporated the applicant's proposed method of performing air lock tests as discussed above and in the applicant's letter of January 26, 1985, into the Fermi-2 Technical Specifications.

6.4 Control Room Habitability Systems

6.4.1 Radiological Dose Protection

In the SER we issued in July 1981, we concluded that the calculated dose estimates for the Fermi-2 control room meet the requirements of General Design Criterion (GDC) 19 of Appendix A to 10 CFR Part 50. We further concluded that the design of the Fermi-2 control room provides an acceptable means of maintaining the control room in a safe and habitable condition by providing adequate protection under radiation accident conditions.

Subsequently, the applicant submitted Amendment No. 58 to its FSAR in July 1984 in which it revised its estimate of the time to draw down the secondary containment to a partial vacuum. (Refer to Sections 6.2.3 and 15.2.3 of this supplement.) Based on the applicant's latest estimate of the secondary containment drawdown time, we have revised Table 6.4-1 of the SER and replaced it in its entirety with Table 6.4-1 of this supplement. The only significant change in this table is the calculated thyroid and whole-body doses to the control room operators in the first eight hours following the initiation of the postulated loss-of-coolant accident. The net result is that the total thyroid dose increases from an estimated value of 15.7 rem to 16.1 rem while the estimated whole-body dose increases from 1.50 rem to 1.53 rem. Both these revised doses are within the guideline values contained in GDC 19 of Appendix A to CFR Part 50. Accordingly, we find that the calculated dose estimates for the Fermi-2 control room still meet the requirements for GDC 19.

With respect to the habitability of the Fermi-2 control room under postulated accident conditions, our inspection of the Fermi-2 control room ventilating ducts during the week of November 5 through 9, 1984, indicated that one of the assumptions used in estimating the dose rates to the control room operators might not be valid. (We assume in our analysis that no more than 10 cubic feet per minute of unfiltered air infiltrates into the control room.) The results of this inspection effort is described in Report No. 50-341/84-43, dated January 11, 1985. In this report, we expressed our concern that contrary to the guidelines in Regulatory Guide 1.52, silicone sealants were used to seal leaks found during leakage acceptance tests on the control room filter system ducts and housings. Both of these components are engineered safety features (ESF) and, therefore, must meet applicable guidelines. In response to our concerns, the applicant submitted additional information in its letter dated January 8, 1985, to clarify its FSAR commitment regarding Regulatory Guide 1.52. The applicant stated that the Fermi-2 heating, ventilating and air-conditioning (HVAC) ducts, filters and filter housings were designed prior to issuance of both Regulatory Guide 1.52 and ANSI/ASME Standard N509-1976, which is referred to in Regulatory Guide 1.52. The applicant further stated that the Fermi-2 HVAC systems were designed to the standards available at that time; namely, Sheet Metal and Air-Conditioning Contractors' National Association, Inc. (SMACNA) High Velocity Duct Construction Standards and ORNL-NSIC-65. The applicant also states in its FSAR that the control center filtration system ductwork component design criteria are not in conformance with Position C.3.n of Regulatory Guide 1.52, which states in part that the design of the HVAC ductwork should conform with Section 5.10 of ANSI N509/1976. Instead, the applicant states in its FSAR that the design of the HVAC ductwork is in conformance with Section 2.8 of ORNL-NSIC-65. In its letter dated January 8, 1985, the applicant committed to submit a revision to its FSAR which would provide an exception to its conformance with Section 2.8 of ORNL-NSIC-65. Specifically, the applicant states that the longitudinal seams of the HVAC ductwork: (1) are a mechanical lock type; (2) are externally brazed; (3) have a sealant applied to the internal duct seam; and (4) have a sealant applied externally to the seam to enhance low leakage characteristics. However, ANSI N509-1976 which is referred to in the latest revision (1978) of Regulatory Guide 1.52, requires that ducts for ESF systems shall be welded. Further, the latest revision of ANSI N509-1976 (i.e., ANSI N509-1980) requires that longitudinal seams for ESF ducts may only use a mechanical lock type qualified by test for leak-tightness. Both ANSI N509-1976 and ANSI N509-1980 refer to the SMACNA standards which address mechanical lock seams and the use of sealants to make seams airtight.

We have evaluated the applicant's submittals cited above and find them acceptable, pending verification in a forthcoming amendment to the FSAR. However, the applicant has not provided reasonable assurance that certain portions of the control center HVAC ductwork will be sufficiently leak-tight to minimize inleakage of unfiltered air into the HVAC ducts to assure adequate radiological dose protection for control room operators over the life of the Fermi-2 facility. Specifically, portions of the ESF make-up filter system duct and the ESF recirculation filter system duct, are outside the control room zone. Silicone sealant is used on the seams of these ducts to minimize inleakage of unfiltered air into the ducts to assure that the estimated radiological dose rates for the control room operators meet the requirements of GDC 19 of Appendix A to 10 CFR Part 50. We find that the applicant has not provided reasonable assurance that this silicone sealant will adequately minimize the inleakage rate over the 40-year life of the Fermi-2 facility. To address our concerns on this matter, the applicant proposed in its letter dated January 8, 1985, to test the airflow rates of the filtered outside air, the recirculation airflow rate, and the total make-up airflow rate in the control room in accordance with the applicable Technical Specification requirements to identify any significant increases in inleakage which might occur due to potential degradation of the silicone sealant on the HVAC duct seams. We find this commitment acceptable and conclude that this issue is closed.

During emergency operation, the intake of the control center air-conditioning system is closed and this system recirculates about 31,000 cubic feet per minute of air which is not passed through the charcoal filters and the HEPA filters. A portion of the duct for this ESF system is outside the control room zone and a silicone sealant is used on the seams of this duct to minimize the unfiltered control room infiltration rate to provide radiological dose protection for the control room operators. However, the applicant has not provided reasonable assurance that the design of the external portion of the HVAC duct, including the silicone sealant, will minimize the unfiltered control room infiltration rate over the 40-year life of the plant. In its letter dated January 8, 1985, the applicant committed to provide reasonable assurance of radiological protection for the control room operators by providing either: (1) periodic leakage tests on the external portion of the air-conditioning duct; (2) replacement of the external portion of this duct with a welded duct; or (3) a demonstration that the design of that portion of this duct which is external to the control room will provide adequate protection over the life of the plant. We will place a condition in the license incorporating the applicant's commitment on this matter. On this basis, we find that the design of the Fermi-2 control room will still meet the requirements of GDC 19 of Appendix A to 10 CFR Part 50.

Table 6.4-1A Staff assumptions and estimates of the radiological consequences to control room operators following a loss-of-coolant accident at Fermi-2

Control room free volume	50,536 ft^2	
Filtered recirculation flow rate	1200 cfm	
Recirculation filter efficiencies	95% elemental iodine 95% organic iodine 95% particulate iodine	
Unfiltered control room infiltration rate (assumed)	10 cfm	
Control room filtered air pressurization rate	1800 cfm	
Pressurization (makeup) air filter efficiencies	95% for elemental iodine 95% for organic iodine 95% for particulate iodine	
Duration of accident	30 days	
Breathing rate of operators in control room for the course of the accident	3.47×10^{-4} m^3/sec	
Meteorological dispersion factors (wind speeds for all sectors)		
0 - 8 hours	1.83×10^{-3} sec/m^4	
8 - 24 hours	1.24×10^{-3} sec/m^3	
24 - 96 hours	4.6×10^{-4} sec/m^3	
96 - 720 hours	1.2×10^{-4} sec/m^3	
Iodine protection factors	203	
Geometry factor for finite cloud model	30	
Doses to control room operators	Thyroid dose (rem)	Whole-body dose (rem)
0 - 8 hours	4.9	1.11
8 - 24 hours	3.9	.27
24 - 96 hours	4.4	.11
96 - 720 hours	2.9	.04
TOTAL	16.1	1.53

7 INSTRUMENTATION AND CONTROL

7.1 Introduction

7.1.2 Specific Findings

Safety System Setpoints

In the SER we issued in July 1981, we stated that the resolution of the safety-related instrumentation setpoints, the sensor ranges and the sensor accuracies would be provided in the review of the final Technical Specifications for the Fermi-2 facility. Subsequently, a generic approach has been established for resolving the issue of the instrument setpoints and the sensor accuracies. The final resolution of these two matters has been deferred and an interim resolution accepted. (Refer to Section 7.2.2 of this supplement.)

7.2 Reactor Trip System

7.2.2 Specific Findings

Trip Setpoint Values

In the SER we issued in July, 1981, we identified a concern regarding the selection of instrument trip setpoints and stated that the applicant had to submit additional information to demonstrate its conformance with our regulations relevant to the issue of protection system setpoints. The applicable regulations are Sections 50.36 and 50.46 of 10 CFR Part 50 and General Design Criterion (GDC) 20 of Appendix A to 10 CFR Part 50. We require in GDC 20 that the protection system shall be designed to: (1) initiate automatically the operation of appropriate systems, including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences; and (2) sense accident conditions and to initiate the operation of systems and components important to safety.

We define in Section 50.36 of 10 CFR Part 50, limiting safety system settings for nuclear reactors as settings for automatic devices related to those variables having significant functions. We require that when a limiting safety system setting is specified for a variable on which a safety limit has been placed, that setting shall be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. We specify in Section 50.46 of 10 CFR Part 50, the performance criteria for the emergency core cooling systems. These criteria include a maximum peak cladding temperature, a maximum allowable amount of cladding oxidation, a maximum total amount of hydrogen generated, and a requirement that the core geometry remain amenable to cooling for long-term removal of decay heat. We also provide guidance on acceptable methods for complying with these regulations in Regulatory Guide 1.105, "Instrumentation Setpoints."

The applicant joined with several other BWR owners to form the Licensing Review Group (LRG) - Instrumentation Setpoints Methodology Group (ISMG). On July 14, 1983, we met with the ISMG for a presentation of an outline of its proposed methodology for establishing instrument trip setpoints. To respond to our questions, another meeting with the ISMG was held on January 31, 1984. In our letter dated May 15, 1984, to J. F. Carolan (Chairman, ISMG), we provided our assessment of the ISMG methodology. Our evaluation identified several deficiencies in the proposed ISMG methodology. To resolve these deficiencies, we requested that the ISMG provide additional information for ten specific concerns we identified. In response to our evaluation and request for additional information, the ISMG provided an action plan for resolving the outstanding issues in its letter dated June 29, 1984. In its letter dated October 5, 1984, the applicant committed to the work scope and schedule proposed in the ISMG action plan. The final acceptability of the protection system instrumentation setpoints will be addressed following completion of our review of this matter.

We conclude there is reasonable assurance, based on the information supplied in our meetings with the ISMG, that the forthcoming more detailed information on the setpoint methodology being developed by this group will verify the acceptability of the proposed setpoints. In the interim, we find the setpoints proposed by the applicant in its Technical Specifications for the Fermi-2 facility, to be acceptable.

7.3 Engineered Safety Feature Systems

7.3.2 Specific Findings

Low-Low Set Relief Logic System

The issue of establishing an engineered safety feature (ESF) to minimize pool dynamic loads in the torus of the Fermi-2 Mark I pressure suppression system was not considered in the SER we issued in July 1981. However, we have subsequently raised this issue as a concern for the BWR pressure suppression systems. It is our position that the pressure settings at which some of the safety/relief valves (SRVs) discharge steam into the torus and the time interval of this discharge can significantly enhance the safety of the Fermi-2 facility.

In response to our concern on this matter, the applicant has modified the actuation circuitry of the Fermi-2 SRVs to incorporate a low-low set (LLS) relief function to reduce challenges to the SRVs and to reduce the pool dynamic loads resulting from the discharge of the SRVs. The LLS is an automatic SRV control system which, upon initiation, will assign lower opening and closing setpoints to two of the fifteen SRVs in the Fermi-2 facility. These lower setpoints are selected such that once initiated, the LLS controlled SRVs will stay open longer than would otherwise have occurred thereby releasing more steam (i.e., energy). This will result in a longer interval between SRV discharges since it will take a longer time for the decay heat in the reactor core to repressurize the reactor pressure vessel is sufficient steam is discharged through the SRVs to reduce the pressure in the reactor well below the level for normal operation.

Each of the two separate logics for the LLS relief functions consists of a reactor pressure sensing channel which is enabled by a separate reactor vessel high pressure signal indicating a scram is required and a signal indicating that one or more SRVs are open. The two LLS SRVs have slightly different opening and closing setpoints so that only one SRV at a time will reopen on increasing pressure following an initial SRV actuation and closure. This arrangement serves to dampen reactor pressure surges. The LLS logic automatically seals itself into control of the two selected valves and actuates an annunciator in the control room. This logic remains sealed in until manually reset by the operator.

Each of the two LLS logics are powered by 130 Vdc sources from separate divisions. The LLS function is designed with redundant components to satisfy the single failure design criteria; no single LLS failure will prevent opening of an SRV or result in an inadvertent LLS logic seal-in. The LLS has been designed and constructed in accordance with the requirements of IEEE Standards 279-1971, 323-1974 and 344-1975 as well as with the provisions of Regulatory Guide 1.22.

Based on our review of the proposed Fermi-2 LLS logic system, we find that the design of this ESF system meets the applicable regulatory requirements and guidelines. Accordingly, we find the design of the low-low set relief logic system to be acceptable.

7.4 Systems Required for Safe Shutdown

7.4.2 Specific Findings

Standby Liquid Control System

In the SER we issued in July 1981, we concluded that the design of the standby liquid control system (SLCS) is acceptable. This conclusion was based on the applicant's statement in its FSAR that the system is identical to the designs we approved for Dresden 2 and 3.

The SLCS at the Fermi-2 facility is a special capability backup system provided by the applicant to satisfy the requirements of General Design Criterion 26 of Appendix A to 10 CFR Part 50. This system is independent of, and diverse to, the control rods for shutting down the reactor from full power steady-state operating conditions at anytime in the core life in the event that multiple failures prevent the insertion of the control rods. Portions of the system are redundant. Accordingly, this manually-operated system is subject to a single failure. Although the SLCS is not fully redundant within itself, power busses, pumps and explosive-operated injection valves are redundant so that a single component may be removed from service for maintenance during plant operation.

Subsequent to the issuances of the SER, we reviewed the SLCS design features in greater detail. Based on this additional review, we found differences between the Fermi-2 design of the SLCS and that of Dresden 2 and 3. The SLCS at Dresden 2 and 3 are safety-related, seismically qualified, Class 1E systems. Although functionally similar in design to Dresden 2 and 3, the Fermi-2 SLCS had not been classified by the applicant as safety-related. Accordingly, the SLCS design did not conform to the same standards that would be applied to a safety-related system. Further, the system components had not been covered by the policies and procedures of the applicant's QA program.

On July 17, 1983, we met with the applicant's representatives and the General Electric Company to discuss the Fermi-2 SLCS. At this meeting, the applicant provided a detailed comparison between the Fermi-2 SLCS and that at Dresden 2 and 3. From this comparison, we determined that there were significant differences between both the documentation of component quality and the electrical separation at the two plants (i.e., Fermi-2 and Dresden 2,3).

To resolve our concerns in these areas, the applicant proposed to: (1) perform a QA/QC audit of the SLCS cable terminations; (2) provide the results of high voltage tests on the SLCS cables; (3) perform a drawing review and an engineering evaluation to confirm the adequacy of the electrical separation; (4) review the SLCS design and construction documentation and walk-down the SLCS to verify the quality of construction; (5) henceforth apply the same QA controls and procedures as is applied to safety-related equipment after completion of the SLCS startup test program; and (6) propose plant specific Technical Specifications for the Fermi-2 facility to ensure the reliability and availability of this system.

In its letter dated June 22, 1984, the applicant provided additional information on its SLCS. The QA/QC audits, tests, evaluations, design review and system walkdown confirmed that the SLCS has been built to what the applicant considers good quality construction standards.

Based on our review of the Fermi-2 SLCS, the applicant's program to confirm the quality of its design and construction, and the applicant's proposal to include the SLCS in its safety-related QA program, we find that the differences between the Fermi-2 SLCS and that at Dresden 2 and 3, are acceptable even though the designs are not identical as originally stated by the applicant. On this basis, we find the Fermi-2 SLCS to be acceptable.

7.5 Safety-Related Display Instrumentation

7.5.2 Specific Findings

Regulatory Guide 1.97

In the SER we issued in July 1981, we addressed the acceptability of the post-accident monitoring instrumentation. We provide guidance in Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," for instrumentation used to monitor plant variables and systems during and following an accident. In response to our request, the applicant is currently performing a review to determine the level of its conformance to the provisions of Regulatory Guide 1.97. In its letter dated April 15, 1983, the applicant provided a status of its review and committed to provide by June 1985, a final report on Regulatory Guide 1.97. We find this approach and schedule acceptable.

7.7 Control Systems Not Required For Safety

7.7.2 Specific Findings

Anticipated Transient Without Scram--Recirculation Pump Trip

In the SER we issued in July 1981, we concluded that the design of the recirculation pump trip (RPT) to resolve the issue of an anticipated transient without scram (ATWS), is acceptable. Subsequently, the Commission amended its regulations on June 26, 1984, to add Section 50.62 to 10 CFR Part 50. This section of our regulations now requires each boiling water reactor facility to have an alternate rod injection system which is diverse to the reactor trip system from the sensor output to the final actuating device. The alternate rod injection system must also have redundant, scram air header exhaust valves. In addition, each nuclear power plant with a boiling water reactor must have a standby liquid control system capable of injecting 86 gallons per minute of a solution containing sodium pentaborate at a concentration of 13 percent by weight. The initiation of the standby liquid control system must be automatic for those plants granted a construction permit prior to July 26, 1984, and which have already been designed and built to include this feature. Further, each boiling water reactor facility must have equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS.

In accordance with the provisions of Section 50.62 of 10 CFR Part 50, we require the applicant to submit a schedule for meeting the requirements of Section 50.62 no later than 180 days following the issuance by the NRC of quality assurance guidance for the ATWS mitigating system. We will review the design of the ATWS mitigating features for the Fermi-2 facility when they are submitted to verify compliance with Section 50.62 of 10 CFR Part 50. We will provide the results of our review in a future supplement to the SER.

9 AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

9.1.4 Fuel Handling System

9.1.4.1 Status of Heavy Load Handling

In the SER we issued in July 1981, we stated our concerns and our requirements for the handling of heavy loads near spent fuel. We also stated that our guidelines for the safe handling of heavy loads are contained in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." Since the development by the applicant, and our subsequent review, of those changes and modifications to the design, operation, maintenance, inspections and testing of cranes and hoists as well as the development of the appropriate operator training would extend over a long period, we required the applicant to institute certain interim measures which could be readily implemented. These interim measures were to be implemented prior to the issuance of the Fermi-2 operating license. In its letter dated May 12, 1981, the applicant committed to implement these interim measures for handling heavy loads. We stated in the SER that we found this commitment acceptable. We also stated in the SER that we required a Technical Specification regarding compliance with ANSI Standard B30.2-1967 for the reactor building crane.

We required in our generic letter on this matter dated December 22, 1982, that the applicant respond in two stages. The first response from the applicant (Phase I) was to identify the load handling equipment at the Fermi-2 facility within the scope of NUREG-0612 and to describe the associated general load handling operations including safe load paths, procedures, operator training, special and general purpose lifting devices, the maintenance, testing and repair of equipment and the handling equipment specifications. The second response from the applicant (Phase II) was to show that either single failure-proof handling equipment was not needed or that single failure-proof equipment had been provided. In the event the the applicant did not fully meet our guidelines in NUREG-0612, it was to discuss and commit to changes and modifications which would be required in order to fully satisfy the guidelines of NUREG-0612. This supplement contains our evaluation of the applicant's responses to Phases I and II.

9.1.4.2 Evaluation of the Applicant's Responses

We and our consultant, EG&G Idaho, Inc., (EG&G), have reviewed the applicant's submittals on this matter. We have prepared two appendices to this supplement covering the two phases cited above, with the assistance of EG&G. In Appendix N to this supplement, we conclude that our guidelines in Section 5.1.1 of NUREG-0612 for Phase I, have been satisfied. In Appendix P to this supplement, we conclude that our guidelines in Sections 5.1.4, 5.1.5 and 5.1.6 of NUREG-0612 have also been satisfied. Since the completion of Phases I and II more than satisfies the interim protection measures described in Section 5.3 of NUREG-0612 and addresses compliance with ANSI Standard B30.2-1976, we conclude that we no longer require the implementation of interim measures for handling of heavy loads at the Fermi-2 facility prior to licensing and that a Technical Specification addressing ANSI B30.2-1967 is no longer required. On this basis, we find that this matter is now resolved.

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

9.5.1 Fire Protection

In the SER we issued in July 1981, we stated that we had not at that time completed our review of the fire protection for the control room. In Supplement No. 2 of the SER, we stated that, contingent on the completion of certain modifications to the Fermi-2 facility, the applicant's proposed fire protection program was acceptable. Our acceptance included the approval of some requests by the applicant for deviations from our requirements for fire protection. Appendix E to the SER contained our detailed evaluation of the applicant's proposed fire protection program. This was completely superseded by a revised Appendix E in Supplement No. 2 of the SER.

Subsequently, during our Fermi-2 site audit on May 14 through May 18, 1984, of the fire protection provided for safe shutdown equipment, we found that the applicant had not provided fire protection for the control room in accordance with its commitments identified in Supplement No. 2. We also found the separation between redundant, safety-related electrical divisions in the relay room to be unacceptable based on the presence of intervening combustibles. To resolve these issues, we met with the applicant a number of times and the applicant submitted a series of letters extending from August 3, 1984, to March 4, 1985. Our detailed evaluation of the applicant's revised fire protection program is contained in Appendix E to this supplement; this latest Appendix E supersedes the two previous versions except as noted.

The applicant also submitted a number of additional deviation requests which we have reviewed and approved. These deviation requests are identified in Section X of Appendix E to this supplement.

Based on our review of the additional information submitted by the applicant, we find the applicant's proposed fire protection program with the deviations we have approved in Appendix E to this supplement, is in conformance with our guidelines in Appendix A to Branch Technical Position ASB 9.5-1, Appendix R to 10 CFR Part 50 and General Design Criterion 3 of Appendix A to 10 CFR Part 50. On this basis, we find the Fermi-2 fire protection program to be acceptable. We will condition the Fermi-2 operating license to include the commitments made by the applicant.

9.5.7 Emergency Diesel Engine Lubricating Oil System

In the SER we issued in July 1981, we found that the emergency diesel engine lubricating oil system met the requirements of the applicable General Design Criteria (Appendix A to 10 CFR Part 50) and also met the guidance in the applicable regulatory guides and in the Standard Review Plan (NUREG-0800). Our acceptance was subject to the installation of acceptable modifications to prevent dry starting of the diesels. In Supplement No. 2 to the SER, we evaluated modifications to the lube oil system proposed by the applicant. These modifications consisted of providing for partial filling of the upper lube oil supply header and a lube oil booster/accumulator system which forces lube oil into the upper lube oil header during starting. We stated our finding in Supplement No. 2 that the proposed modifications cited above were acceptable.

As discussed in the SER, the standby ac power system for the Fermi-2 facility consists of four emergency diesel-generators (EDGs) identified as Nos. 11, 12, 13 and 14. These units are Model 3800 T6 8-1/8, manufactured by the Fairbanks-Morse Engine Division of Colt Industries. Each unit is a 12 cylinder, opposed piston diesel rated for 3967 horsepower at 900 revolutions per minute.

On January 10, 1985, while conducting a 24-hour surveillance test on EDG No. 11, two trips occurred almost simultaneously, 14 minutes into the test run. While running at full rated load (i.e., 2850 kW), the engine tripped on a signal indicating low lubricating oil pressure, followed a few milliseconds later by a high crankcase pressure signal. Initial diagnosis by the applicant, its consultants, Failure Analysis Associates (FaAA), vendor representative(s) and members of the NRC staff revealed an abnormally low differential pressure across the oil filter and an abnormally high differential pressure across the oil strainer downstream of the oil filter. These indications suggested that the filter was plugged. Subsequent inspection by the applicant revealed a large amount of metallic debris and filings in the lube oil filter and strainer. A complete inspection of EDG No. 11 revealed that the upper crankshaft and main bearings 1 through 7, connecting rod bearings 1 through 7, pistons 2, 3 and 4 and the thrust bearing were damaged.

Based on the extensive damage found on EDG No. 11, the applicant inspected EDG Nos. 12, 13 and 14. In the inspection of the EDG No. 12 lube oil filter, bearing material was found. The internal inspection of EDG No. 12 also revealed damage to the upper main crankshaft bearing shells 1, 2, 3, 7 and 8 and one connecting rod bearing shell showed some initial sign of distress; i.e., distinct indications of bearing wear. The other components inspected were the upper crankshaft, the remaining bearing shells, the pistons, the connecting rods, the upper crank bearing saddles, and the fuel injection camshaft lobes. All these components appeared normal and displayed no symptoms of distress. The damaged and distressed parts for both EDG Nos. 11 and 12 have been replaced, the units reassembled and run in accordance with the break in tests recommended by the vendor and through surveillance tests in accordance with the Fermi-2 Technical Specifications.

Inspection of the EDG Nos. 13 and 14 lube oil filters, the upper crankshaft main and connecting rod bearings 1 through 6, revealed no evidence of distress. EDG Nos. 13 and 14 have been reassembled and gone through the required surveillance testing and declared operable by the applicant; FaAA and the vendor agree with this statement.

Several months prior to the above incident, the applicant, working with the manufacturer, took action to improve the reliability of the Fermi-2 EDGs by extensive piping and subsystem modifications to reduce the amount of draindown and lube oil piping system voids when the engines are on standby service. An oil booster/accumulator was also added which uses starting air to inject oil into the upper crankline during the starting cycle.

These modifications were incorporated to help prevent dry starts as discussed in Supplement No. 2 to the SER. Based on a recommendation by the vendor, the applicant deleted the requirement for operating the manual prelube system after installation of the modifications cited above. Following this vendor recommendation and since January 1984, all planned starts at the Fermi-2 facility have been conducted relying only on the booster prelube system whereas other nuclear plants with Fairbanks-Morse diesels continued to manually prelube their engines prior to planned starts, even after the oil booster/accumulator modification was installed.

The applicant compared the operating procedures at the Fermi-2 facility with those at other nuclear plants with Fairbanks-Morse opposed piston engines. This survey shows that every nuclear power plant except Fermi-2, manually prelubricates the entire engine between 30 seconds to 3 minutes before every planned start of a diesel.

The Fermi-2 facility is unique in that it is the only plant to rely on the keep-warm oil circulating pump to prelubricate the lower crankshaft and a starting air actuated, oil booster cylinder (in the booster tank) to deliver about 1.25 gallons of oil to the upper crankline during the first 3 seconds of the engine start sequence. Because of the problems at the Fermi-2 facility and the absence of similar problems at other nuclear plants which do not rely on the oil booster system for prelubrication of the upper crankline, and the fact that an analysis of the oil booster system effectiveness to lubricate the upper crankshaft and bearings was never demonstrated by Fairbanks-Morse during its development, we were concerned about the adequacy of the oil booster system to perform its intended function. The applicant requested its consultant, FaAA, to perform an analysis of the oil passages and the void volume of the piping system which must be filled before lube oil can reach the bearings. The results of this analysis by FaAA indicates that the total volume to be filled is in excess of the 1.2 gallon supply from the oil booster system. The inadequacy of the oil booster prelube system was further confirmed by a comparison of the 12 upper crankshaft main and connecting rod bearings on EDG No. 11. Based on an examination of these parts and the fact that the bearings closest to the oil supply were relatively undamaged while those furthest from the oil booster tank were badly damaged, it is believed that the failures occurred as a result of the bearing damage sustained during earlier operation which in turn was a result of poor oil distribution during starting sequences of the engine prior to January 10, 1985.

The applicant and FaAA analyzed the operating experience data, the engine starting and operating procedures and the inspection results. They concluded that the upper crankline component failures of EDG Nos. 11 and 12 were the result of inadequate lubrication resulting from not using the installed manual prelubrication system. (As noted previously, the applicant relied on the inadequate oil booster/accumulator system for prelubing the upper crankline.) Examination of the upper and lower crankshafts in EDG No. 11 suggested that possible engine-wide causes such as crankshaft misalignment, contaminated oil, a cavitating oil pump or an overload exceeding the bearing capacity, probably did not cause the observed damage. Since the upper and lower crankshaft bearings are interchangeable, a lack of observable distress on the lower crankshaft bearings indicates that manufacturing defects in the bearing shells did not contribute to the failure.

To avoid any future recurrence of similar damage to the emergency diesels, the applicant has committed in its letters dated March 6, March 14, and March 15, 1985, to the following course of action:

(a) Revise its procedures to require a manual prelubrication period of about two to three minutes prior to any planned start of the EDGs and to continue manual prelubrication until such time as the generator reaches synchronous speed. After the EDGs are modified to permit slow idle speeds, securing of the prelube pump will be allowed when the engine reaches idle speed. With the engine operating at idle speed after a reasonable period of time, the engine speed will be increased to the operating speed and the unit will be synchronized to the offsite power system once the generator attains the proper voltage and frequency. The applicant will then step load the EDG to full load over a period of time and maintain full load for one hour. The applicant will then step unload the machine over a reasonable time period and shut down the EDG. This will be done in accordance with the vendor's instructions.

(b) Visually inspect and replace the oil filter once per calendar quarter. If bearing material is found in the filter, the bearing to crankshaft clearance will be checked on each upper crankshaft bearing. The clearances will also be checked either every 18 months or after 20 starts without prelubrication (i.e., an unplanned start), whichever occurs first.

(c) Analyze oil samples including an analysis for any metallics, on a monthly basis for trend determination. The oil analysis will include a determination of the pentane insolubles and the total base number.

(d) Perform a spectrographic analysis of the lube oil filter media and any deposits which are found during the quarterly replacement. (The spectrographic analysis is considered to be an acceptable means of predicting excessive bearing wear.)

The commitments cited above will be made a part of the Fermi-2 operating license conditions.

The applicant also proposed Technical Specifications in its letter dated March 9, 1985, to ensure the operability of the Fermi-2 EDGs. We find these acceptable.

Because of our concerns regarding the adequacy of the oil booster/accumulator system, we believe that in its present design configuration, little benefit is gained from its use to prevent dry starting of the upper crankline. However, the oil booster system can contribute in a positive manner to the engine prelube requirements during diesel starts (either planned or unplanned) if it can provide an adequate amount of oil to all upper crankline bearings within sufficient time to prevent bearing damage. With this objective in mind and considering the resultant potential increase in the EDG reliability and availability, we recommend that the applicant study whether it should increase the reservoir tank volume to some appropriate volume larger than 1.25 gallons so that it will provide a proper amount of lube oil, adequately distributed, and in the shortest possible time. The results of such a study should be submitted to us. In the event the results of the study indicate that it would be desirable to increase the oil booster/accumulater system tank capacity, it may be necessary to relocate the larger tank(s) outside the engine casing. In this event, it would be necessary to maintain the lube oil temperature in this modified booster lube oil system commensurate with the present keep-warm system temperature. If the study results indicate that an augmented system will ensure timely distribution of an adequate amount of lube oil to the upper crankline, we will review the enhanced Fermi-2 lube oil licensing conditions to determine if they are still necessary.

In summary, the component failures which resulted in the inoperability of EDG No. 11 were restricted to the upper crankshaft, the pistons, the piston pins, the connecting rods and the bearings. We have determined that the Fermi-2 facility failed to properly lubricate these parts due to the inadequate capacity of the oil booster/accumulator lubrication system and the procedures recommended by the vendor which did not specify the need to prelube the upper crankshaft with the main prelubrication system prior to start of a surveillance test. While we cannot conclude with certainty that this was the sole contributor to the observed failures, we believe that inadequate prelubrication of the upper crankline has been clearly established to be the primary contributor. No other potential contributing factors have been identified. What is known with certainty is which parts displayed distress and/or failed. To assure the required future availability of the Fermi-2 EDGs, we and the applicant have agreed to a comprehensive program which will provide assurance of adequate prelubrication of the upper crankline and an early indication of any degradation of the EDGs. On this basis, we find there is reasonable assurance that similar damage will not occur.

Based on our review of this matter, we conclude that EDG Nos. 11, 12, 13 and 14 are ready for full power operation. We further conclude that based upon the comprehensive monitoring program proposed by the applicant, there is reasonable assurance that the Fermi-2 EDGs will provide satisfactory standby power for all design basis events and thus satisfy General Design Criterion 17 of Appendix A to 10 CFR Part 50. We will condition the Fermi-2 operating license to include the commitments made by the applicant on this matter.

11 RADIOACTIVE WASTE MANAGEMENT

11.2 Radioactive Waste Treatment System Description and Evaluation

11.2.1 Liquid Radwaste Treatment System

In the SER we issued in July 1981, we concluded that the liquid radwaste system proposed by the applicant was acceptable. Subsequently, applicant proposed a modification of the Fermi-2 liquid radwaste system. We stated in Supplement No. 1 to the SER that we would provide our evaluation of this modification in a future supplement to the SER. We reported the results of our review of the proposed modification to the liquid radwaste system in Supplement No. 3 to the SER and concluded in this supplement that this proposed modification was acceptable.

The applicant states in Amendment 60 to its FSAR that if the permanent liquid radwaste processing system is not available at the time of initial fuel load, a vendor-supplied and vendor-operated portable system will be used which will be closely monitored by the applicant. This portable system operates by passing contaminated water through a series of pressure vessels containing either filtration media or ion-exchange resins. When these vessels are removed from service, the media are dewatered in-situ or solidified prior to shipping for disposal.

The applicant recently submitted additional information regarding this portable liquid radwaste system in its letters dated February 18 and February 27, 1985. We have evaluated this additional information against our acceptance criteria in Section 11.2 of the Standard Review Plan (NUREG-0800). Based on this review, we find the proposed portable system to be acceptable with two exceptions. These are that the applicant has not performed a detailed cost-benefit analysis of this portable system as required by Section II.D of Appendix I of 10 CFR Part 50 nor has the applicant demonstrated that the design of the proposed portable system satisfies the design objective doses specified in the option provided by the Commission's Annex to Appendix I, dated Septembe 4, 1985. However, we conclude that for reactor operation at power levels up to five percent of full power, the design of the portable liquid radwaste system satisfies the design objectives specified in the option cited above. Moreover, the applicant has committed to make its permanent liquid radwaste treatment system operable prior to exceeding five percent of full power and has agreed to a license condition which reflects this commitment.

We find that the incorporation of this condition into the Fermi-2 operating license will provide reasonable assurance that public health and safety will not be endangered since the permanent liquid radwaste system will be operable before any significant quantity of liquid radwaste can be generated. On this basis, we find that the use of the portable liquid radwaste system prior to exceeding five percent of full power, is acceptable. We will confirm the operability of the Fermi-2 permanent liquid radwaste system in a future supplement to the SER.

11.2.3 Solid Radioactive Waste Treatment System

In the SER we issued in July 1981, we concluded that the solid radwaste system proposed by the applicant was acceptable. Subsequently, the applicant proposed in Amendment 38 to its FSAR, significant changes in the design of the solid radwaste system. We stated in Supplement No. 1 to the SER that we would provide our evaluation of these proposed changes in a future supplement to the SER. We reported the results of our review of the proposed changes to the solid radwaste system in Supplement No. 3 to the SER and concluded in this supplement that these proposed changes were acceptable.

The applicant states in Amendment 60 to its FSAR that if the permanent solidification system of the Fermi-2 facility is not available at the time of initial fuel load, a vendor-supplied and vendor-operated portable solidification system will be used which will be closely monitored by the applicant. The proposed portable solid radwaste system is described in the vendor's topical report, NUS Topical Report PS-53-00378, submitted to us by the applicant and includes a process control program.

The applicant recently has submitted additional information on the portable solid radwaste system in its letters dated February 18 and February 27, 1985. We have reviewed the dewatering process described in these letters against our acceptance criteria in Section 11.4 of the Standard Review Plan (SRP) and found it to be acceptable. However, we have not completed our review of the vendor's licensing topical report cited above. Our review of this topical report is scheduled for completion by April 30, 1985. Based on our review to date, we tentatively conclude that the proposed portable solid radwaste treatment system is acceptable since it appears to be in accordance with our acceptance criteria in Section 11.4 of the SRP. The Fermi-2 Technical Specifications require the applicant to solidify or dewater radioactive wastes in accordance with the process control program(s) which we approve. On this basis, we find that there is reasonable assurance that public health and safety will not be endangered since our review of the proposed process control program(s) and the topical report cited above will be completed prior to issuance of a full power operating license. Since there will not be any significant amount of solid radwaste generated prior to exceeding five percent of full power, we find the proposed portable solid radwaste system to be acceptable for issuance of the low-power license. We will report our evaluation of this portable solid radwaste system in a future supplement to the SER.

13 CONDUCT OF OPERATIONS

13.1 Organizational Structure of Applicant

C. Plant Staff Organization

In the SER we issued in July 1981, we stated that there was inadequate BWR operating experience among key personnel on the plant staff. We further stated that some key personnel should have extensive commercial BWR operating experience and that this experience should be available to the operating groups and to higher levels of plant management. Specifically, we stated our position that there be at least one person on each operating shift who has commercial BWR operating experience, at least up to attainment of 100 percent power operation. This individual could either be an employee of DECO or be on contract to the applicant. In response to our various requirements on this matter, the applicant submitted a series of letters dated June 3, June 15 and June 18, 1981, in which it provided appropriate commitments which we found acceptable.

The applicant has submitted additional information in its letter dated June 18, 1984, regarding the available operating experience on shift, including its plan to train and use shift advisors. We have reviewed this program for conformance to the guidelines for shift advisors proposed by a number of nuclear utilities; this program was accepted by the Commission with some clarifications. We have evaluated the applicant's proposed program in light of this modified proposal and similar programs at other nuclear utilities.

The industry proposal to the Commission cited above, was first made on February 24, 1984, by an Industry Working Group representing utilities that had nuclear plants under construction or ready for operation. The proposal was focused on the amount of previous operating experience considered to be the minimum desirable on each shift and how that experience could be obtained. On June 14, 1984, the Commission accepted this industry proposal with certain clarifications. Information regarding the Commission action was published in Generic Letter 84-16, dated June 27, 1984. The basic objective of the Commission is that, at time of fuel load, each operating shift will have at least one senior operator who has a minimum of six months of previous hot operating experience including at least six weeks above 20 percent power and including startup and shutdown experience. However, for those plants in the later stages of licensing for which there is insufficient time to provide adequate hot experience for plant operating personnel, the Commission will accept the use of experienced advisors on each of the operating shifts. The minimum acceptable qualifications for these shift advisors are four years of power plant experience, including two years of nuclear plant experience, with a minimum of one year operating experience as a licensed senior operator or a suitably qualified operator on a large, commercial nuclear plant of the same type. These advisors are to be trained on the systems, procedures and technical specifications of the plant for which they are to provide advice, and certified to the NRC as being qualified to act as Shift Advisors. The Fermi-2 facility falls within that group of plants eligible to use advisors to provide experienced advice to the operating shifts.

The applicant has provided resumes of five Shift Advisors. Based on our review of their resumes, we conclude that each of the Shift Advisors meets the criteria established by the Commission on this matter. Moreover, these five Shift Advisors were administered written and oral examinations and subsequently received their SRO licenses for Fermi-2.

The duties and responsibilities of the Shift Advisors are contained in a draft copy of the applicant's Administrative Operating Procedure 21.000.01, Rev. 8. We have reviewed this Shift Advisor procedure and, with several exceptions, find it acceptable. Specifically, it is our position that the applicant add the following items to the list of duties and responsibilities for the Shift Advisor contained in Enclosure 7 of Procedure 21.000.01:

a. When necessary, recommend suspension of plant evolutions or activities and, if required, recommend a plant shutdown.

b. If any disagreement between the operating shift personnel and the Shift Advisor arise, resolution will be made by the Operations Engineer.

These items reflect the Industry Working Group proposal on the responsibilities and duties of Shift Advisors as accepted by the Commission. We will confirm implementation of these two items in a future supplement to the SER.

The applicant has also provided an outline of its Shift Advisor Training Program which was conducted over a 17-week period and included all the elements contained in Generic Letter 84-16. Since these Shift Advisors are now licensed personnel, they are required to participate in the applicant's Requalification Program. This exceeds the training and evaluation guidelines for Shift Advisors.

Additionally, the applicant plans to develop a videotape and a procedure review to train the operating shift crews on the role of the Shift Advisor when the applicable administrative procedure has been formally approved.

The medical qualifications of the Shift Advisors have been verified by the medical evaluations contained in their applications for SRO licenses. The applicant has also provided a performance appraisal form for the Shift Advisors. The applicant appraisal period is on a three-month cycle. It is our position that the applicant conduct monthly evaluations for at least the first six months of the Shift Advisor Program. We will confirm the implementation of this item in a future supplement to the SER.

Based on our review of the applicant's submittal of June 18, 1984, on the matter of experienced operators, we conclude that there will be sufficient operating experience available on each shift to meet the guidelines contained in Generic Letter, 84-16 and in the Commission's letter on this matter. We also conclude that the use of Shift Advisors to augment the experience of the applicant's operating shifts is acceptable. Accordingly, we find that the applicant has satisfied its prior commitments regarding on-shift operating experience in an acceptable manner. We find this matter is now resolved subject to confirmation of those items cited above.

However, in order to provide assurance that the required on-shift operating experience will be maintained in an acceptable manner after issuance of the OL, we will condition the operating license as follows:

> At all times the plant is in an operating condition other than cold shutdown or refueling, the Detroit Edison Company (DECO) shall have a licensed senior operator on each shift who has had at least six months of hot operating experience on a similar type plant, including at least six weeks at power levels greater than 20 percent of full power, and who has had start-up and shutdown experience. For those shifts where such an individual is not available on the plant staff, DECO shall provide an advisor who has had at least four years of power plant experience, including two years of nuclear plant experience, and who has had at least one year of experience on shift as a licensed senior operator at a similar type facility. Use of advisors who were licensed only at the reactor operator level or who otherwise do not fully meet the criteria for a shift advisor, will be evaluated on a case-by-case basis. As a minimum, DECO shall train these advisors on the Fermi-2 procedures, technical specifications and plant systems, and shall examine them on these topics at a level sufficient to assure familiarity with the plant. For each shift, the remainder of the shift crew shall be trained in the role of the advisors. The training of the advisors and the remainder of the shift crew shall be completed prior to achieving initial criticality. Prior to achieving criticality, DECO shall certify to the NRC staff the names of the advisors who have been examined and have been determined to be competent to provide advice to the operating shifts. These advisors, or suitably qualified replacements, shall be retained until at least one of the senior operators on each shift has the required experience. The NRC staff shall be notified at least 30 days prior to the release of any special assigned advisor who has been provided in accordance with this license condition.

13.3 Emergency Preparedness Evaluation

13.3.1 Introduction

In Supplement No. 4 to the SER, we identified several emergency planning areas where the applicant had committed to provide additional information. Our evaluation in Supplement No. 4 was based primarily on Revision 2 to the Fermi-2 Radiological Emergency Response Plan (the emergency plan), dated September 1983. Subsequently, the applicant submitted Revision 3A to the Fermi-2 emergency plan in August 1984. This supplement presents in Section 13.3.2, the results of our review of Revision 3A to the emergency plan and the additional submittals which provided information pertaining to the applicant's commitments identified in Supplement No. 4. An evaluation of certain other emergency preparedness subject areas for the Fermi-2 facility are also included in Section 13.3.2 of this report.

The initial findings and determinations on the adequacy of offsite emergency preparedness for the Fermi-2 facility provided by the Federal Emergency Management Agency (FEMA), were presented in Supplement No. 4. In that supplement, we indicated that additional findings had been requested from FEMA. These findings were provided to us by FEMA in its letter dated May 14, 1984, and are presented in Section 13.3.4 of this supplement. Our conclusions on adequacy of the emergency preparedness for the Fermi-2 facility is presented in Section 13.3.5 of this supplement.

13.3.2 Evaluation of the Emergency Plan

This evaluation addresses those confirmatory items identified in Supplement No. 4 to the SER for which the applicant provided additional information. We also provide an update of certain other areas related to emergency preparedness for the Fermi-2 facility. The order of presentation and the numbering of the sections corresponds to the listing of these items in Section 13.3 of Supplement No. 3.

13.3.2.1 Assignment of Responsibility (Organizational Control)

The applicant committed to revise its emergency plan to identify the organizations with responsibilities in the ingestion exposure pathway (i.e., within 50 miles of the Fermi-2 facility) Emergency Planning Zone (EPZ), and to provide a map showing the ingestion exposure EPZ. Revision 3A to the plan indicates that the States of Michigan and Ohio and the Province of Ontario, Canada are the lead governmental organizations within the ingestion exposure EPZ. Figure A-2 of the plan is a map showing the 50-mile radius ingestion exposure EPZ. Based on a review of Revision 3A to the plan, we confirm that the applicant has complied with this commitment.

13.3.2.2 Onsite Emergency Organization

In Supplement No. 4 to the SER, we noted that the applicant intended to comply with the 30-minute and 60-minute staffing augmentation goals of Table 2 in Supplement 1 to NUREG-0737. (These goals are also presented in Table B-1 of NUREG-0654.) However, we also noted that the applicant stated in Revision 2 to its emergency plan that during off-hours, 60 minutes on the average, is required to staff key emergency preparedness response positions. As a result of our findings made during the onsite emergency preparedness appraisal conducted by the NRC on October 11-21, 1983, and reported in Inspection Report No. 50-341/83-24, dated November 28, 1983, the applicant revised the minimum number of staff augmentation personnel for emergencies as shown in Table B-1 of Revision 3A to its emergency plan to more closely conform to the staffing goals of Supplement 1 to NUREG-0737.

We determined in a followup inspection during the week of December 3, 1984, that the applicant had satisfactorily demonstrated in a drill on October 10, 1984, that the staffing augmentation goals could be met. The plan will be revised to state that during off-hours, under normal conditions, key emergency response positions can be staffed within 30 minutes the majority of the time. However, the applicant states that there may be some conditions where up to 60 minutes may be required to meet the staffing augmentation goals. We find that the applicant has made reasonable progress toward meeting the staffing augmentation objectives of Supplement 1 to NUREG-0737 and that the applicant's staffing for the initial facility response and timely augmentation is acceptable.

13.3.2.4 Emergency Classification System

In Supplement No. 4 to the SER, we identified the applicant's commitment to revise the emergency plan implementing procedures (EPIPs) to: (1) incorporate a number of emergency action levels (EALs) which were under development; (2) include appropriate information on the correlation between the containment high range radiation monitors (CHRRMs) and core damage source terms; and (3) incorporate the methodology for assessing a radiological emergency in the event that key monitoring instrumentation is either offscale or inoperable. We have confimed that the applicant has complied with these commitments in a review of the following procedures:

(a) EP-101, "Classification of Emergencies," Revision 1, dated November 21, 1984.

(b) EP-546, "Calculation of Estimated Containment High Range Radiation Monitor or SGTS/AXM Monitor Readings If Instruments are Inoperable or Offscale," Revision 0, November 9, 1984.

(c) EP-547, "Rapid Estimate of Core/Final Damage Based on Containment High Range Radiation Monitor," Revision 0, November 9, 1984.

13.3.2.5 Notification Methods and Procedures

In Supplement No. 4 to the SER, we indicated that the applicant had committed to revise EP-290, "Emergency Notifications," regarding clarification of the State of Michigan notification forms to be used for initial and follow-up messages to offsite authorities. We have reviewed Revision 1 to EP-290 and confirmed that the applicant has complied with this commitment.

We also identified in Supplement No. 4 that the prompt alert and notification system was not fully functional in that the control panel at the Monroe City-County Joint Communications Center had not been installed. Both the NRC, in a followup inspection during the week of December 3, 1984, and FEMA have confirmed that this system is installed and operational. In its letter to the NRC dated November 28, 1984, FEMA reported that an analysis of the prompt alert and notification system for the Fermi-2 facility has been completed and FEMA has determined that there is reasonable assurance that the system is adequate to promptly alert and notify the public in the event of an accident at the Fermi-2 facility. (Refer to Section 13.3.4 of this supplement.)

In Supplement No. 4, we referred to a request made by the applicant to Monroe County for more explicit information in the County plan regarding the County's responsibility for prompt decision-making and public notification during a rapidly moving event. In its letter dated December 17, 1984, the applicant provided an excerpt from the revised Monroe County emergency plan dated October 1984, which states in the Basic Plan, Section V, under Concept of Operations, that in the event of a nuclear incident where offsite releases have occurred, or when there is the imminent threat thereof, the Chairperson of the Board of Commissioners will declare a state of emergency thereby fully activating the

Monroe County plan. This excerpt and other information in the County plan indicates that local officials have the responsibility and authority to take protective measures for the public in the plume exposure EPZ based on recommendations from plant operators even in situations where releases have not yet occurred.

We find that the applicant has provided a satisfactory response for this item. However, we recommend that the applicant continue to coordinate planning efforts with offsite authorities to ensure that the necessary procedures are in place with designated alternates for key positons and response time criteria for County officials to promptly alert and notify the public in the event of a situation requiring urgent action.

13.3.2.7 Public Information

We stated in Supplement No. 4 that the public information brochure had been distributed to the general public prior to the radiological emergency exercise held at the Fermi-2 facility in February 1982 and was scheduled to be redistributed prior to the June 26, 1984, full-participation exercise. We have confirmed during a followup inspection on December 5, 1984, that the Fermi-2 public information brochure was redistributed to the public in the plume exposure pathway EPZ prior to the June 1984 exercise.

13.3.2.8 Emergency Facilities and Equipment

In Supplement No. 4, we concluded that, on an interim basis, the emergency response facilities (i.e., the EOF, the TSC, and the OSC) were adequate to support a response effort in the event of an emergency. We further stated, based on information provided by the applicant, that the emergency response facilities (ERFs) would be fully functional by September 1984 with the installation of the emergency response information system (ERIS). ERIS is an automated data acquisition system which provides data for the safety parameter display system (SPDS) and the dose assessment function. In its letter to the NRC dated November 12, 1984, the applicant informed the staff that ERIS/SPDS would not be fully functional until December 1985. This schedule is a matter to be negotiated with the NRC in accordance with Supplement 1 to NUREG-0737.

Observations which we made during the full-participation exercise on June 26, 1984, confirmed our previous findings in Supplement No. 4 that the ERFs are adequate to support a response effort and that the applicant's annual dose assessment capability is acceptable as an interim methodology until ERIS is fully functional. As indicated in Supplement No. 4, we will conduct a post-implementation appraisal of the applicant's ERFs including ERIS/SPDS in accordance with Supplement 1 to NUREG-0737 on the schedule developed between the applicant and the NRC.

In Supplement No. 3, we made reference to a short-term meteorological study being conducted by the applicant to characterize the possible effects of Lake Erie on plume transport from the Fermi-2 facility. (This phenomenon is referred to as the lake breeze effect.) We had requested the applicant to provide the results of this study and to revise its meteorological model which is used to determine offsite dose projections in emergency situations if the lake breeze effects were significant. A potential atmospheric occurrence at a lakeside location such as the Fermi-2 site is the development of a thermal internal boundary layer (TIBL) on shore. The presence of a TIBL can affect the diffusion and transport of gaseous effluents released at lakeside locations. In an attempt to define the characteristics of the TIBL at the Fermi-2 site, the applicant compared the results of its short-term TIBL measurements with three empirical models.

The results of this study were submitted by the applicant in its letter dated May 8, 1984. This study demonstrated that none of the three models adequately predicted the TIBL characteristics for the Fermi-2 site. The applicant has proposed to continue to study the relationship between onsite meteorological measurements and the TIBL in an effort to develop a correlation which could be utilized in its dose projection model. We agree with the applicant's findings in its study and support the applicant's continued assessment of the lake-breeze effect at the Fermi-2 site.

As an interim approach, the applicant stated in its letter dated December 17, 1984, that emergency plan implementing procedures EP-544, "Meteorological Data Assessment," and EP-545, "Protective Action Recommendation Guidelines," will be revised to include: (1) criteria to account for the occurrence of lake breeze effects; and (2) associated protective actions should a lake breeze effect occur during an emergency at the Fermi-2 site. (The conditions when this phenomenon can significantly affect the offsite dose projection are during daylight hours from April through October for stability classes A, B, or C with the wind direction coming from 57° through 168°.) We find this approach to be acceptable and will confirm that the applicant has complied with this commitment during a followup inspection of the emergency preparedness program.

13.3.2.10 Protective Response

We stated in Supplement No. 4 that the applicant had incorporated predetermined protective action recommendations based on plant conditions into emergency plan implementing procedure EP-545, "Protective Action Guideline Recommendations." However, we also noted that the Fermi-2 emergency plan indicated that protective actions were based only on dose projections. The applicant committed to revise its plan to reflect the fact that protective actions should be based on plant conditions as well as on dose projections. Table J-1 of Revision 3A of the emergency plan, which is a schematic diagram of the process to be used for developing offsite protective action recommendations, clearly indicates that such recommendations will be based on plant and reactor core conditions in addition to dose projections.

In its letter dated December 17, 1984, the applicant informed us that the description of protective response in the emergency plan would be further revised. This additional revision will clarify that protective action recommendations will be based on plant conditions as well as projected offsite doses. We find that the applicant has provided a satisfactory response to this item.

We recommended in Supplement No. 4 that the applicant include information regarding special facility population in emergency plan implementing procedure EP-545, "Protective Action Guideline Recommendations." Our review of EP-545, Revision 0, has established that this procedure includes maps showing the location of schools, hospitals and nursing homes within 10 miles of the Fermi-2 site and that the maps also indicate the number of students enrolled in each school and the number of patients in each institution. EP-545 includes a table giving the evacuation time estimates for the evacuation of the special population segment of the total population by distance and evacuation zone for both normal and adverse weather condition. We find that EP-545 satisfactorily conforms to our recommendation on this matter in Supplement No. 4.

13.3.2.15 Radiological Emergency Response Training

In Supplement No. 4, we indicated that the Fermi-2 emergency plan would be revised by the applicant to include a description of its training program for offsite emergency response personnel. We have confirmed that Revision 3A of the plan includes in Section 0.2, information on the applicant's training program for offsite support organizations. Specifically, the applicant provides training for certain support groups which respond directly to the site such as ambulance service and fire fighting, and participates in a joint State, county and utility training program for other local offsite support organizations. The Fermi-2 emergency plan indicates that this training is given on an annual basis. We find that the applicant's revision to its emergency plan concerning training for offsite response personnel is acceptable.

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

FEMA's interim findings on offsite emergency plans and preparedness were provided in Supplement No. 4 to the SER. 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 on February 2, 1982, FEMA reported that an adequate level of offsite emergency preparedness existed for the Fermi-2 facility. We indicated in Supplement No. 4 that we had requested further FEMA support in reviewing the revised emergency plan for Monroe County (a draft which was dated December 1983) and a separate plan developed for Brownstown Township in Wayne County. In a supplemental, interim finding report provided to the NRC on May 14, 1984, FEMA reported that based on its review of the revised Monroe County and Brownstown Township radiological emergency plans, there is reasonable assurance that the plans are adequate and capable of being implemented in the event of an accident at the Fermi-2 site.

Subsequently, FEMA provided a report to the NRC dated October 15, 1984, on the emergency preparedness exercise conducted at the Fermi-2 site on June 26, 1984. This was a full-participation exercise for the State of Michigan, Monroe and Wayne Counties, and Brownstown Township. No significant deficiencies in offsite preparedness were identified by FEMA. The exercise report listed several lesser inadequacies in the offsite plans and FEMA Region V has requested the State of Michigan to develop a schedule of corrective actions to address these items.

In its letter to the NRC dated November 28, 1984, FEMA reported that an analysis of the prompt alert and notification system for the Fermi-2 facility has been completed pursuant to FEMA rule 44 CFR 350. FEMA has determined that this system meets the specific design requirements of NUREG-0654/FEMA-REP-1, Revision 1, and FEMA-43 and that there is reasonable assurance that the system is adequate to promptly alert and notify the public in the event of an accident at the Fermi-2 plant.

On December 11, 1984, the Monroe County Board of Commissioners formally adopted the Monroe County emergency plan dated October 1984. This plan will be forwarded by the State of Michigan to FEMA for formal review and administrative approval pursuant to 44 CFR 350 of FEMA's rules. As indicated above, FEMA has provided the NRC with an interim finding of plan adequacy for Monroe County based on a review of the draft plan dated December 1983. A preliminary examination by us of the October 1984 plan indicates that the concept of operations for emergency responses for Monroe County remains essentially the same compared to the earlier draft plan. However, we have requested FEMA to review the October 1984 plan officially adopted by Monroe County to confirm that: (1) no substantive changes have been made from the draft County plan dated December 1983 which formed the basis for the FEMA finding of adequacy; and (2) offsite emergency planning for Monroe County remains adequate to support an emergency response at the Fermi-2 site. We will confirm this matter in a future supplement to the SER prior to authorization for operation of the Fermi-2 facility above five percent of rated power.

13.3.5 Interim Conclusions

Based on our review of Revision 3A of the Fermi-2 Radiological Emergency Response Plan and additional information submitted by the applicant, we find that those items previously identified in Supplements 3 and 4 to the SER as requiring additional information or confirmation, have been satisfactorily addressed. We conclude that the level of emergency planning and preparedness for the Fermi-2 site provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency which might occur at the Fermi-2 facility during fuel loading and low-power operations (i.e., up to five percent of rated power).

The Federal Emergency Management Agency (FEMA) has provided findings and determinations which substantiate that an adequate level of offsite emergency planning and preparedness exists for the Fermi-2 facility. We have requested confirmation from FEMA that the revised Monroe County emergency plan dated October 1984 continues to support FEMA's finding of plan adequacy for Monroe County which was based on a review of a draft plan dated December 1983. Upon receipt of this supplemental finding from FEMA, we will provide our conclusion on the overall state of onsite and offsite emergency preparedness for Fermi-2 in a future supplement to the SER prior to authorization for operation above five percent of rated power.

13.5 Industrial Security

13.5.1 Introduction

In the SER we issued in July 1981, we stated that subject to certain revisions to the applicant's physical security plans, we concluded that the applicant's proposed plans were acceptable. Subsequently, the applicant submitted Amendment 2 to its Physical Security Plan in response to our concerns on specific items. In Supplement No. 1 to the SER, we stated that we would provide our evaluation of the acceptability of the applicant's revised physical security plan in a future supplement to the SER. We also stated in Supplement No. 1 that we had found the applicant's revised Safeguards Contingency Plan and its Security Force Training and Qualification Plan to be acceptable.

The Fermi-2 security plans originally submitted by the applicant over the last several years are summarized below. They have since been revised and amended:

"Enrico Fermi Atomic Power Plant Unit 2 Physical Security Plan," Revision 4, dated April 1983; (transmitted June 29, 1983); "Enrico Fermi Atomic Power Plant Unit 2, Safeguards Contingency Plan," Revision 1, undated (transmittal letter dated July 20, 1981); and "Enrico Fermi Atomic Power Plant Unit 2, Security Personnel Training and Qualification Plan," Revision 1, undated (transmittal letter dated July 20, 1981).

This supplement summarizes how the applicant proposes to meet the requirements of 10 CFR Part 73. Our evaluation is composed of a basic analysis which is available to the public, and a protected Appendix.

13.5.2 Physical Security Organization

To satisfy the requirements of Section 73.55(b) of 10 CFR Part 73, the applicant has provided a physical security organization which includes a Shift Lieutenant who is onsite at all times and who has the authority to direct the physical protection activities. To implement the commitments made in its physical security plan, its training and qualification plan, and its safeguards contingency plan, the applicant has developed written security procedures specifying the duties of the security organization members. These procedures are available for inspection. The training program and critical security tasks and duties for the security organization personnel are defined in the applicant's report entitled, "Enrico Fermi Security Personnel Qualification and Training Plan." Based on our review of this report, we find that the applicant meets the requirements of Appendix B to 10 CFR Part 73, for the training, equipping and requalification of the security organization members. We also find that the proposed physical security plan and the proposed training program provide commitments which preclude the assignment of any individual to a security-related duty or task prior to that individual being trained, equipped and qualified to perform the assigned duty in accordance with the approved guard training and qualification plan. On this basis, we find the Fermi-2 physical security organization to be acceptable.

13.5.3 Physical Barriers

To meet the requirements of Section 73.55(c) of 10 CFR Part 73, the applicant has provided a protected area barrier which complies with the barrier definition in Section 73.2(f)(1). The applicant has also provided an isolation zone of at least 20 feet on both sides of the barrier, with the exception of the locations listed in the Appendix, to permit observation of activities along the barrier. We have reviewed those locations and determined that the security measures in place are satisfactory and continue to meet the requirements of Section 73.55(c) of 10 CFR Part 73. Additionally, an illumination level of 0.2 foot-candles is maintained for the isolation zones, the protected area barrier and external portions of the protected area. On this basis, we find the Fermi-2 physical barriers to be acceptable.

13.5.4 Identification of Vital Areas

The design bases for the applicant's program for identifying vital equipment includes the regulatory definition of what is vital, release limits as defined in 10 CFR Part 100 and our guidance contained in Regulatory Guide 1.29 and Review Guideline No. 17. The applicant's program uses conservative assumptions (i.e., no credit is taken for the availability of offsite power and equipment which is not protected as vital, is assumed to be unavailable) and a detailed plant analysis including a fault tree, to identify both single items of equipment and combinations thereof which require protection. The Appendix contains a detailed discussion of the applicant's program and identifies those areas and pieces of equipment which the applicant has determined to be vital.

Vital equipment is located within areas designated as vital which in turn are located within the protected area. Passage through at least two barriers, as defined in Sections 73.2(f)(1) and 73.2(f)(2) of 10 CFR Part 73, is required to gain access to vital equipment. Vital area barriers are separated from the protected area barrier.

Both the control room and the central alarm station (CAS) are provided with bullet-resistant walls, doors, ceilings, floors, and windows. Based on our review and on the analysis set forth in paragraph D of the Appendix, we conclude that the applicant's program for identification and protection of vital equipment satisfies the requirements of 10 CFR Part 73. However, this program is subject to onsite validation by the NRC staff in the future, and to subsequent changes if found to be necessary. With this stipulation, we find the identification of vital areas to be acceptable.

13.5.5 Access Requirements

In accordance with Section 73.55(d) of 10 CFR Part 73, all points of personnel and vehicle access to the protected area are controlled. The individual responsible for controlling the final point of access into the protected area is located in a bullet-resistant structure. As part of the access control program, vehicles (except under emergency conditions), personnel, packages, and materials entering the protected area are searched for explosives, firearms and incendiary devices either by electronic search equipment and/or a physical search.

Vehicles admitted to the protected area, except certain vehicles designated by the applicants, are controlled by escorts. The vehicles so designated are limited to on-site functions at the Fermi-2 site and remain in the protected area except for operational maintenance, repair, security and emergency purposes. Positive control over these vehicles is maintained by personnel authorized to use the vehicles or by the escort personnel. A picture badge/key card system, utilizing encoded information, identifies individuals who are authorized unescorted access to protected and vital areas. This badge/card system is used to control access to these areas. Individuals not authorized for unescorted access are issued non-picture badges which indicate that an escort is required. Access authorization is limited to those individuals who have a need for access to perform their duties.

Unoccupied vital areas are locked and alarmed. During periods of refueling or major maintenance, access to the reactor containment(s) is positively controlled by a member of the security organization to assure that only authorized individuals and materials are permitted to enter. In addition, all doors and personnel/equipment hatches into the reactor containment(s) are locked and alarmed. Keys, locks, combinations and related equipment are changed on an annual basis. In addition, when an individual's access authorization has been terminated due to either a lack of reliability or trustworthiness or for poor work performance, the keys, locks, combinations and related equipment to which that person had access, are changed.

13.5.6 Detection Aids

To meet the requirements of Section 73.55(e) of 10 CFR Part 73, the applicant has installed intrusion detection systems at the protected area barrier, at entrances to vital areas and at all emergency exits. Alarms from the intrusion detection system annunciate within the continuously manned central alarm station and a secondary alarm station (SAS) located within the protected area. The central alarm station is located so that the interior of the station is not visible from outside the perimeter of the protected area. In addition, the central alarm station is constructed so that its walls, floors, ceilings, doors and windows are bullet-resistant. The alarm stations are located and designed so that a single act cannot interdict the capability of calling for assistance or responding to alarms. The central alarm station will perform no other functions or duties which would interfere with its alarm response function. The transmission lines of the intrusion detection system and the associated alarm annunciation hardware are self-checking and tamper-indicating. When activated, the alarm annunciators will indicate the type of alarm and its location. An automatic indication of when the alarm system is on standby power is provided in the central alarm station. On this basis, we find the Fermi-2 detection aids to be acceptable.

13.5.7 Communications

As required in Section 73.55(f) of 10 CFR Part 73, the applicant has provided a capability for continuous communications between the central and secondary alarm station operators, guards, watchmen and the armed response personnel through the use of a conventional telephone system and a security radio system. In addition, direct communication with the local law enforcement authorities is maintained through the use of a conventional telephone system and two-way FM radio links. All non-portable communication links, except the conventional telephone system, are provided with an uninterruptable emergency power source. On this basis, we find the Fermi-2 security-related communication provisions to be acceptable.

13.5.8 Test and Maintenance Requirements

To meet the requirements of Section 73.55(g) of 10 CFR Part 73, the applicant has established a program for testing and maintenance of all intrusion alarms, emergency alarms, communication equipment, physical barriers and other security-related devices and equipment. Equipment or devices which do not meet their design performance criteria or have failed to otherwise operate, will be compensated for by appropriate compensatory measures as defined in the report entitled "Enrico Fermi Atomic Power Plant, Unit 2 Security Plan" and in Fermi-2 procedures. The compensatory measures defined in these plans will provide reasonable assurance that the effectiveness of the security system is not reduced by failures or other contingencies affecting the operation of the security-related equipment or structures. Intrusion detection systems are tested for proper performance at the beginning and end of any period in which they will be used for security. Such testing will be conducted at least once every seven days.

Communication systems for onsite communications are tested at the beginning of each security shift. Offsite communication systems are tested at least once each day.

Audits of the security program will be conducted once every 12 months by personnel independent of the site security management and supervision. These audits will focus on the effectiveness of the physical protection provided by the onsite security organization which is implementing the approved security program plans. The audits will include, but will not be limited to: (1) a review of the security procedures and practices; (2) system testing and maintenance programs; and (3) local law enforcement assistance agreements. A report will be prepared documenting these audit findings and any recommendations; the audit report will be submitted to the Fermi-2 plant management.

13.5.9 Response Requirements

To meet the requirements of Section 73.55(h) of 10 CFR Part 73, the applicant will have armed responders immediately available for response duties on all shifts, consistent with the requirements of 10 CFR Part 73. The basis for establishing the number of armed responders is contained in the Appendix. In addition, the applicant has established and documented its liaison with local law enforcement authorities who will provide additional response support in the event of a security event at the Fermi-2 facility.

We find that the applicant's safeguards contingency plan for dealing with thefts, threats and potential radiological sabotage events satisfies the requirements of Appendix C to 10 CFR Part 73. This plan identifies those security events which could initiate a radiological sabotage event and identifies the applicant's preplanning, response resources, safeguards contingency participants and coordination activities for each such identified event. Through this plan, upon the detection of either an abnormal presence or activities within the protected or vital areas, the applicant would initiate its response activities using its available resources. These response activities and objectives include: (1) the neutralization of the existing threat by requiring the response force members to interpose themselves between the potential adversary and their objectives; (2) instructions to use force commensurate with that used by the adversary; and (3) designated authority to request sufficient assistance from the local law enforcement authorities to maintain control over the situation.

To assist in the assessment/response activities, the applicant has installed for its security organization, a closed circuit television system which provides the capability to observe the entire protected area perimeter, isolation zones and a majority of the protected area.

13.5.10 Employee Screening Program

To meet the requirements of Section 73.55(a) of 10 CFR Part 73, to protect against the design basis threat as defined in Section 73.1(a)(1)(ii), the applicant has provided an employee screening program. Personnel who successfully complete the employee screening program or its equivalent, may be granted unescorted access to protected and vital areas at the Fermi-2 site. All other personnel requiring access to the site are escorted by persons authorized and trained for escort duties. These escorts have successfully completed the employee screening program. The employee screening program is based on accepted industry standards and includes a background investigation, a psychological evaluation and a continuing observation program. In addition, the applicant may recognize the screening program of other nuclear utilities or contractors based upon a comparability review conducted by the applicant. The plan also provides for an exclusion (i.e., a "grandfather clause") which allows recognition of a certain period of trustworthy service with either the applicant or its contractors, as being equivalent to the overall employee screening program. We have reviewed the applicant's screening program against the accepted industry standards (i.e., ANSI N18.17 1973) and have determined that the Fermi-2 employee screening program is acceptable.

13.5.11 Summary of Evaluation

Based on our review of the appropriate documents and visits to the site, we conclude that the protection provided by the applicant against radiological sabotage at the Fermi-2 facility meets the requirements of 10 CFR Part 73. Accordingly, the proposed protection at the Fermi-2 site will ensure that the health and safety of the public will not be endangered. On this basis, we find the Fermi-2 industrial security program to be acceptable.

15 SAFETY ANALYSIS

15.2 Accidents

15.2.3 Radiological Consequences of Design Basis Accidents

15.2.3.1 Loss-of-Coolant Accident (Radiological Consideration)

In the SER we issued in July 1981, we concluded in Section 15.2.3.1.B that there was reasonable assurance that the total radiological consequences of a postulated loss-of-coolant accident (LOCA) would be within the exposure guidelines set forth in Section 100.11 of 10 CFR Part 100. Subsequently, the applicant submitted Amendment No. 58 to the FSAR (July 1984) in which it revised its estimate of the time to drawdown the secondary containment to a partial vacuum of negative one-quarter inch (water gauge). (Refer to Section 6.2.3 of the SER.) The applicant estimates that this drawdown time is now about ten minutes rather than its original estimate of six minutes. As a result, we have revised our estimates of the radiological consequences of a postulated LOCA. The following sections summarize our revised evaluation and conclusions.

Our revised estimates are based on the assumption that it takes 560 seconds (9.3 minutes) to drawdown the secondary containment to a negative pressure of 0.25 inches (water gauge) following the postulated LOCA. All other assumptions we made in Section 15.2.3.1 of the SER remain unchanged.

A. Staff Evaluation

1. Containment Leakage Contribution

The change in the drawdown time estimated by the applicant affects only the amount of released unfiltered containment leakage. Because the secondary containment is not maintained at a one-quarter inch negative pressure (water gauge) during the first 560 seconds of the accident, all containment leakage during this period is assumed to be released unfiltered to the environment. Subsequent to this initial 560-second period, the primary containment leakage (other than bypass leakage) is assumed to be processed immediately by the standby gas treatment system (SGTS) filters before being exhausted to the environment.

Our revised calculated doses resulting from the postulated LOCA for the larger containment leakage contribution in the exclusion area and within the low population zone boundaries are presented in Table 15.1.A; this table supercedes Table 15.1 in the SER. The other leakage components, (i.e., the MSIV and ESF leakages) contributing to the calculated doses remain unchanged as do the doses from all other postulated accidents. The only significant changes in doses resulting from a postulated LOCA are: (1) the total dose to the thyroid at the exclusion boundary increases from 150 rem to 185 rem; and (2) the total dose to the thyroid within the low population zone increases from 76.5 rem to 79.4 rem.

B. Staff Conclusions

We have reviewed the applicant's revised analysis and have performed our own independent revised analysis of the radiological consequences of a postulated LOCA from the one leakage path (i.e., the containment leakage) affected by the applicant's revision of the estimated drawdown time. This revised analysis is discussed in Section 15.2.3.A above, and the results are presented in a revised table (i.e., Table 15.1A).

Our prior conclusion on this matter remains unchanged. Namely, the distances at the Fermi-2 site to the exclusion boundary and to the boundaries of the low population zone, in conjunction with the engineered safety features of the Fermi-2 facility, are sufficient to provide reasonable assurance that the total radiological consequences of a postulated LOCA are within the exposure guidelines set forth in Section 100.11 of 10 CFR Part 100. This conclusion is based on our review of the applicant's revised analysis and on our own independent revised analysis, which we performed to verify that the calculated total doses are within the applicable guidelines of 10 CFR Part 100.

Table 15.1A Radiological consequences of design basis accidents

Postulated accident	Exclusion boundary (rem) Thyroid	Exclusion boundary (rem) Whole Body	Low Population zone (rem) Thyroid	Low Population zone (rem) Whole body
Loss of coolant:				
Containment Leakage				
0-2 hours	180	6.1	14.9	.50
2-8 hours			16.1	.52
8-24 hours			11.9	.24
24-96 hours			19.1	.10
96-720 hours			13.2	.04
Total containment leakage*	180	6.0	75.2	1.4
ECCS component leakage	5	0.01	4.2	0.3
Total	185	6.1	79.4	1.7
Steam line break outside secondary containment:				
Normal long-term operation	1.5	<0.1	0.13	<0.1
Normal short-term operation	29.4	<0.1	2.4	<0.1
Control rod drop	0.3	<0.1	0.13	<0.1
Fuel handling	0.6	0.2	<0.1	<0.1
Instrument line break	1.6	<0.1	0.13	<0.1

*The MSIV leakage dose contribution is expected to be very small as discussed in Section 15.2.3.1.A.2 of the SER and, therefore, was not included in this table.

22 TMI-2 REQUIREMENTS

I Operational Safety

I.C Operational Procedures

I.C.7 Nuclear Steam Supply System Vendor Review of Procedures

This matter has been addressed and closed out in Inspection Report No. 84-58. On this basis, we find that this matter is resolved.

I.D.1 Control Room Design Review

Position

In the SER we issued in July 1981, we stated our position regarding Item I.D.1, "Control Room Design Reviews," of Task I.D, "Control Room Design," of the NRC Action Plan developed as a result of the TMI-2 accident (NUREG-0660); Item I.D.1 is also discussed in NUREG-0737. The objective, as stated NUREG-0660, is to improve the ability of nuclear power plant control room operators to prevent or cope with accidents if they occur by improving the information provided to them. Subsequently, we issued Supplement 1 to NUREG-0737, dated December 17, 1982, in which we confirmed and clarified our requirements in NUREG-0660 regarding a detailed control room design review (DCRDR). In Supplement 1 to NUREG-0737, we require each applicant for an operating license (OL) to conduct their DCRDR on a schedule to be negotiated with the NRC.

In NUREG-0700, we describe four phases of the DCRDR to be performed by OL applicants. These phases are: (1) planning; (2) review; (3) assessment and implementation; and (4) reporting. We provide guidance for the evaluation of the DCRDR in NUREG-0700 and in Appendix A of Section 18.1 of the Standard Review Plan (NUREG-0800).

As a requirement of Supplement 1 to NUREG-0737, OL applicants are required to submit a program plan which describes how the following nine elements of the DCRDR will be accomplished:

(a) Establishment of a qualified multi-disciplinary review team.

(b) A function and task analyses to identify control room operator tasks and information and control requirements during emergency operations.

(c) A comparison of display and control requirements with a control room inventory.

(d) A control room survey to identify deviations from accepted human factors principles.

(e) An assessment of human engineering discrepancies (HEDs) to determine which HEDs are significant and should be corrected.

(f) Selection of design improvements.

(g) A verification that selected desigr improvements will provide the necessary correction.

(h) Coordination of control room improvements with changes from other programs and systems including the safety display parameter system (SPDS), operator training, instrumentation installed in conformance with Regulatory Guide 1.97, and upgrade of emergency operating procedures.

We require each OL applicant to submit a summary report when they have completed their DCRDR. This report should describe the proposed control room changes, the implementation schedules, and provide justification for leaving HEDs significant to safety either uncorrected or partially corrected. We will evaluate the organization, the process, and the results of each DCRDR. The evaluation of the applicant's DCRDR efforts will consist of the following five elements which are described in NUREG-0800:

(a) An evaluation of the Program Plan report submitted by the applicant.

(b) A visit to some to the plant sites to audit the progress of the DCRDR programs.

(c) An evaluation of the applicant's DCRDR Summary Report.

(d) A possible pre-implementation audit.

(e) The preparation of a supplement to the SER which will present the results of our evaluation.

We further require in Supplement 1 to NUREG-0737 that significant HEDs be corrected. Improvements which can be accomplished with an enhancement program may be made promptly. Other control room upgrades may begin following issuance of the supplement to the SER, resolution of any open issues, and our approval of a schedule for upgrade.

A human factors evaluation of the design of the remote shutdown capability provided to meet the requirements of General Design Criteria 19 of Appendix A and Appendix R to 10 CFR Part 50 is not specifically identified as a requirement in Supplement 1 to NUREG-0737. However, we recommend that the scope of the DCRDR include a human factors evaluation of the design of the remote shutdown capability. To the extent practical, without delaying completion of the DCRDR, we also recommend that the DCRDR address any control room modifications and additions (e.g., controls and displays for inadequate core cooling and reactor system vents) made or planned as a result of other post-TMI actions, as well as the lessons learned from operating reactor events such as the Salem ATWS event. Implications of the Salem ATWS event are discussed in NUREG-1000 and required actions are described in Section 1.2, "Post Trip Review - Data and Information Capability," of the enclosure to Generic Letter 83-28.

Discussion and Conclusions

IN the SER we issued in July 1981, we stated our conclusion that with the implementation of the corrective actions contained in Appendix D to the SER and upon completion of actions required to resolve the five open items, the potential for operator error leading to serious consequences as a result of human factors considerations in the control room will be sufficiently low to permit safe startup and operation of the Fermi-2 facility. In Supplement No. 1 to the SER, we provided our favorable evaluation of three of the five open items (i.e., the sound level measurements, the procedures for permanent modifications and color coding in the control room).

Since the applicant has been unable to complete its DCRDR prior to issuance of the Fermi-2 OL, we required it to make a preliminary design assessment (PDA) of its control room to identify HEDs and establish a schedule which we would approve, for correcting any deficiencies. We also require the applicant to complete the DCRDR on the same schedule as licensees with operating plants. As a result of these requirements, the applicant performed a PDA of the Fermi-2 control room and submitted its findings in its report dated April 14, 1981. Our review of this PDA and our on-site review of the Fermi-2 control room are contained in our report entitled, "Control Room Design Review/Audit Report," dated May 26, 1981. The resolution of our findings was discussed in the SER and in Supplement No. 1 to the SER. The applicant submitted revised commitments for resolving some of our findings in its letter dated September 27, 1984, which we find acceptable. The applicant has implemented all modifications required prior to issuance of the Fermi-2 operating license and no items relating to this matter remain open.

We conclude that the applicant has satisfactorily implemented the control room improvements required prior to issuance of an operating license and that this will minimize the potential for operator error leading to serious consequences as a result of human factors considerations in the Fermi-2 control room.

The applicant must comply with the requirements of Supplement 1 to NUREG-0737 for the conduct of a DCRDR. The applicant submitted its DCRDR Program Plan in its letter dated August 16, 1984. In its letter dated April 15, 1983, the applicant committed to provide its DCRDR Summary Report by September 30, 1985. We require that control room design deficiencies identified in our Control Room Design Review/Audit Report as having Priority 3 ratings, be addressed by the applicant during the conduct of its DCRDR and each such finding must be reported in the DCRDR Summary Report. This report must also describe the disposition of any other findings that the applicant committed to evaluate and/or implement in its letters of June 9, 1981, July 31, 1981, July 25, 1984, and September 27, 1984, after issuance of the Fermi-2 operating license. We will implement these requirements as a condition of the license.

II SITING AND DESIGN

II.B.3 Post Accident Sampling Capability

Discussions and Conclusions

In the SER we issued in July 1981, we stated that the applicant had not submitted sufficient information for us to evaluate its compliance with our position on this matter. Accordingly, we proposed to condition the Fermi-2 operating license to satisfy our position. Subsequently, the applicant submitted additional information on this matter and we replaced the Discussions and Conclusions section in the SER for Item II.B.3 with a new section in Supplement No. 2 to the SER. In Supplement No. 2, we concluded that the provisions proposed by the applicant for the Fermi-2 post-accident sampling system satisfied our position on the sampling and analysis requirements of Item II.B.3 in NUREG-0737.

However, the applicant had not at that time completed its procedure for relating radioactive isotopes to estimated core damage. Accordingly, we established a requirement in Supplement No. 2 that the applicant submit an interim procedure and also stated in this supplement that we would provide our evaluation of these procedures in a future supplement to the SER. The applicant subsequently proposed an interim procedure for estimating core damage which we evaluated in Supplement No. 3 to the SER.

We have now completed our review of ten of the eleven criteria in Item II.B.3 of NUREG-0737 and found them acceptable. Our review included the letters submitted by the applicant on December 18, 1971; July 7, 1982; June 18, 1982; and May 3, 1983. The last unresolved matter is related to the final procedure for estimating core damage.

Subsequently, the applicant provided additional information on this final procedure t estimate core damage in its letter dated August 16, 1984. This procedure is identified as Procedure 78.000.15, Revision 1 and is intended to estimate core damage during accident conditions based on the generic procedure proposed by the BWR Owners Group, dated June 17, 1983.

The core damage estimates in the applicant's procedure are based on utilizing post-accident sampling system measurements of Iodine-131 and Cesium-137 concentrations in the primary coolant and Xenon-133 and Krypton-85 concentrations in the primary containment. Additional procedures are provided for estimating the extent of a metal-water reaction (i.e., a zirconium-water reaction) based on measured hydrogen concentrations in the primary containment and for estimating the extent of core damage based on the containment radiation monitors. Reactor vessel water-level is also used to establish whether there has been adequate core cooling through the course of an accident. This meets Criterion (6) and is, therefore, acceptable. We will condition the Fermi-2 operating license to require that the post-accident sampling system (PASS) be operational before exceeding five percent of rated power. Implementation of all the requirements of Item II.B.3 in NUREG-0737 is not necessary prior to low power operation since only small quantities of radionuclide inventory will be generated in the reactor fueland, therefore, will not affect the health and safety of the public. Prior to exceeding five percent power, the applicant must demonstrate the capability to promptly obtain reactor coolant samples in the event of an accident in which there is core damage. We will confirm this demonstration in a future supplement to the SER. Based on our evaluation, we conclude that the applicant's post-accident sampling system meets all the requirements of Item II.B.3 of NUREG-0737 and is, therefore, acceptable.

II.D.1 Performance Testing of Boiling Water Reactor and Pressurized Water Reactor Relief and Safety Valves

Discussions and Conclusion

In the SER we issued in July 1981, we concluded that the applicant had partially satisfied our requirements and position on Item II.D.1 of NUREG-0737 regarding the performance testing of safety/relief valves (SRVs). We stated that we would provide in a future supplement to the SER, our evaluation of the generic test program on SRVs being conducted by the BWR Owners Group. The applicant committed in Amendment 33 to its FSAR to participate in, and adopt the results of, this BWR Owners Group generic test program. We have now completed our review of this generic test program for the performance testing of SRVs and of the applicant's response to six questions which are specific to the Fermi-2 facility.

This review and our evaluation is presented in Appendix L to this supplement. In this appendix, we conclude that the applicant has provided an acceptable response to the requirements of NUREG-0737. The basis for our conclusion is that the applicant participated in an acceptable SRV test program designed to: (1) qualify the operability of prototypical valves; and (2) demonstrate that their operation would not invalidate the integrity of the associated equipment and piping. Our analysis and review of both the test results and the additional information submitted by the applicant, indicated the direct applicability of the prototypical valve test data and the valve system performances to the Fermi-2 valves and systems covered by the generic test program.

Accordingly, we conclude that the applicant has fully met our requirements for Item II.D.1 of NUREG-0737 which thereby provides reasonable assurance that the reactor primary coolant pressure boundary will have, by testing, a low probability of abnormal leakage (General Design Criterion 14) and that the reactor primary coolant pressure boundary and its associated components (i.e., piping, valves and supports) have been designed with sufficient margin so that the design limits are not exceeded during discharge of the SRVs (General Design Criterion 15). Further, the prototypical tests and the successful performance of the valves and associated components demonstrated that this equipment has been constructed in accordance with high quality standards (General Design Criterion 30). On this basis, we find the applicant's program for performance testing of safety and relief valves to be acceptable.

II.E System Design

II.E.4.2 Containment Isolation Dependability

In the SER we issued in July 1981, we provided our evaluation of the seven clarifications regarding Item II.E.4.2 which we issued in NUREG-0737. In Item (6) of this section in the SER entitled "Discussion and Conclusions," we stated that we found the applicant's program to demonstrate purge valve operability to be acceptable. However, we also stated that we would perform a confirmatory audit prior to issuing the Fermi-2 operating license. In supplement No. 3 to the SER, we found that the applicant had satisfactorily demonstrated in the audit conducted on December 2, 1981, that the 6-inch, 10-inch, 20-inch and 24-inch purge valves are capable of closure from a full open position under the pressure loads resulting from the postulated loss-of-coolant accident (LOCA). Our conclusion on this matter was contingent on certain confirmatory items. These confirmatory items were related to: (1) the valve and valve disc orientation; (2) implementation of the applicant's commitments into appropriate Technical Specifications; and (3) certain provisions regarding valves with handwheels.

With respect to the first of these items, the applicant committed in its letter dated January 4, 1982, to reorient the purge valve discs with the disc flat face upstream with two exceptions. Our favorable evaluation of this is contained in Section 4 of Appendix Q to this supplement. We find that this item is now resolved.

The Fermi-2 Technical Specifications have been modified to reflect the applicant's commitments on purge and vent valve operability. (Refer to Paragraph 3.6.3 and Group 14 of Table 3.6.3-1 of the Fermi-2 Technical Specifications.) We find that this matter is now resolved. The last item regarding handwheels on the purge and vent valves has been resolved by implementing written procedures.

We also stated in Supplement No. 3 that we needed additional information to complete our review. This information was basically related to the structural capability of the purge valves under the postulated loads. The applicant subsequently provided this information in a series of letters and meetings extending from 1981 to March 1985. We have reviewed and evaluated this additional information with substantial assistance from our consultant, Brookhaven National Laboratory. This review is summarized in Appendix Q to this supplement. We conclude that the applicant has demonstrated the operability of the purge and vent valves for the Fermi-2 facility under the conditions of the design basis accident in combination with seismic loads. On this basis, we find that this issue is now resolved.

II.F.1 Additional Accident-Monitoring Instrumentation

Attachment, Noble Gas Effluent Monitor

In the SER we issued in July 1981, we concluded that the applicant had provided adequate instrumentation to satisfy our requirements in NUREG-0737 regarding the extended monitoring of noble gases. However, in Report No. 50-341/85-27, enclosed in the letter from Region III to the applicant dated August 10, 1984, we stated that our inspectors observed that the sample lines of the post-accident effluent monitoring system were not heat traced. It is our concern that heat tracing of these sample lines may be necessary to: (1) preclude water traps; (2) minimize deposition of iodine vapor and particulates on the inner surfaces of sampling lines; and (3) to prevent excessive moisture on the collector which may destroy the usefulness of the filter media. In response to our concern on this matter, the applicant stated in its letter dated November 28, 1984, that installation, preoperational testing, procedures, and training in the use of the standby gas treatment system (SGTS) sampling and monitoring equipment is not yet complete. The applicant further stated that the SGTS sample lines will be heat traced to enhance the post-accident sampling capabilities of this equipment. The applicant also requested a waiver so that it could delay completion of the items cited above, prior to exceeding five percent of rated power. Operation of the Fermi-2 facility at or below five percent power will not generate sufficient fission products in the reactor fuel to warrant the need for the SGTS post-accident sampling and monitoring equipment. Accordingly, we find that the waiver requested by the applicant will not endanger public health or safety and is otherwise in the public interest. We conclude that this issue is closed, pending confirmation of the completion of the items cited above prior to exceeding five percent of rated power.

Attachment 2, Sampling and Analysis of Plant Effluents

In the SER we issued in July 1981, we concluded that the applicant's proposed design features for sampling and analysis of plant effluents was acceptable. However, in Report No. 50-341/84-27 cited in the preceding section of this supplement, we stated that the applicant had not yet developed correction factors for sample line losses due to iodine plateout and particulate depositions so as to assure the collection of representative samples. In response to our concern on this matter, the applicant suggested in its letter dated January 8, 1985, that a license condition might be imposed which would require the applicant to verify, prior to startup after the first refueling outage, that the sampling system performs its intended function. It is the applicant's position that the noble gas monitor can be used to project the magnitude of radioiodine and particulate releases in the event that an accident occurs during the first fuel cycle. We find that incorporating such a condition in the Fermi-2 operating license would provide adequate protection, will not endanger life or property, or the common defense and security, and is otherwise in the public interest. On this basis, we conclude that this issue is closed pending verification that the sampling system performs its intended function prior to startup after the first refueling outage.

Attachment 3, Containment High-Range Radiation Monitor

Discussion and Conclusion

In the SER we issued in July 1981, we stated that we found the applicant's commitment for high-range containment radiation monitors to be acceptable. Subsequently, the applicant requested in its letter dated November 1, 1984, that it be granted a partial exception from the calibration requirements of Item II.F.1(3) of NUREG-0737.

We require in Items II.F.1(3), in part, that a special environmental calibration be performed by the manufacturer, prior to initial use. on at least one point per decade in the exposure rate range between 1 rad per hour to 1000 rads per hour. The applicant states in its letter of November 1, 1984, that the Fermi-2 high range detectors were certified by its manufacturer at two points: 10 rads per hour and 50 rads per hour. In lieu of a more extensive calibration by the manufacturer, the applicant has performed an in-situ source calibration for each detector, at two points, 1 rad per hour and 10 rads per hour. Furthermore, the applicant states that it has performed an in-situ electronic calibration for the monitors using electronic signal substitution for a wide range of decades (i.e., 1 to 10^8 rads per hour). These calibrations are considered by the applicant to be adequate to demonstrate the capability of the high-range monitors to qualitatively indicate core damage during and following a postulated design basis accident.

We evaluated the applicant's proposed alternative calibration of their high-range radiation monitors and find it acceptable.

II.F.2 Instrumentation for Detection of Inadequate Core Cooling

In the SER we issued in July 1981, we stated that we would impose two conditions in the Fermi-2 operating license requiring the applicant to: (1) incorporate in-core thermocouples into the design of a system to monitor for inadequate core cooling (ICC) prior to June 1983; and (2) provide additional information regarding the inclusion of these thermocouples in the finalized ICC monitoring system. In Supplement No. 1 to the SER, we added a third proposed licensing condition based on a recommendation from the Advisory Committee on Reactor Safeguards (ACRS). Specifically, the ACRS recommended that we reevaluate our requirement regarding incore thermocouples (Refer to Item (6) in Section 18 of Supplement No. 1.) In response to this ACRS recommendation we proposed the third licensing condition which would have required the applicant to perform a study to determine the most suitable location of the in-core thermocouples.

Subsequently, the BWR Owners Group (BWROG) of which the applicant is a member, submitted a report on this matter. (This BWROG report was later incorporated as Appendix B to the BWROG report, SLI-8218.) Basically, the BWROG concluded that the effectiveness of in-core thermocouples as an indicator of inadequate core cooling is very limited and recommended to us that in-core thermocouples not be used to detect inadequate core cooling.

After reviewing the BWROG recommendation, we questioned the reliability of existing water level instrumentation as the sole indicator of inadequate core cooling. Accordingly, we requested that a further study be performed by the BWROG to evaluate the need for upgrading existing water level instruments to make them more reliable indicators of inadequate core cooling. We also suggested that the BWROG determine whether other instrumentation, including in-core thermocouples, might be needed in the monitoring systems for BWR facilities.

In response, the BWROG submitted two reports in 1982:

(1) SLI-8211, dated July 1982, "Review of Reactor Water Level Measurement System." This report contains the BWROG's evaluation of existing water level instruments and makes recommendations for their improvement.

(2) SLI-8218, dated December 1982, "Inadequate Core Cooling Detection in BWR's." This report presents an evaluation of additional instrumentation as diverse indicators of inadequate core cooling and makes recommendations regarding the need for such additional instrumentation, including in-core thermocouples, for monitoring systems in BWR facilities.

At our request, the applicant also submitted a plant specific evaluation in its letters dated November 16, 1982, September 23, 1983, April 23, 1984, November 15, 1984, and November 28, 1984 addressing the applicability to the Fermi-2 facility of the BWROG's findings and recommendations in reports SLI-8211 and SLI-8218.

We have completed our review of the SLI-8211 report and the applicant's responses to our concerns regarding the reliability of the existing water level instrumentation. The findings from our review of the SLI-8211 report are included in the Generic Letter 84-23, dated October 26, 1984. We find that the proposed Fermi-2 water level system conforms with our first two recommendations for improvement specified in the Generic Letter 84-23.

The third recommendation for improvement in Generic Letter 84-23 is related to the changes to the protection system logic which may be required for those plants in which operator action may be needed to mitigate the consequences of a possible break in a reference leg and a concurrent single failure in a protection system channel associated with an intact reference leg. Implementing these changes will generally require that additional transmitters be added to satisfy the single failure criterion. This third improvement, which we are presently evaluating, may be needed in those plants where an analysis has demonstrated a potential vulnerability.

Since we are still evaluating this third possible improvement, we are not establishing it as a requirement at this time. However, should our continuing review indicate that this matter should be assigned a significantly high priority, we will identify what actions are required in a new generic letter. Our results from the review of this issue will be applied to the applicant; if this is appropriate.

We have completed our review of the SLI-8218 report and have accepted its recommendations that, if the water level instrumentation in a BWR plant conforms with the recommendations in the SLI-8211 report, no additional instrumentation is required for the detection of inadequate core cooling. Since the proposed Fermi-2 water level instrumentation conforms with the recommendations of SLI-8211, we find that there is no additional instrumentation required for the detection of inadequate core cooling. On this basis, we find that the license conditions regarding the detection of inadequate core cooling, need not be placed in the Fermi-2 operating license. Subject to the results of our continuing review of possible improvements in the reactor protection system discussed above, we find that the applicant has addressed Item II.F.2 of NUREG-0737 in an acceptable manner.

II.K.3 Final Recommendations of Bulletins and Orders Task Force

Item 18 Modification of Automatic Depressurization System Logic -- Feasibility for Increased Diversity for Some Event Sequences

Discussion and Conclusion

In the SER we issued in July 1981, we concluded that the applicant had proposed an acceptable design modification to eliminate the need for manual actuation to assure adequate core cooling. Subsequently, the applicant adopted by reference in its letter dated July 31, 1985 the analyses and the results of the BWR Owners Group Report on Item II.K.3.18 of NUREG-0737. Additionally, the applicant has committed to modify the logic of the automatic depressurization system (ADS) to bypass the high drywell pressure trip after a sustained signal indicating a low water level in the reactor pressure vessel and to add a manual switch which may be used to inhibit ADS actuation, if necessary. This is consistent with

Option 4 of the BWR Owners Group study on this matter and which we find acceptable with the following four requirements: (1) installation of this modification must be completed prior to startup following the first refueling outage; (2) the Fermi-2 Technical Specifications must be amended to reflect the addition of the bypass timer and the manual inhibit switch; (3) the use of the inhibit switch must be addressed in the plant emergency procedures; and (4) a plant specific analysis for the Fermi-2 facility must be provided to justify the bypass timer setting.

On this basis, we find the conceptual design for the proposed ADS logic modifications to be acceptable for resolution of Item II.K.3.18.

Item 31 Plant-Specific Calculations to Show Compliance with 10 CFR Part 50.46

In the SER we issued in July 1981, we found acceptable the applicant's commitment to provide plant specific analyses for the Fermi-2 facility if any model changes are required in accordance with Item II.K.3.30 of the SER. Subsequently, we issued Generic Letter No. 83-35, "Clarification of TMI Action Plan Item II.K.3.31," dated November 2, 1983, in which we requested all applicants for an operating license to submit plant specific analyses of a postulated loss-of-coolant accident (LOCA) using the evaluation models which were revised in accordance with Item II.K.3.30. The applicant submitted its response to our Generic Letter in its letter dated March 14, 1984.

Our review of the model submitted by the General Electric Company (GE) to satisfy our requirements of Item II.K.3.30 found the existing GE small-break LOCA model to be in compliance with Appendix K to 10 CFR Part 50. Accordingly, plant specific analyses for the Fermi-2 facility other than those already submitted and approved, need not be submitted to satisfy Item II.K.3.31. On this basis, we find that no further action is required by the applicant to satisfy our requirements for Item II.K.3.31 of NUREG-0737.

III Emergency Preparations and Radiation Protection

III.D Radiation Protection

III.D.1.1 Integrity of Systems Outside Containment Likely to Contain Radioactive Material

In the SER we issued in July 1981, we stated that the applicant's proposed leak reduction, preventative maintenance and leak testing program was acceptable. We also stated that we would verify that the leakage testing had been completed prior to issuing the Fermi-2 operating license.

In its letter dated March 27, 1984, and in Amendment 57 to its FSAR, the applicant revised its description of its proposed leakage reduction program to comply with the requirements of Item III.D.1.1 of NUREG-0737. This revised program includes continuing preventive maintenance to minimize leakage from the systems outside containment which could contain highly radioactive fluids during serious transients or accident conditions. We have reviewed the proposed leakage reduction program and find it to be in compliance with the requirements of Item III.D.1.1 of NUREG-0737 and, therefore, acceptable, with the following exceptions:

(a) The applicant has stated that inspecting for leaks using the helium leak detection method may be considered for some gaseous systems whereas we require in NUREG-0737 that testing of gaseous systems should include helium leak detection or equivalent methods.

(b) The applicant has not described, as we require in NUREG-0737, a program to reduce potential paths due to design and/or operator deficiencies as discussed in our generic letter dated October 17, 1979, to all operating nuclear power plants regarding the North Anna and other related incidents.

(c) The applicant has stated that a report will be submitted to the NRC staff about the time when full power will be achieved in the Fermi-2 facility, of the recorded leakage and the preventive/corrective maintenance performed as a direct result of the applicant's evaluation of this leakage, whereas we require in NUREG-0737 that this matter be implemented by applicants for an operating license prior to issuance of a full power license.

Accordingly, we will require these open items to be resolved by the applicant prior to issuance of a full power license for the Fermi-2 facility.

APPENDIX A

CONTINUATION OF CHRONOLOGY OF RADIOLOGICAL SAFETY REVIEW

September 5, 1984	Letter from applicant concerning inservice testing of pumps and valves.
September 13, 1984	Representatives from NRC & Detroit Edison Company meet in Bethesda, Md. to discuss fire protection measures and the applicant wi[illegible]resent its conceptual design for an alternate [illegible] station. (Summary issued).
September 14, 1984	Representatives from NRC & Detroit Edison meet in Bethesda, Md. to discuss applicant's evaluation of coatings used in the containment drywell and wetwell. (Summary issued)
September 25, 1984	Letter to applicant concerning Fermi-2 Physical Security Plan, Amendment 5.
September 26, 1984	Letter from applicant concerning amended Physical Security Plan.
September 27, 1984	Letter from applicant transmitting FSAR Amendment 59.
September 27, 1984	Letter from applicant concerning annulus pressurization piping load reevaluation.
September 27, 1984	Letter from applicant concerning response to GE SIL 402.
September 28, 1984	Letter to applicant concerning issuance of Supplement No. 4 to the SER for Fermi 2 - NUREG-0798 Supplement No. 4.
October 5, 1984	Letter to applicant transmitting 20 copies of NUREG-0798 Supplement No. 4 (SER Supplement No. 4).
October 5, 1984	Letter from applicant concerning commitment to LRG Instrument Setpoint Methodology Group's Program.
October 6, 1984	Letter from applicant concerning schedule of emergency response information system.

October 11, 1984	Letter from applicant concerning corrections to the FSAR.
October 11, 1984	Letter from applicant concerning Purge Valve Operability.
October 11, 1984	Letter from applicant concerning Primary Containment Coatings Evaluation - Transmittal of Additional Information.
October 11, 1984	Letter from applicant concerning radwaste processing.
October 19, 1984	Letter from CYGNA concerning NRC Design Review Questions - Independent Design Verification Program - Fermi 2.
October 22, 1984	Letter from applicant concerning Design of Alternative Shutdown Approach.
October 22, 1984	Letter from applicant concerning request for exemption to Appendix J.
October 23, 1984	Letter to applicant concerning Fermi-2 Plant Technical Specifications.
October 31, 1984	Letter to applicant concerning approval of Fermi-2 Offsite Dose Calculation Model.
November 1, 1984	Representatives from NRC & Detroit Edison meet in Bethesda, Md. to discuss the applicant's proposed alternate shutdown panel for fire protection (Summary issued).
November 1, 1984	Letter from applicant concerning clarification of position regarding NUREG-0737 post-accident sampling and monitoring capabilities.
November 2, 1984	Representatives from NRC & Detroit Edison Company meet in Bethesda, Md. to discuss the applicant's proposed interim compensatory measures for fire protection and the proposed alternate shutdown panel. (Summary issued)
November 8, 1984	Letter from applicant concerning Fermi 2 Fuel Loading Schedule.
November 12, 1984	Letter from applicant concerning completion schedule for ERIS/SPDS.
November 16, 1984	Letter from CYGNA concerning conversation summary - 11/5/84 - Independent Design Verification Program - Detroit - Enrico Fermi 2.

November 21, 1984	Letter from applicant concerning inservice testing program clarification.
November 26, 1984	Letter from applicant concerning Fermi 2 Technical Specification Certification.
November 27, 1984	Letter to applicant concerning Fermi 2 Draft License.
November 28, 1984	Letter from applicant concerning Comments on SSER 4.
November 29, 1984	Letter from applicant transmitting portions of the amended physical security plan.
December 6, 1984	Letter from applicant transmitting six copies of page 5-19 of the Fermi 2 Security Plan Revisions.
December 12, 1984	Representatives from NRC & DE meet in Newport, Michigan (Fermi 2 Site) to permit NRR management to assess the operational readiness of the Fermi 2 Facility. (Summary issued)
December 12, 1984	Letter from applicant concerning comments on draft operating license.
December 13, 1984	Representatives from NRC & DE meet in Bethesda, Md. to discuss Fermi 2 shore barrier. (Summary issued)
December 13, 1984	Letter from applicant concerning additional clarification of position regarding NUREG-0737 Post-Accident Sampling and Monitoring Capabilities.
December 17, 1984	Letter from applicant concerning FSAR Amendment 60.
December 17, 1984	Letter from applicant transmitting a response to Section 13.3 of Supplement 4 to Safety Evaluation Report (NUREG-0798).
December 18, 1984	Letter from applicant concerning revised process control program.
December 28, 1984	Letter from applicant transmitting a Certificate of Service for Amendment 60.
January 8, 1985	Letter from applicant concerning clarification of position on Silicone Duct Sealant and Other Issues.
January 10, 1985	Letter from applicant concerning purge valves - Additional Information.
January 10, 1985	Letter from applicant concerning Technical Specification Change Request.

January 10, 1985	Letter from applicant concerning CAHRMS Monitor Location.
January 10, 1985	Letter from applicant concerning Primary Containment Coatings Evaluation - Transmittal of Responses to Six Additional NRC Staff Questions.
January 15, 1985	Letter from applicant concerning Contract for Disposal of Spent Nuclear Fuel and/or High Level Radioactive Waste.
January 16, 1985	Letter from applicant concerning Shore Barrier - Supplemental Information.
January 19, 1985	Letter from applicant requesting an exemption to Appendix J. (Air Locks)
January 22, 1985	Letter from applicant concerning change request to the draft Technical Specification.
January 24, 1985	Letter from applicant submitting additional information concerning primary containment coatings.
January 26, 1985	Letter from applicant requesting an exemption to Appendix J.
February 1, 1985	Representatives from NRC and Detroit Edison meet in Bethesda, Md. to discuss the recent damage to EDG Nos. 11 & 12 and the results of the disassembly and inspection efforts on EDG Nos. 11, 12, 13 and 14.
February 14, 1985	Letter from applicant requesting a revision to the draft Fermi-2 Technical Specifications.
February 20, 1985	Representatives from NRC and Detroit Edison meet in Newport, Michigan, (Fermi-2 site) to permit NRR management to discuss the status of Fermi-2 and its operational readiness.
February 26, 1985	Letter to applicant transmitting the revised draft license from Fermi-2.

APPENDIX E

SAFETY EVALUATION REPORT

ON THE

FIRE PROTECTION PROGRAM

FOR THE FERMI-2 FACILITY

This appendix replaces and supersedes Appendix E to the SER issued in July 1981 and the revised Appendix E in Supplement No. 2 to the SER issued in January 1982 with one exception. (Our evaluation of the redundant Class 1E remote shutdown panels in Supplement No. 2 is still valid.)

TABLE OF CONTENTS

I. INTRODUCTION

A. Basis for Reevaluation

We reviewed the proposed fire protection program for the Fermi-2 facility and provided our evaluation of this program in Section 9.5.1 and Appendix E of the SER we issued in July 1981. In Section 9.5.1 of the SER, we stated that except for the control room, we had completed our review of the fire protection for the Fermi-2 facility. We stated in Appendix E to the SER that with the applicant's commitment to make certain modifications prior to fuel load, that the proposed Fermi-2 fire protection program met: (1) the guidelines in Appendix A to BTP ASB 9.5-1; (2) the technical requirements of Appendix R to 10 CFR Part 50; and (3) General Design Criterion 3 of Appendix A to 10 CFR Part 50. On this basis, we found the proposed Fermi-2 fire protection program to be acceptable. In Section 9.5.1 of Supplement No. 2 to the SER, we provided our evaluation of the fire protection for the control room and reaffirmed our prior conclusion regarding the acceptability of the Fermi-2 fire protection program. We also provided in Supplement No. 2, a revised Appendix E which superseded Appendix E in the SER.

Subsequently, during our Fermi-2 site audit on May 14 through May 18, 1984, of the fire protection provided for safe shutdown systems, our inspection team found that the applicant had not provided fire protection for the control room in accordance with its commitments identified in Supplement No. 2. To resolve our concerns on this matter, the applicant in its letters dated August 3, August 4, August 16 and October 22, 1984, provided additional information and requested deviations from Appendix R to 10 CFR 50, and committed to provide an alternative shutdown capability independent of the control room, the cable spreading room, and the relay room. The applicant in its letters dated January 23, February 4 and March 4, 1985, requested our approval of additional deviations from the requirements of Appendix A to Branch Technical Position ASB 9.5-1, and Appendix R to 10 CFR Part 50.

Accordingly, we initiated an extensive rereview of the proposed Fermi-2 fire protection program including the requested deviations cited above. Since the applicant proposed a significant number of revisions and requested a number of deviations, we are issuing a revised Appendix E in this supplement. The present Appendix E replaces and supersedes the two previously issued appendices cited above with one exception noted in Section VII.B.1 of this appendix.

B. Scope of Review

We reviewed the proposed Fermi-2 fire protection program and fire hazards analysis originally submitted by the applicant in October 1977. The Fermi-2 is a one-unit site. This Fermi-2 reevaluation was in response to our request to the applicant to review its fire protection program against the guidelines of Appendix A to Branch Technical Position (BTP) ASB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants." (Henceforth, this will be referred to as Appendix A.) As part of our review, we visited the plant site to examine the relationship of safety-related components, systems and structures in specific plant areas to both combustible materials and to fire detection and suppression systems. The overall objective of our review was to ensure that in the event of a fire at the Fermi-2 facility, the personnel and the plant equipment would be adequate to safely shut down the reactor, to maintain the plant in a safe shutdown condition, and to minimize the release of radioactivity to the environment.

Our review included an evaluation of the automatic and manually-operated water and gaseous fire suppression systems, the fire detection systems, fire barriers, fire doors and dampers, fire protection administrative controls, and the fire brigade size and training.

We reviewed the Fermi-2 fire protection program for conformance with Section 9.5.1, "Fire Protection," of the Standard Review Plan (NUREG-0800). This document contains, in BTP CMEB 9.5-1, the technical requirements of Appendix A to BTP ASB 9.5-1 and Appendix R to 10 CFR 50. Since the applicant has compared its program to these guidelines, this appendix also references these guidelines.

II. FIRE PROTECTION SYSTEMS DESCRIPTION AND EVALUATION

A. Water Supply Systems

The water supply system in the Fermi-2 facility consists of two fire pumps connected to a 12-inch carbon steel, coated and wrapped pipe yard main loop. There is one electric motor and one diesel-driven fire pump; both are rated at 2500 gallons per minute (gpm) at 150 pounds per square inch, gauge, (psig) head each. The fire pumps are UL listed. The controllers for these pumps are not listed but meet the general design and functional requirements of listed controllers which are. Their design and installation conforms to the general guidelines of National Fire Protection Association Standard (NFPA) 20, "Standard for the Installation of Centrifugal Fire Pumps." These pumps are located in the general service water pump house with the diesel fire pump enclosed in a two-hour fire-rated enclosure with automatic sprinklers.

The pumps take suction from the general service water pump header which is supplied from Lake Erie. The fire main system is connected and is pressurized by the general service water system but is capable of complete isolation from the general service water system with a check valve provided to prevent flow from the fire main loop into the general service water system. The pumps discharge into two separate connections to the underground 12-inch yard main loop.

The general service water pumps operate continuously, maintaining a pressure of 150 psig. The fire pumps start automatically on low header pressure. If the water supply system pressure falls to 130 psi, the electric motor-driven fire pump would start automatically. If the system pressure falls to 110 psi, the diesel-engine-driven pump would start. The diesel pump will also start if there is a loss of power to its controller. The pumps can be stopped only at the pump controller panels located in the immediate area. Separate alarms are provided in the control room to monitor pump operation, prime mover availability, and failure of a fire pump to start.

In the event of a loss of offsite power, only the diesel fire pump will be operational. A "closed fuel" valve will stop the diesel engine once the fuel in the line is consumed. At our request, the applicant agreed in its letter dated June 18, 1981, to lock open the supply valve from the elevated diesel fire pump fuel oil storage tank located outside the service water building.

The largest single fire suppression system water demand for areas which need to be protected, is 1500 gpm. Adding 500 gpm for hose streams creates a total water demand of 2000 gpm. Either of the two fire pumps can deliver the required water flow.

A total of 11 yard hydrants are provided at intervals not exceeding 300 feet. A fire hose is provided for each hydrant. Sufficient hose will be provided to cover all areas between hydrants with adequate capacity and pressure.

All valves in the fire protection water supply system are locked open and are under administrative controls. All valves in the fire protection system will be periodically checked to verify position. The water supply valves meet the requirements of Section C.3.b of Appendix A and are, therefore, acceptable.

We find that the water supply system can deliver the required water demand with one pump out of service. Based on our review and the applicant's commitment to lock open the fuel supply valve for the diesel fire pump, we conclude that the proposed water supply system is adequate; and meets the guidelines of Section C.2 of Appendix A. On this basis, we find the water supply system to be acceptable.

B. Sprinkler and Standpipe Systems

The wet pipe sprinkler systems and deluge systems are designed to the requirements of NFPA Standard 13, "Standard for Installation of Sprinkler Systems," and NFPA 14, "Standard for Water Spray Fixed Systems." The manual hose stations are connected separately to the underground water supply loop or building loop header. Appropriate sectional and diversion valves are provided so that primary and secondary fire protection water supplies will always be available should a single break develop. (Refer also to the section covering the residual heat removal building.) The water supply valves to the sprinklers are supervised in accordance with our guidelines. In addition, the sprinkler and standpipe systems have water flow alarms which annunciate in the control room.

The areas which have been equipped with either sprinkler or spray systems, as stated by the applicant in its letter dated June 18, 1981, as the result of its fire hazards analysis, include the following:

Reactor Building

Torus room, Zone 1, elevation 560'
Basement NE corner room, Zone 2, elevation 540'
HPCI turbine and pump room, Zone 3, elevation 540'
Corridor area, Zone 4, elevation 562'
First floor, Zone 5, elevation 583' (railroad bay)
Second floor, Zone 6, elevation 613' (cable trays)

Auxiliary Building

Basement, Zone 1, elevation 551' and 562'
Mezzanine and cable tray area, Zone 2, elevation 583-603'
Ventilation equipment area, Zone 15, elevation 677' (manual water spray)
Cable spreading room, Zone 7, elevation 630'6" (manual spray system)

Residual Heat Removal Complex

Fuel oil storage tank room

Radwaste Building

Baled waste storage area
Voltage regulator (automatic deluge)
Chemical stores
Coalescer rooms
Extruder - evaporator rooms
Drum turn table room
Drum capper room
Drum transfer corridor
Drum conveyor room
First floor main corridor
Main decontamination room

Turbine Building

Reactor feed pump turbine
Turbine oil reservoir
Main lube oil reservoir
Oil storage and turbine oil tank rooms
First floor equipment hatch
Second floor pipe space
Hydrogen seal oil unit (automatic deluge)

Outside Areas

North main transformer (automatic deluge)
South main transformer (automatic deluge)
North system service transformer (automatic deluge)
South system service transformer (automatic deluge)

General Service Water Pumphouse

Diesel fire pump room

Manual hose stations are located throughout the plant to ensure that an effective hose stream can be directed to any safety-related area in the plant. The standpipe systems are consistent with the requirements of NFPA 14, "Standpipe and Hose Systems for Sizing, Spacing, and Pipe Support Requirements."

We conclude that the proposed sprinkler and standpipe systems meet the guidelines of Appendix A to BTP ASB 9.5-1. On this basis, we find the sprinkler and standpipe systems to be acceptable.

C. Gaseous Fire Suppression Systems

The areas which have been equipped with a low-pressure carbon dioxide system include the following:

RHR Building

Emergency diesel-generators
Miscellaneous room, Zone 11, elevation 643'-6"
Cable tunnel, Zone 5, elevation 613'-6"
Cable tray area, Zone 8, elevation 631'

The areas which have been equipped with an automatic, total flooding Halon system include the following:

Auxiliary Building

SGTS, Zone 14, elevation 677'-6"
Cable spreading room
Computer room
Under computer room floor
Relay room

Automatic carbon dioxide systems are activated by heat and/or smoke detectors. Detection devices activate alarms to indicate the presence of a fire and activate control equipment to initiate discharge of fire extinguishing agents. A time delay of sufficient time to enable personnel to leave the areas is provided for each system. Activation of the system may also be accomplished at local points.

The residual heat removal (RHR) complex has its own low pressure carbon dioxide system consisting of two six-ton storage tanks, one for each division in the RHR complex. There is a separate discharge system for each diesel-generator room.

A Halon 1301, total flooding system is used as the primary extinguishing agent in the nonsafety-related computer room underfloor spaces. Products of combustion activate automatic discharge of Halon into this space. The system is activated by both ionization and photoelectric detectors on a Class A fire alarm circuit. At our request, the applicant agreed in its letter dated June 18, 1981, to provide an automatic, total flooding, Halon 1301 system for the cable spreading room in the auxiliary building because of the personnel safety factors associated with a total flooding, carbon dioxide system.

We were concerned that a fire in the computer room could spread into the control room. At our request, the applicant agreed in its letter dated June 16, 1981, to provide a total flooding, Halon 1301, automatic fire suppression system for the computer room located within the control room complex.

We have reviewed the design criteria and bases for the carbon dioxide and the Halon fire suppression systems and conclude that these systems satisfy the provisions of Appendix A to BTP ASB 9.5-1 and are in accordance with the applicable portions of NFPA Standards 12 and 12A. They are, therefore, acceptable.

D. Fire Detection Systems

The fire detection systems consist of the detectors, the associated electrical power supplies, and the annunciation panels. The types of detectors used are: ionization devices which are activated by products of combustion; thermal; and photoelectric devices; and heat sensing cable. Fire detection systems provide an audible and visual alarm which annunciate in the Fermi-2 control room. Local audible and/or visual alarms are also provided. The fire detection systems are installed in all areas having safety-related equipment and/or safety-related cables. These include the control room area, and new and spent fuel pool storage areas, and areas having concentrations of safety-related cables.

The fire detection systems are installed according to NFPA 72D, "Standard for the Installation, Maintenance, and Use of Proprietary Protection Signalling Systems." The applicant has requested a deviation from the requirement to install a permanent recording device arranged to automatically provide a permanant record of fire alarm signals coming into the control room. This device has not been installed as required by Paragraph 1212 of NFPA 72D.

Specifically, NFPA 72D requires a permanent recording device at central supervising stations because of the possibility of an alarm signal being received and not being acknowledged. In the case of the control room which is continuously manned, all alarms are recorded and logged by the operators and alarms signals which are received in the control room can only be reset manually by an operator at the local fire alarm control panel. Therefore, we find that the lack of a permanent recording device is an acceptable deviation from Section E.1(a) of Appendix A.

Those fire detection systems which are used to actuate suppression systems are Class A systems as defined in NFPA 72D. All other redundant safety-related division areas have a cross-zoned Class B system. The two fire detector groups are powered from separate non-Class 1E motor control centers. Each motor control center is fed from opposite divisional Class 1E switchgear. Normal offsite power provides the primary supply for the detectors. Upon loss of offsite power, the detectors are automatically connected to the onsite emergency diesel-generator. All fire alarm circuits and alarm bell circuits are electrically supervised.

We were concerned that smoke detection is not provided in the northeast corner of the auxiliary building at elevation 615' in the stairway adjacent to the relay room. At our request, the applicant agreed in its letter dated June 18, 1981, to install additional smoke detectors.

In its letter dated February 4, 1985, the applicant requested a deviation from the requirement to provide early warning fire detection in the torus area in accordance with NFPA 72E. The torus area has complete automatic sprinkler coverage. The system contains a water flow alarm which is connected to the fire detection system. Upon activation of any sprinkler head, a fire alarm signal will be annunciated in the control room. Fire detectors in the torus area consist of eight ionization smoke detectors which are located adjacent to the exhaust duct grills. However, these detectors do not conform to the spacing requirements of NFPA 72E.

If a fire should occur, the automatic sprinkler system would be expected to activate and suppress the fire. When the system is activated, a waterflow alarm would be annunciated in the control room as discussed above which would summon the fire brigade.

Therefore, we conclude that the automatic sprinkler system which is installed in the torus area, with a flow switch connected to the fire detection system and with the existing smoke detectors, will provide adequate early warning detection for anticipated fires in the torus area. We also conclude that the installation of additional early warning fire detectors would not significantly increase the level of fire safety. Accordingly, we find that the applicant's request for a deviation from Section E.1(a) of Appendix A to BTP ASB 9.5-1 is acceptable.

We have reviewed the proposed Fermi-2 fire detection systems to ensure that fire detectors are adequate to provide detection and alarm of any fires which could occur. These systems are installed with due consideration for the use of detector spacings less than those recommended for smooth, unobstructed ceilings. We have also reviewed the design criteria of the fire detection systems to ensure that they conform to the applicable sections of NFPA 72D and 72E. We conclude that the design and the installation of the proposed Fermi-2 fire detection systems with the deviations we approved above, meets the guidelines of Appendix A to BTP ASB 9.5-1, and are, therefore, acceptable.

III. OTHER ITEMS RELATED TO FIRE PROTECTION PROGRAMS

A. Fire Barrier and Fire Barrier Penetrations

The walls separating safety-related buildings are rated as three-hour fire walls. The floor/ceiling assemblies separating areas in buildings containing safe shutdown systems are rated as three-hour fire barriers. For fire areas not having a three-hour fire-rated assembly, we analyzed each one individually with respect to its fuel load, its fire suppression and detection systems, and its proximity to safe shutdown equipment. We concluded that the fire-rated assemblies provided were adequate for the areas affected and satisfied the guidelines in Section D.1.d and D.1.j of Appendix A to BTP ASB 9.5-1. On this basis, we find the fire barriers and fire barrier penetrations to be acceptable.

Three-hour fire-rated penetration seals are provided for all penetrations of fire-rated walls of floors/ceilings tested in accordance with ASTM E-119 with the exception of penetration seals in the relay room stairwell. In its letter dated August 4, 1984, the applicant provided the results of its analysis of the fire resisting capability of the specific penetration seals used for cable tray cross-over penetrations in the relay room stairwell. This analysis utilizes heat transfer calculations, correlated to actual fire tests, to determine the temperature rise through the penetration. This analysis demonstrates that after a three-hour exposure in accordance with the requirements of ASTM E-119, the maximum temperature rise in the seal will not exceed our acceptance criteria of 325°F. We have reviewed the applicant's analysis and agree with its conclusions.

We, therefore, conclude that the seals used in the relay room stairwell are acceptable.

In its letters dated August 3 and August 4, 1984, the applicant submitted test data concerning the qualification tests of the proposed one-hour fire barrier material to be used for the protection of cable trays. In our review of this material, we noted that the temperatures recorded inside the protective cable wrap envelope exceeded our acceptance criteria of 325°F. At our request, the applicant analyzed the thermocouple data and compared it to the data produced during the qualification testing of a one-hour barrier material which we had previously accepted. Because of a difference in thermocouple locations, higher temperatures were recorded during the testing of the material proposed for use in the Fermi-2 facility.

If the thermocouple had been placed in locations consistent with those in the qualification tests of the approved material, similar results would have occurred. To support the conclusions of this analysis, the applicant submitted in its letter dated October 22, 1984, a report by Underwriters Laboratories, Inc. (UL), an independent, nationally recognized fire testing laboratory. This report independently compared the test results cited above and concluded that, with minor deviations, had the applicant's thermocouples been placed in locations identically to those on the approved fire barrier material, similar results would have been obtained. We agree with these conclusions. We, therefore, conclude that the applicant's one-hour fire barrier material provides an equivalent level of safety as the material previously approved. On this basis we find the proposed fire wrap material to be acceptable.

B. <u>Fire Doors and Dampers</u>

We have reviewed the placement of fire doors and verified that all doorway openings to areas containing safe shutdown equipment or circuits are provided with fire doors with ratings commensurate with the fire ratings of the wall. Based on the applicant's previous submittals on this matter, we found that the applicant had provided three-hour fire doors and dampers wherever ventilation ducts or openings penetrate three-hour fire-rated walls or ceiling/floor assemblies. However, in its letter dated March 4, 1985, the applicant requested a deviation from the requirement to provide a 1 1/2 hour fire rated door in the control room complex (i.e., fire door R3-13). Door R3-13 separates the control room from the turbine building extension at elevation 643'-6". This door is located about 25 feet from the turbine building third floor. The fixed combustibles in the turbine building extension are negligible. The turbine is located about 70 feet from door R3-13, behind a concrete shield wall. Lube oil hazards are enclosed in fire-rated rooms with automatic sprinkler protection. Early warning fire detection has been installed in the turbine building extension and adjacent areas.

The early warning fire detection in the turbine building extension will provide reasonable assurance that a fire will be discovered in its incipient stage and be extinguished by the fire brigade. It is our position that the 1 1/2-hour fire resistant door, R3-13, will provide adequate protection to prevent fires in the area of the turbine building extension from penetrating the control room complex fire barrier. Accordingly, we find that the installation of a 1 1/2-hour fire-resistant door in the control room complex is an acceptable deviation from our guidelines in Section D.1(j) of Appendix A to BTP ASB 9.5-1.

Since some fire dampers were not installed to the manufacturer's specifications, we were concerned that a fire in the immediate area of these fire dampers could collapse the ventilation duct and consequently, pull the fire damper out of the wall. At our request, the applicant agreed in its letter dated June 18, 1981, to reinstall all fire dampers according to the manufacturer's instructions.

Based on the applicant's previous submittals on this matter, we concluded that 3-hour fire-rated dampers would be installed in all 3-hour fire-rated barriers. However, in its letter dated February 4, 1985, the applicant requested a deviation from the requirement to provide 1 1/2 hour fire-rated dampers in certain 3-hour fire-rated barriers. These locations are:

Damper No.	Location
FO 85	Wall separating the control room from Fire Zone 13
FO 90	Wall separating the HVAC duct chase from Fire Zone 13
FO 99	Floor/Ceiling separating the control room from the division 2 CC HVAC room
FO 100	Wall separating the control room from the division 2 CC HVAC room
FO 101	Wall separating the control room from the division 1 CC HVAC room
FO 102	Floor/Ceiling separating the control room from the division 1 CC HVAC room

In this same letter, the applicant also stated that in the following locations, two 1 1/2-hour fire rated dampers in series had been installed in 3-hour fire-rated barriers:

Damper No.	Location
FO 81 A,B	Wall separating the auxiliary building and the control room
FO 92 A,B	Wall separating the auxiliary building and the control room
FO 83 A,B	Wall separating the auxiliary building and the control room
FO 84 A,B	Wall separating the auxiliary building and the control room

The applicant has provided justification for the use of the single and series 1 1/2-hour fire-rated dampers in the 3-hour barriers based on the following:

(1) The fuel load is negligible on either side of the barrier in which the dampers are installed.

(2) Early warning fire detection is provided on each side of the barrier in which the dampers are installed.

The existing early warning fire detection capability will provide reasonable assurance that a fire will be discovered in its incipient stage and be extinguished by the fire brigade within a short time span. The negligible fuel load in each area provides reasonable assurance that the 1 1/2-hour fire-rated dampers will provide adequate protection.

Based on our review, we find that the use of 1 1/2-hour fire-rated dampers in the 3-hour barriers cited above is an acceptable deviation from our guidelines in Section D.1(j) of Appendix A to BTP ASB 9.5-1.

Based on our review and the commitments made by the applicant, we conclude that the proposed fire doors and dampers at the Fermi-2 facility with the deviations we approved above, will be provided in accordance with the guidelines in Section D.1.j of Appendix A to BTP ASB 9.5-1. On this basis, we find the fire doors and dampers to be acceptable.

IV. EMERGENCY LIGHTING

The applicant has installed self-contained eight-hour battery pack emergency lighting in all areas of the plant which could be manned by operators to bring the plant to a safe cold shutdown and in access and egress routes to and from all fire areas.

We conclude that the proposed emergency lighting at the Fermi-2 facility meets the requirements of Appendix A to BTP ASB 9.5-1 and the provisions of Section III.J of Appendix R to 10 CFR Part 50. On this basis, we find the proposed emergency lighting at Fermi-2 to be acceptable.

V. FIRE PROTECTION FOR SPECIFIC AREAS

A. Control Room Complex

The control room complex is separated from the turbine and reactor building as well as other areas of the plant by three-hour fire-rated walls and floor/ceiling assemblies.

Originally, the peripheral rooms including the computer room, in the control room complex did not have one-hour fire walls and doors to separate them from the control room. Also, no automatic fire suppression system was provided for these rooms. We were concerned that fire in one of these rooms might spread into the control room.

At our request, the applicant committed in its letter dated June 16, 1981, to provide one-hour fire-rated walls and doors to separate the peripheral rooms with a potential of a fire hazard from the control room. The applicant will also provide a total flooding, Halon 1301 system for the computer room. The area under the raised floor of the computer room is presently protected by a total flooding, automatic Halon system.

In its letter dated August 16, 1984, the applicant committed to provide an alternate shutdown capability independent of the control room, the cable spreading room and the relay room. Our review of this alternate shutdown capability is contained in Sections VI and VII of this appendix.

Based on our review, we conclude that the proposed fire protection for the control room complex is in accordance with Section F.2 of Appendix A to BTP ASB 9.5-1. On this basis, we find the fire protection for the control room complex to be acceptable.

B. Cable Spreading Room

The cable spreading room is separated from the balance of the plant by three-hour fire-rated barriers. Both safe shutdown divisions are installed in this room. The applicant has provided an alternate shutdown system for the cable spreading room. This independent alternate shutdown system is reviewed in Section VI and VII of this appendix.

The primary fire suppression in the cable spreading room is a total flooding, automatic Halon 1301 system. At our request, the applicant agreed in its letter dated June 18, 1981, to provide a fixed water suppression system as a backup to the Halon 1301 system. This water suppression system meets the guidelines in Section F.3 of Appendix A to BTP ASB 9.5-1. The applicant will use portable blowers to manually remove products of combustion.

Early warning fire detection is provided by smoke detectors. Manual fire fighting capability is provided by portable fire extinguishers.

Based on our review, we conclude that the proposed fire protection for the cable spreading room is in accordance with the guidelines in Section F.3 of Appendix A to BTP ASB 9.5-1. On this basis, we find the fire protection for the cable spreading room to be acceptable.

C. Containment and Reactor Building

The drywell atmosphere of the containment will be inerted with a 97-percent concentration of nitrogen, thereby eliminating any potential fire hazard from lubricating oil or hydraulic fluid systems during operation. Containment and reactor building fire protection features include hose stations, fire detectors, fire extinguishers, automatic sprinklers, and fire control barriers. Ionization smoke detectors are distributed throughout the drywell and provide an alarm and annunciation in the control room. Since the containment is inerted, the provisions of III.O of Appendix R to 10 CFR Part 50 are met.

We have reviewed the applicant's fire protection for the areas inside the Fermi-2 containment and the reactor building and conclude that the proposed fire protection meets the guidelines of Appendix A to BTP ASB 9.5-1. On this basis, we find the fire protection inside the containment and the reactor building to be acceptable.

D. Emergency Diesel-Generator Rooms

The residual heat removal (RHR) complex is located in a separate, detached building and contains the emergency diesel-generators, the diesel oil storage tanks, and the RHR service water pumps, as well as other safety-related equipment and cables. Train I is separated from Train II by a blank, three-hour fire-rated concrete wall. The diesel fuel-oil storage tanks are separated from the other areas by three-hour fire-rated walls and protected by an automatic sprinkler system.

Smoke detectors which initiate alarms and provide annunciation in the control room are provided for the divisions one and two pump rooms and are in the diesel-generator switchgear room.

A single feed from the underground fire main will provide the primary and secondary fire protection for the RHR complex. Accordingly, a break or a closed valve in this line would eliminate all automatic and manual fire protection in the building. At our request, the applicant agreed in its letter dated June 18, 1981, to provide a second feed from the outside under-ground fire main into the building to a common header. This will be properly valved to the extent that one of the two suppression systems will always be available.

Based on our evaluation and the commitments made by the applicant, we conclude that the proposed fire protection for the diesel-generator rooms meets the guidelines of Appendix A to BTP ASB 9.5-1. On this basis, we find the fire protection for the diesel-generator rooms to be acceptable.

E. Fire Protection Measures Not to be Completed Prior to Initial Fuel Load

In its letters dated February 4 and March 4, 1985, the applicant requested approval for completing the installation of certain fire detectors, providing labels on certain fire doors and reperforming the hydrostatic testing of the fire protection yard piping prior to exceeding five percent of full power.

The applicant has committed in the letters cited above to implement the applicable action statements in the Fermi-2 Technical Specifications in each area where the installation of early warning fire detectors has not been completed. In addition, the applicant in its letter dated February 4, 1985, indicated that all modifications to the fire doors recommended by Underwriters Laboratory (UL) have been completed and that the labeling of the fire doors by UL which will be completed before exceeding five percent of full power, will confirm that the modifications have been made in the manner recommended by UL.

The fission product inventory in the Fermi-2 core will not be appreciable prior to exceeding five percent of full power. Accordingly, the health and safety of the public will not be endangered by low power operation (i.e., less than five percent power) before these modifications are completed. Based on the applicant's commitment to implement the applicable Fermi-2 Technical Specification action statements and the low fission product inventory, we find that the applicant will provide adequate fire protection measures. Accordingly, we grant the applicant's request for completing the installation of additional early warning fire detectors, the labeling of fire doors and reperforming the hydrostatic testing cited above, prior to exceeding five percent of full power.

F. Other Plant Areas

The applicant's Fire Hazards Analysis addressed other plant areas not specifically discussed in this report. The applicant has committed to install additional detectors, portable extinguishers, and automatic sprinklers prior to fuel load.

We find that the proposed fire protection for these areas and this commitment by the applicant, is in accordance with the guidelines of Appendix A to BTP ASB 9.5-1 and is, therefore, acceptable.

VI. FIRE PROTECTION OF SAFE SHUTDOWN CAPABILITY

In its letters dated August 16 and October 22, 1984, the applicant committed to provide an alternate shutdown capability, independent of the control room, the cable spreading room, and the relay room. Review of the this independent alternate shutdown system is discussed in Section VII. The applicant will implement the necessary modifications no later than December 31, 1986. or sooner in accordance with a license condition we will impose. In the interim, the licensee has committed in its letter dated October 22, 1984, to provide additional compensatory measures for those areas for which the alternative shutdown capability will eventually be provided, including the following eight areas:

Fire Area	Location	Elevation.
1	Auxiliary Bldg. basement	551'-0" & 562' 0"
2	Auxiliary Bldg. mezzanine	603'-6" & 583' 6"
3	Auxiliary Bldg. relay room	613'-6"
7	Auxiliary Bldg. cable spreading room	630'-6"
8	Auxiliary Bldg. cable tray area	631'-0"
9	Auxiliary Bldg. control room	643'-6"
11	Auxiliary Bldg. third floor	643'-6"
13	Auxiliary Bldg. fourth floor	659'-6"

These areas are currently not protected in accordance with Section III.G of Appendix R to 10 CFR Part 50.

Except for fire area 9 (i.e., the control room) and the fourth floor of the auxiliary building (fire area 13), automatic suppression and detection systems are provided throughout each fire area. However, the control room is continuously manned and the auxiliary building fourth floor has a very low combustible loading. In addition, the applicant has committed in its letter of October 22, 1984, to erect a radiant energy shield to protect one train of redundant components in fire area 13.

In addition to these active and passive fire protection features, the applicant has committed in its letter dated October 22, 1984, to institute in accordance with the applicable Fermi-2 Technical Specifications, a roving fire watch to check each of the areas cited above on an hourly basis. One fire watch will remain in the relay room continuously since all essential circuits required to achieve a safe shutdown could be affected by an uncontrolled fire in this area.

The eight areas for which an independent alternate shutdown capability will be provided have additional passive fire protection in the form of partial fire barriers for selected safe shutdown components. The applicant originally committed to meet Section III.G of Appendix R to 10 CFR Part 50 by providing one-hour fire-rated barriers and automatic suppression systems in some of these areas. However, before the fire barrier installation could be completed, the applicant decided to utilize an independent alternate shutdown system. Since the partially completed fire barriers have not yet been removed, they may provide several minutes of additional fire resistance for the cables so protected.

The applicant provided the results of its analysis within each fire area to determine the specific location of redundant cables and equipment. In fire area 1 (the auxiliary building basement), fire area 2 (the auxiliary building mezzanine), and fire area 9 (the auxiliary building control room), cables were identified which are not protected by either a one-hour fire-rated fire barrier, a partial barrier or 20 feet of separation. For these areas, the applicant will develop interim procedures to permit safe shutdown by alternative means.

Based on our evaluation, we conclude that the interim compensatory measures proposed by the applicant provide reasonable assurance that a potential fire would be sufficiently limited so as to maintain free of fire damge, one train of cables and equipment needed to achieve a safe shutdown.

Our review of the proposed independent alternate shutdown capability is discussed in Section VII of this appendix.

For the remainder of the plant fire areas not covered by the independent alternate shutdown system, the applicant committed in its letter dated August 3, 1984, either to meet the requirements of Section III.G of Appendix R to 10 CFR 50 or to identify and justify any deviations. The applicant identified 16 such deviations. We have evaluated these deviations to ensure that one train of cables and equipment needed to safely shut down the plant will be maintained free of fire damage and to ensure that a level of fire protection safety equivalent to the technical requirements of Section III.G of Appendix R to 10 CFR Part 50, is provided. Based on our review, we conclude that the following deviations requested by the applicant, are acceptable:

(1) Reactor Building, Fire Zone 1, Torus Elevation 540'-0" to 583'-6"

While the redundant cables in this area are separated by 20 feet, there are three intervening cable trays between the redundant cables. Offsetting this, the area is fully sprinklered. In its letter dated August 3, 1984, the applicant committed to provide fire stops in the open intervening cable trays. The fire stops will be installed near column line 12. Based on this commitment, we conclude that the configuration of the torus, the full automatic sprinkler coverage and the proposed cable tray fire stops provide reasonable assurance that a fire will not spread between redundant divisions. On this basis, we find this deviation acceptable.

(2) Corridor, Reactor Building, Fire Zone 4, Elevation 562'-0"

The applicant requested a deviation from Section III.G.2 of Appendix R to 10 CFR Part 50 for this area. Specifically, the applicant states that it cannot provide 20 feet of separation which is free of intervening combustibles. While redundant cables in this area are separated by greater than 20 feet, intervening combustibles in the form of a horizontal cable tray traverses between the redundant divisions. In its letter dated August 3, 1984, the applicant committed to provide a fire stop in the open intervening cable tray and to provide additional automatic sprinkler coverage. Based on the applicant's commitments, we conclude that adequate fire protection has been provided for the redundant cables in the area. The proposed fire stop will provide reasonable assurance that a fire will not spread between redundant divisions. On this basis, we find this deviation acceptable.

(3) Reactor Building, Fire Zone 5, Elevation 583'-6"

The reactor building is divided into a north and south zone at column line 12. The north zone contains division one cables and equipment and the south zone contains division two cables and equipment. In its letter dated August 3, 1984, the applicant committed to provide a separation zone on the west side of the reactor building at column line 12 consisting of 20 feet of separation between the redundant divisions. This zone will be free of intervening combustibles and will have automatic sprinklers. Based on this commitment, we conclude that the combustible free zone, the configuration of the reactor building, the high ceilings and the low combustible loadings will provide reasonable assurance that a fire will not spread between redundant divisions. On this basis, we find this deviation acceptable.

(4) Reactor Building, Fire Zone 6, Elevation 613'-6"

This floor of the reactor building is also divided into a north-south configuration at column line 12. Division one cables and equipment are located in the northern zone and division two cables are located in the southern zone. In its letter dated August 3, 1984, the applicant committed to provide separation zones on both the east and west sides of the reactor building. These zones will have partial sprinkler systems at column line 12. Any intervening open cable trays will be provided with fire stops. Based on these commitments, we conclude that the separation zones combined with the configuration of the reactor building provides reasonable assurance that a fire will not spread between redundant divisions. On this basis, we find this deviation acceptable.

(5) Reactor Building, Fire Zone 7, Elevation 641'-6"

This area contains only division one components. However, it is not separated from the next floor level by three-hour fire-rated barriers Division two equipment on the next floor up is separated by greater than 50 feet with no intervening combustibles and the combustible loading in this area is low. In its letter dated August 3, 1984, the applicant committed to provide a metal cover on all vertical cable trays near column lines F-13. Based on this commitment, we conclude that the wide separation distances and low combustible loading provides reasonable assurance that a fire will not spread between redundant divisions. On this basis, we find this deviation acceptable.

(6) Mezzanine and Cable Tray Area, Auxiliary Building, Elevations 583'-6" and 603'-6", Fire Zone 2

The applicant requested a deviation from Section III.G.2 of Appendix R which requires that there be 20 feet of separation between redundant divisions which is free of intervening combustibles. While the redundant cables in this area are separated by greater than 20 feet, the separation between them is traversed by horizontal cable trays. The applicant has provided automatic sprinkler coverage for the total area in addition to cable tray sprinkler protection. In its letter dated August 3, 1984, the applicant committed to provide fire stops in the intervening cable trays to prevent the spread of a fire from one set of redundant cables to the other. Based on this commitment, we conclude that the automatic sprinkler protection in conjunction with the proposed cable tray fire stops, provides reasonable assurance that a fire will not spread between redundant divisions. On this basis, we find this deviation acceptable.

(7) Ventilation Equipment Area, Auxiliary Building, Elevation 677'-6", Fire Zone 15

This area contains ventilation equipment and a limited number of cables which are required to achieve a safe shutdown. All division two cables are provided with a one-hour fire barrier and the combustible loading in this area is low. However, only the charcoal filters in this area are provided with a water spray system. Because of the lack of combustibles and the one-hour fire barrier cited above, we conclude that the addition of an automatic suppression system throughout the area would not greatly enhance the existing level of fire protection. On this basis, we find this deviation acceptable.

(8) Steam Tunnel, Turbine Building, Elevation 583'

This area contains redundant HPCI and RCIC valves; the minimum separation between these redundant valves is seven feet. Automatic suppression is not provided in this area. Early warning fire detection is provided by heat monitoring instrumentation which provides alarms in the control room. The steam tunnel is a high radiation area and access during operation is limited. The fuel load in the area is negligible. The ceiling height in this area is 57 feet. The redundant HPCI and RCIC valves are installed about 3 feet and 4 feet, respectively, above the floor. In the event a fire occurred in this area, it would involve transient combustibles. Since this area is a high radiation area with limited access, the potential for the accumulation of a large amounts of transient combustibles is unlikely. Additionally, the high ceiling will allow hot gases from a potential fire to rise and dissipate thereby allowing adequate time for the fire brigade to respond and extinguish the fire prior to damage of the redundant valves. The installation of either additional early warning fire detectors or automatic suppression in the steam tunnel would not greatly enhance fire protection. On this basis, we find this deviation acceptable.

(9) Division Two Control Room Ventilation Equipment Room, Auxiliary Building, Elevation 677'-6", Fire Zone 14

This area contains redundant trains of safety-related heating, ventilating, and air-conditioning (HVAC) cables and equipment. The applicant previously committed to provide three-hour fire-rated barriers for all division one cables in this area. Moreover, automatic fire suppression has not been provided in this area. In its letter dated August 3, 1984, the applicant revised this prior commitment. The applicant now proposes to utilize a one-hour fire-rated barrier in lieu of a three-hour fire barrier. The combustible loading in fire zone 14 is 7,600 btu/ft^2 which would correspond to an equivalent fire severity of less than ten minutes on the ASTM E-119 Standard Time-Temperature Curve. An area-wide smoke detection system is provided. Based on our review, we conclude that due to the limited combustible loading, a one-hour fire barrier without an automatic fire suppression system provides reasonable assurance that a fire in this area will not damage redundant cables. On this basis, we find this deviation acceptable.

(10) <u>Cable Tunnel, Auxiliary Building, Elevation 613'-6", Fire Zone 5</u>

The applicant previously committed to provide one-hour fire barriers and an automatic suppression system for the protection of redundant cables in this area. In its letter dated August 3, 1984, the applicant revised this prior commitment. The applicant now proposes to provide a three-hour barrier between redundant divisions and to provide a manually operated sprinkler system. This commitment meets the intent of Section III.G of Appendix R to 10 CFR Part 50. On this basis, we find this deviation acceptable.

(11) <u>Miscellaneous Rooms, Auxiliary Building, Elevation 643'-6", Fire Zone 11</u>

Division one and division two battery chargers and the associated equipment in this zone are separated by a 4-inch solid concrete brick wall which has a three-hour fire door. This wall has a minimum fire rating of 1.5 hours. The carbon dioxide system protects that area where the division one equipment is located; fire suppression is not provided in the division two cubicle. However, smoke detection is provided for both areas. Further, the combustible loading is low; it consists of less than six cable trays associated with the battery chargers. Because of the low fuel loading and the 1.5 hour rated wall separating redundant divisions, we have reasonable assurance that one train will remain free of fire damage. On this basis, we find this deviation acceptable.

(12) <u>Control Room, Auxiliary Building, Elevation 643'-6" to 655'-6", Fire Zone 9</u>

An independent, alternate shutdown capability will be provided for the control room. A fixed suppression system is not provided for this area. However, since the control room is continuously manned, there is reasonable assurance that a potential fire will be promptly detected and extinguished. The addition of a fixed suppression system would not greatly enhance the level of fire protection. On this basis, we find this deviation acceptable.

(13) <u>Reactor Building, Northeast Corner Room, Elevation 540'-0" and 562'-0", Fire Zone 2</u>

The safety-related equipment in this area are the RCIC pump and turbine. An automatic sprinkler system is provided at the 540' elevation but not at the 562' elevation. Cables required to achieve a safe shutdown are provided with a one-hour fire barrier in this area. Fire detection instrumentation is provided throughout this area and the combustible loading is low. Based on our review, we conclude that the low combustible loading and the sprinkler protection on elevation 540'-0" will provide reasonable assurance that a fire in this area will not damage redundant equipment. On this basis, we find this deviation acceptable.

(14) Southeast Corner Room, Reactor Building, Elevation 540'-0" and 562'-0", Fire Zone 2

This area is similar to the northeast corner room described in the previous deviation request. The combustible loading is low (i.e., less than 6 cable trays) and fire detection instrumentation is provided throughout this area. While no fire suppression system is provided, cables required to achieve a safe shutdown are provided with one-hour fire barriers. Based on our review, we conclude that the low combustible loading and the one-hour fire barriers provide reasonable assurance that a fire in this area will not damage redundant equipment. On this basis, we find this deviation acceptable.

(15) Auxiliary Building Basement, Elevation 551'-0" and 564'-0", Fire Zone 1

The applicant requested a deviation from Section III.G.2 of Appendix R to 10 CFR Part 50 which requires that there be 20 feet of separation free of intervening combustibles between redundant divisions. Redundant cables in this area are not separated by 20 feet free of intervening combustibles. However, total automatic sprinkler coverage has been provided throughout this area. In its letter dated August 3, 1984, the applicant committed to provide fire stops between column lines 9 and 11 in the intervening cable trays to prevent the spread of a fire from one set of redundant cables to the other. Based on our review, we conclude that the full coverage automatic sprinkler system in conjunction with the proposed cable tray fire stops, provides reasonable assurance that a fire in this area will not damage redundant equipment. On this basis, we find this deviation acceptable.

(16) Auxiliary Building, Fire Zone 13

There are redundant testability/trip cabinets in this area. The cabinets contain HPCI, RCIC, ADS and LPCI components. The cabinets are separated by 30 feet with no intervening combustibles. The division two cabinet is protected on two sides by fire-rated concrete walls and early warning detection is provided in this area. However, an automatic fire suppression system has not been provided. In its letter dated August 3, 1984, the applicant committed to provide a wall constructed of one-hour fire-rated barrier material in front of the cabinet to act as a radiant energy shield. This shield will provide a structure open only at the end; the opening is required for cooling. The fire loading in the area is low and consists of less than 6 cable trays in the northeast corner of the room. The combustibles are separated from the nearest testability/trip cabinet by at least 40 feet and full fire detection instrumentation is provided in this area. Based on our review, we conclude that this combination of features provides reasonable assurance that a fire in this area will not damage redundant equipment. On the basis, we find that the lack of an automatic suppression system is acceptable.

VII. INDEPENDENT ALTERNATE SHUTDOWN CAPABILITY

A. Introduction

In Appendix E to the SER we issued in July 1981, we concluded that the fire protection proposed by the applicant to achieve a safe shutdown capability for all areas outside the control room, was acceptable. This conclusion was based in part on the applicant's commitment to provide as a minimum, one-hour fire-rated barriers and fixed fire suppression systems to protect at least one redundant train of equipment for certain areas where redundant equipment required to achieve a safe shutdown were within 20 feet of each other. In Appendix E of Supplement No. 2 to the SER, we concluded that the safe shutdown capability in the event of a control room fire, was also acceptable based on the probability of limited fire damage in the control room, two divisionalized remote shutdown panels neither of which was totally isolated electrically from the control room, and certain control room panel and ventilation system features which were unique to the design of the Fermi-2 facility. The basis for our estimate of limited fire damage in the control room was a full-scale test of the effect of a postulated fire immediately adjacent to the control panels.

Subsequently, the applicant has determined that some of the unique control room features which provided the original basis for our conclusion regarding the reasonable assurance that a fire in the control room would only damage one division, could not be readily accomplished. Consequently, the applicant proposed in its letter dated August 16, 1984, and in two letters dated October 22, 1984, to install an independent alternate shutdown capability for the control room.

In addition, based on its decision to provide this alternate shutdown capability for the control room, the applicant decided not to complete the installation of one-hour fire barriers for certain plant areas. Instead, the applicant now proposes to also make the alternate shutdown system independent of these other plant areas. Those areas which will have an incomplete barrier installation and which will also be provided with an alternate shutdown capability are the cable spreading room, the relay room, and fire zones 1, 2, 8, 11 and 13 of the auxiliary building. The applicant proposes to have the alternate shutdown system installed and operational prior to startup following the first refueling outage. During the interim operating period, the applicant proposes specific interim compensatory measures for shutting down the plant in the event of limited fires in these areas. The proposed interim measures are intended to limit the size of a fire.

The interim measures also include procedures, depending on the specific location of the fire, for the remote operation of equipment. These interim operating procedures will basically supplement the normal procedures used to achieve a shutdown either from the control room or from the division one or two remote shutdown panels following a loss of offsite power. The specific equipment operating procedures are evaluated in this section, including our estimate of the degree of difficulty to perform the required operations and the time required to complete the operations. Those proposed interim measures which address fire watches and measures for limiting the size of a fire, are evaluated in Section VI of this appendix. This evaluation covers only those areas to be provided with alternate shutdown capability. The capability of achieving a safe shutdown after a fire in other areas of the Fermi-2 facility have been evaluated in a previous supplement to the SER. (Refer to Appendix E of Supplement No. 2.)

B. Proposed Design Of The Independent Alternate Shutdown System

1. General Description

The alternate shutdown system now proposed by the applicant is designed to provide a safe shutdown capability which is separate and remote from the control center complex (i.e., the control room, the relay room and the cable spreading room) and five other fire zones (zones 1, 2, 8, 11, and 13 of the auxiliary building). This independent alternate shutdown capability would be activated in the event of a fire in the control center complex or the other zones cited above, which damaged equipment and/or cables in these areas to the extent that redundant divisions of shutdown equipment were affected. If only one division of shutdown equipment were to be damaged, a safe shutdown may be accomplished either from the control room or from the division one or two Class 1E remote shutdown panels. (Our evaluation of the post-fire shutdown capability of these two Class 1E remote shutdown panels was provided on Page E-15 of Appendix E of Supplement No. 2 to the SER.)

The proposed independent alternate shutdown system is designed with the capability to accomplish the following goals: (1) achieve and maintain the reactor in a subcritical condition; (2) maintain reactor coolant inventory; (3) achieve and maintain hot shutdown; (4) achieve a cold shutdown within 72 hours; and (5) maintain the reactor in a cold shutdown condition thereafter. These objectives are to be accomplished without taking credit for post-fire repairs to equipment required to achieve either a hot or cold shutdown. Specifically, this proposal by the applicant will involve the installation of a new dedicated shutdown panel designated as the "3L panel" and the installation of instrumentation, controls and transfer switches on the 3L panel and in other areas of the plant. The 3L panel, the switches and the associated instrumentation will be electrically and physically isolated from the fire zones cited above.

2. Systems Used to Achieve the Alternate Shutdown Capability

The independent alternate shutdown system consists of combustion turbine generator (CTG) No. 1, the standby feedwater (SBFW) system, the residual heat removal (RHR) system, the RHR service water (RHRSW) system, the emergency equipment cooling water (EECW) system, the emergency equipment service water (EESW) system, a dedicated control panel (3L), and the associated instrumentation. The necessary support systems include the drywell cooling fans and the RHR room cooling fans. Cooling water is supplied to these coolers by the EECW system which also cools the RHR pump lube oil coolers. All other systems are self-supporting for the period during which they are required to operate. One automatic depressurization system (ADS) valve is provided with isolated controls at the 3L panel to provide a path for the transfer of decay heat from the reactor vessel to the suppression pool.

3. Alternate Shutdown System Description

The independent alternate shutdown system proposed for the control center complex and the five other fire zones, uses existing division one systems plus one division two ADS valve and the newly proposed isolation transfer and control switches located throughout the plant and on the new 3L panel.

The 3L panel is provided solely to achieve an independent alternate shutdown capability in the event of a fire and is, therefore, not designed to Class 1E requirements. The existing divisional remote shutdown panels provide the Class 1E remote shutdown capability. (These are the remote shutdown panels cited in Section VII.B.1 of this appendix and described on Page E-15 of Appendix E to Supplement No. 2 to the SER.) The 3L panel has transfer and control switches for CTG No. 1, the SBFW system, one ADS valve and a 120 KV breaker control. The instrumentation at the 3L panel provides readings of the reactor pressure, the reactor water level, the condensate storage tank water level, the torus water temperature and level, the drywell temperature, the SBFW flow rate and a bus voltage monitor.

Isolation transfer switches and local controls which are independent of the fire areas of concern, are provided near the 3L panel and at various locations in the reactor building for breakers and motor control centers (MCC) required to achieve a safe shutdown. One operator will be required to go to the RHR complex to operate breakers in that building.

C. Evaluation

In the event of a fire in the control center complex or in one of the fire zones of concern which would prevent achieving a safe shutdown from the control room, the only required operator action in the control room is a manual reactor scram if an automatic trip has not already occurred due to the fire. Following a reactor trip, the pressure in the reactor pressure vessel is maintained by the safety/relief valves (SRV) functioning in the safety mode. Makeup water to the pressure vessel will be provided by the SBFW system. In order to start the SBFW system at the 3L panel, CTG No. 1 must be started from the 3L panel and ready for loading; this will take about six minutes. The water source for the SBFW system is the condensate storage tank (CST).

While the plant is being maintained in a hot shutdown condition from the 3L panel using the SBFW system and one ADS valve, the RHR system must be lined up in the suppression pool (i.e., the torus) cooling mode after stripping the Class 1E dc and ac buses. Selected balance of plant dc and ac loads are also stripped to minimize the electrical loads on the buses and possible voltage surges which could affect CTG No. 1. Stripping the buses will prevent spurious signals from adversely affecting the capability to achieve a safe plant shutdown. After reenergizing the division one Class 1E buses, only those loads necessary for maintaining hot shutdown and for achieving and maintaining cold shutdown, will be loaded onto the buses.

The EECW, EESW and RHRSW are already lined up for post-fire cooling. No local operation or realigning of valves is necessary in the event of a fire. Any valves which could spuriously operate and affect operation of safety-related systems are either provided with isolated transfer and control switches or fail in the desired position when the buses are stripped and are not reenergized. All valves in the RHR system whose spurious operation could affect the torus cooling mode or shutdown cooling mode are also provided with isolation transfer and control switches at various MCCs.

The SBFW system has the same capacity as the RCIC system and is, therefore, considered capable of bringing the plant from hot shutdown to the point where the RHR system is capable of being placed in the shutdown cooling mode. Once all support systems are operating, cooldown can be controlled from the 3L panel by using the SBFW system to maintain water level in the reactor and the division two SRV operating in the ADS mode to reduce pressure in the reactor pressure vessel by discharging steam into the torus. When the RHR cut-in temperature and pressure are reached, the RHR system will be realigned into the shutdown cooling mode. The equipment and components necessary for going to the shutdown cooling mode have been previously isolated from the fire areas of concern. Local verification that the RHR suction divisional cross-connect valves are closed will be required. If they have spuriously opened, dc power can be restored for valve operation. Their position is of no consequence when the RHR system is operating in the torus cooling mode since they are isolated from the pump suction during that mode.

Assuming no spurious operations, the SBFW system must be in operation within 20 minutes after evacuating the control room. The worst spurious operation which could occur would be the inadverent opening of an SRV. This is prevented by stripping the Class 1E dc buses. If the valve were to open prior to stripping the buses, the proposed operating procedure requires the operator to deenergize the SRV solenoids at a local panel. The time required to deenergize the solenoids will not affect the capability of the SBFW system to achieve and maintain hot shutdown since it will be operating within six to ten minutes after control room evacuation.

The next function required which is critically dependent on time is drywell cooling. This function should be established within 60 minutes of a reactor scram. Suppression pool cooling should be initiated within two to three hours following a reactor scram.

The proposed design of the alternate shutdown system supplemented by appropriate procedures, will require about 30 minutes to initiate drywell cooling and an additional 30 minutes is required to initiate suppression pool cooling. On this basis, we find that there is a sufficient margin of time to initiate drywell cooling and suppression pool cooling.

We find that the proposed independent alternate shutdown system meets our requirement for achieving cold shutdown within 72 hours and that no repairs following a postulated fire are required to achieve this goal. Shutdown cooling will be started in less than 10 hours after a fire. Since the SBFW system will be operating without forced ventilation for the motor, it is estimated that the design maximum temperature of the motor will be reached in about 60 hours. Therefore, shutdown cooling is required to be established prior to this time limit so that the SBFW system can be shutdown.

The applicant has indicated that the emergency lighting has been evaluated and modified as necessary to provide adequate lighting for access to and egress from the local alternate shutdown locations.

The proposal for a new alternate shutdown system required that a reanalysis of certain associated circuits be performed since, previously, only one division in each of the plant fire zones had to be analyzed. This reanalysis covered the control center complex and the other fire zones for which the independent alternate shutdown capability will be provided. Only those circuits associated with the potential for the spurious operation of safety-related equipment were identified as a concern since the common power supply and common enclosure of associated circuits did not change from the previous analyses. For those spurious operations which could have an adverse effect on the proper functioning of the alternate shutdown capability, the applicant committed to take corrective actions to prevent those potential spurious actuations. To prevent some spurious actuations, isolation transfer switches were installed. But for the most part, a procedural approach is used to prevent or correct adverse spurious operations.

For those systems required for the alternate shutdown design, spurious operations of valves which could adversely affect the system's flow path and which are required to operate to achieve a safe shutdown, are provided with isolation transfer and control switches. To prevent or correct spurious operations of equipment such as the main steam isolation valves, the core spray system, the RCIC system, and the torus water management (TWM) system, the proposed procedures call for stripping the loads off the Class 1E ac and dc buses; only those loads necessary for the independent alternate shutdown system will be reenergized. To prevent spurious actuations from adversely affecting the reliability of the CTG, selected BOP loads are also stripped from the BOP ac and dc buses prior to bus reenergization.

Spurious operation of an SRV is prevented while in the alternate shutdown mode by stripping the Class 1E buses. A supplemental procedure to deenergize the SRV solenoids will be followed if spurious operation occurs prior to that point in the alternate shutdown procedure (i.e., stripping the Class 1E buses).

To prevent depletion of the water inventory in the torus, the RCIC, HPCI, and TWM systems are all disabled by stripping the Class 1E buses. Although the amount of water which could be removed during the time required to strip the buses would not significantly affect the decay heat removal capacity of the system, the test return valve from the RCIC and HPCI systems to the CST will be electrically disabled (i.e., tagged out) during normal plant operation. This will prohibit the potential flow path from the torus to the CST and alleviate concerns regarding the time delays associated with stripping the buses. The return valve will only be operable during HPCI or RCIC testing periods. Tagging out the breaker for the test return valve will also prevent spurious diversion of SBFW system flow back to the CST in the event there was a spurious opening of the two test return valves.

Systems which could deplete the CST water supply due to spurious operation were also identified and will be disabled by stripping them from their buses as part of the implementation of the alternate shutdown procedure. A sufficient water inventory would still be available to bring the plant to cold shutdown in the time it takes to strip the buses.

Although multiple "hot shorts" would be required to cause redundant RHR pump suction valves to open, we require the applicant to operate the Fermi-2 facility with power removed from one or two of the RHR suction isolation valves. The Fermi-2 facility has two parallel isolation valves in series with a single isolation valve. Either the two parallel valves or the single isolation valve will have power locked out during normal plant operation. We will condition the Fermi-2 operating license to ensure this electrical disabling of the appropriate RHR suction isolation valve(s).

The applicant states that the independent alternate shutdown can be accomplished by two operators, one at the 3L panel and one to perform those operations which are not accomplished at the 3L panel, including local isolation of the SRV solenoids as required. Since the operator at the 3L panel starts and controls the CTG and the SBFW system with no need for local operator action, it appears that two operators may be sufficient since the time for accomplishing the other actions in the rest of the procedure is not critical. Further, these other operations are not difficult to perform.

We have reviewed the process and intrumentation drawings (P&IDs) for those systems required for the alternate safe shutdown to both the hot and cold conditions and considered the potential adverse spurious actuations which might prevent these systems from performing their required functions. We have also compared the results with the equipment and components which will be provided with isolation transfer and control switches. Based on that comparison, we conclude that the proposed isolation switches are adequate to perform the alternate shutdown independent of the fire areas of concern.

Based on our preceding evaluation, we conclude that the proposed independent alternate shutdown system design is in accordance with the acceptance criteria in Appendix A of Section 9.5.1 to the Standard Review Plan (NUREG-0800). On this basis, we find that the design of the proposed independent alternate system is acceptable.

D. Interim Operation

The applicant has committed to have the alternate shutdown system installed and operable prior to startup after the first refueling outage. In the interim, the applicant has proposed measures to limit the size of a potential fire and has proposed procedures which include the remote operation of equipment in the event of a fire in the control center complex or one of the other five fire zones requiring an independent alternate shutdown capability. This section of the appendix evaluates the method of shutdown proposed in the event a fire in one of these areas occurs in the interim period before the independent alternate shutdown system is installed and actuated. The analysis and the additional fire protection measures which limit the size of a potential fire are evaluated in Section IV of this appendix.

For each fire zone of concern, the major division of equipment and raceways was identified. The opposite division would then be used for shutdown in the event of a fire in that zone. Each raceway was then reviewed to determine whether the raceway was completely wrapped with a one-hour fire-rated protective envelope, was separate from the redundant safe shutdown division by greater than 20 feet, or did not contain circuits required to shutdown the plant if a fire occurred in that area. For those raceways which were found to meet one of the above criteria, no further review was required. For those raceways which were found not to be completely fire wrapped, a review of the circuits was performed and when manual operator action could be performed outside the fire zone under review, a procedure outline was identified.

For the control room (fire zone 9), the applicant's analysis shows that a fire in any single panel will not adversely affect the capability to achieve a safe plant shutdown. Normal remote shutdown procedures will be used to conduct the shutdown from either the division one or two remote shutdown panels, depending on the exact location of the fire. (These are the remote shutdown panels described and evaluated on page E-15 of Appendix E in Supplement No. 2 to the SER.) These procedures will be supplemented with interim procedures for equipment operation if the fire were to affect panel H11-P807 (auxiliary systems) or H11-P601 (division one ECCS). A fire in panel H11-P807 may disable the ultimate heat sink (UHS) cooling tower fans and could cause a spurious closure of the tower bypass valve. To prevent a spurious closure of the cooling tower bypass valve which would block the service water system flow, an operator would have to go to the RHR complex and manually open the bypass valve, if required, after tripping power to this valve. This action would have to be performed in a relatively short time (i.e., a short-term action) in order to ensure a flow path to the UHS basin. When a cooling tower fan will be required to operate (i.e., a long-term operation), an operator will go to the RHR complex and manually open cooling tower inlet valve 1A. In the relay room, an operator will lift three leads from a terminal board in termination cabinet P868 to isolate the circuits in the control room. The operator at one of the remote shutdown panels will then operate a transfer switch and a control switch to operate fan A in the high speed mode.

In the event of a fire in H11-P601, it may be necessary to perform manual actions to deenergize the SRV solenoids (a short-term operation) and to restore the indication of the torus water level (a long-term action). The short-term action for deenergizing the SRVs requires opening one disconnect switch in each of four cabinets located in the relay room. For the long-term restoration of the torus water level indication, various leads have to be lifted in relay room panel P915 and jumpers must be installed between various terminal board points.

For a fire in the auxiliary building basement (fire zone 1), it is possible to lose both divisions of the control air supply even with a limited fire. Loss of the control air supply would result in a loss of the control center HVAC due to closure of air-operated dampers. Since the control center complex will be relied upon for shutdown during the interim operating period, ventilation and air-conditioning must be restored but is not required over the short-term. Without the control center HVAC, the ambient temperature could rise to 120°F in about four days. (This is the limiting design temperature for equipment in the control room.) Since shutdown of the plant can also be performed from one of the remote shutdown panels in the switchgear rooms in the auxiliary building, the equipment design temperature limits are the main concern with respect to losing the control center ventilation system. The switchgear rooms are each equipped with individual fan cooling units. Assuming offsite power is restored after 72 hours and the fire in zone 1 is extinguished within 72 hours, the Fermi-2 air compressors can be started and aligned to supply air to the division two noninterruptible air supply (NIAS) system which will restore the control center HVAC. Reentry to the fire zone is required to manually open the bypass manual valve at the south control air receiver.

In the event of a fire in the auxiliary building mezzanine (fire zone 2), operator action is required in the short-term to monitor the reactor pressure at a local panel and to restore and ensure proper operation of the EECW system.

An operator will be required to be permanently stationed at the local instrument panel and another operator would have to adjust a valve regulator to bleed off the air supply to the EECW temperature control valve to cause the valve to go full open.

A fire in any one of the other fire areas of concern (zones 3, 7, 8, 11, 13) including the cable spreading room and the relay room, will not require supplemental procedures based in part on the applicant's proposed provisions for cable wrap, separation, and limited fire size which will limit damage to only one safe shutdown division. Safe shutdown for a fire in these areas can be accomplished from the control room using the division one or division two shutdown equipment, depending on which division is postulated to be damaged by the fire. For the analysis of a limited fire in these areas, spurious adverse actuations can be accommodated by automatic protection systems since one division remains operable.

The short-term supplemental procedures for deenergizing the SRV solenoids, monitoring reactor pressure, ensuring proper operation of the EECW system, and ensuring a service water flow path to the UHS, requires uncomplicated actions which the reactor operator should be capable of accomplishing in a short time period. However, the time to open the cooling tower bypass valve is critical since it is possible for this valve to go closed due to a spurious actuation thereby blocking the flow required for diesel-generator cooling. Accordingly, we require during the interim operating period before the independent alternate shutdown system is actuated, that the applicant remove power from this bypass valve during normal plant operation to prevent a spurious closure. We also require that the applicant remove power from RHR suction valves during the interim operating period as discussed previously in the evaluation of the independent alternate shutdown system.

The long-term supplemental procedures for cooling tower fan operation, restoration of the torus water level indication, and restoration of the control center HVAC system are more difficult to perform and should be accomplished by properly qualified operating personnel. However, time is not a critical factor for these actions and reliance on offsite personnel is acceptable. Based on our review of the actions required to achieve a safe shutdown during the interim operating period, a minimum of three onsite operators appears to be required to achieve a safe shutdown following a fire in any one of the fire zones of concern. This includes the operator permanently stationed to monitor reactor pressure in the event of a fire in fire zone 2 (i.e., the auxiliary building mezzanine). A review of the actual shutdown procedures should determine if more than three operators are necessary for maintaining safe shutdown until additional personnel arrive onsite.

The applicant has proposed to rely on its proposed interim procedures which are based on the assumption of a limited fire until the independent alternate shutdown system is installed and operable. This will be no later than December 31, 1986 based on a license condition we will impose. We concur with the applicant's proposal provided that during the interim operating period, power is removed from the RHR suction valves and the cooling tower bypass valve for the division one tower. For the RHR valves, power may have to be removed from one or two valves since there are three valves in the RHR suction line, one valve in series with two parallel valves. We also require that the material needed to perform certain repairs identified in the interim procedures, be on hand at all times and stored in a location outside the fire zones of concern. Emergency lighting for access and egrees routes, for performing needed repairs, and for operation of equipment is also required. Further, the applicant's procedures for maintaining a hot shutdown should not rely on short-term re-entry into the fire zone which is assumed to have a fire. Functions such as torus cooling and drywell cooling should be actuated outside the affected fire zone.

Based on our review of the applicant's interim shutdown methods and supplemental interim procedures which include hot shutdown repairs and local operations, and based on the assumption of a limited fire occurring which will affect only one safe shutdown division, the proposed interim measures for alternate shutdown are an acceptable temporary deviation from the criteria in Appendix A to Section 9.5.1 of the Standard Review Plan. (Refer to Section VI of this appendix in which we find that there is reasonable assurance that any fires which might occur in the interim period, will be limited.) On this basis, we conclude that the proposed alternate shutdown system for interim operation of the Fermi-2 facility is acceptable with the following conditions placed in the operating license.

(a) The applicant will revise its procedures for a fire in any one of the fire zones of concern to include the supplemental interim procedures for alternate shutdown and train the reactor operators in these procedures prior to initial criticality.

(b) During the interim period until the alternate shutdown system is operational, power will be removed from the division one cooling tower bypass valve and the appropriate RHR shutdown cooling mode suction valves as discussed in Appendix E of Supplement No. 5 to the SER.

(c) The independent alternate shutdown system shall be operational prior to startup following the first refueling outage or prior to startup following the first known extended outage of three weeks or longer after we have approved the applicable Technical Specifications for this alternate shutdown system but no later than December 31, 1986.

VIII. ADMINISTRATIVE CONTROLS AND FIRE BRIGADE

The administrative controls for fire protection consist of the fire protection organization, the fire brigade training program, the controls over combustibles and ignition sources, the pre-fire plans, the procedures for fighting fires, and the quality assurance program. The fire brigade will be composed of five members for each shift. To have proper coverage during all phases of operation, members of each shift crew will be trained in fire protection in accordance with our guidance, including Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants." The applicant has agreed to implement the fire protection program contained in the our supplemental guidance, "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance," dated August 29, 1977, including (1) fire brigade training; (2) control of combustibles; (3) control of ignition sources; (4) fire fighting procedures; and (5) the Fermi-2 quality assurance program.

In summary, the applicant will implement the appropriate plant administrative controls and procedures before fuel loading. The applicant will also have a five-member fire brigade which meets our guidelines, and is, therefore, acceptable. Based on the commitments made by the applicant, the size of the fire brigade, the necessary equipment for dealing with fires, and the adequacy of the training, we conclude that the proposed training program for fighting fires will conform with the recommendations of the National Fire Protection Association, with Appendix A to BTP ASB 9.5-1 and with our supplemental guidelines. On this basis, we find that the administrative controls for fire protection at the Fermi-2 facility are acceptable.

IX. TECHNICAL SPECIFICATIONS

The applicant has committed to follow the Standard Technical Specifications related to fire protection measures. We find this acceptable.

X. SUMMARY OF APPROVED DEVIATIONS

We have approved in this appendix, a number of deviations from the requirements of Appendix A to Branch Technical Position ASB 9.5-1 and Appendix R to 10 CFR Part 50. This section summarizes these approved deviations including a cross-reference to the section of this appendix where we discuss each deviation.

1. Early warning fire detectors not installed in the torus area in accordance with NFPA 72E. (Section II.D)

2. Lack of a permanent recording device for the fire alarm system. (Section II.D)

3. Installation of 1 1/2-hour fire rated doors and dampers in 3-hour fire-rated barriers. (Section III.B)

4. Redundant equipment not separated by 20 feet, free of intervening combustibles, in the following areas of the Fermi-2 facility. (Section VI)

   (a) Reactor building, Fire Zone 1, torus elevation 540'-6" to 583'-6"
   (b) Reactor building, Fire Zone 4, corridor, elevation 562'-0"
   (c) Reactor building, Fire Zone 6, elevation 613'-6"
   (d) Auxiliary building, Fire Zone 2, Mezzanine and Cable Tray Area, elevation 583'-6" and 603'-6"
   (e) Auxiliary building, Fire Zone 1, elevation 551'-0" and 564'-0"
   (f) Turbine building, steam tunnel, elevation 583'-0"

5. Partial suppression systems in the following areas: (Section VI)

   (a) Reactor building, Fire Zone 2, northeast corner room, elevation 540'-0" and 562'-0"
   (b) Reactor building, Fire Zone 5, elevation 583'-6"
   (c) Reactor building, Fire Zone 6, elevation 613'-6"

6. Lack of automatic suppression systems in the following areas: (Section VI)

   (a) Auxiliary building, Fire Zone 13, trip cabinet area
   (b) Auxiliary building, Fire Zone 14, division 2 control room ventilation equipment area, elevation 677'-6"

(c) Auxiliary building, Fire Zone 15, ventilation equipment area, elevation 677'-6"
(d) Reactor building, Fire Zone 2, southeast corner room, elevation 540'-0" and 562'-0"
(e) Turbine building, steam tunnel, elevation 583'-0"

7. Lack of 3-hour fire-rated barriers to separate redundant equipment in the auxiliary building, Fire Zone 11, elevation 643'-6" and the reactor building, Fire Zone 5. (Section VI).

8. Lack of a fixed suppression system in the auxiliary building, Fire Zone 9, control room, elevation 643'-6" and 655'-6". (Section VI)

XI CONCLUSION

Based on our re-evaluation in this appendix to Supplement No. 5 to the SER, we find the applicant's proposed fire protection program with the deviations we have approved in this appendix, is still in conformance with the guidelines of Appendix A to BTP ASB 9.5-1, Appendix R to 10 CFR Part 50 and General Design Criterion 3 of Appendix A to 10 CFR Part 50. On this basis, we find the Fermi-2 fire protection program to be acceptable. We will condition the Fermi-2 operating license to include the commitments made by the applicant.

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 our consultants follows the list of staff members.

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APPENDIX L

SAFETY EVALUATION REPORT

ON THE

RELIEF AND SAFETY VALVE TESTING

(Item II.D.1 of NUREG-0737)

FOR THE FERMI-2 FACILITY

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

1. INTRODUCTION

1.1 Background

Light water reactor experience has included a number of instances of improper performance of relief and safety valves installed in the primary coolant systems. There have been instances of valves opening below set pressure, valves opening above set pressure and valves failing to either open or reseat. Investigation of these past instances of improper valve performance did not reveal whether they occurred because of a limited qualification of the valve or because of a basic unreliability of the valve design. It is known that the failure of a power-operated relief valve to reseat was a significant contributor to the sequence of events during the accident at the TMI-2 facility. However, such an event in a boiling water reactor (BWR) would not have the same severe consequences. Nevertheless, these facts led the task force which prepared NUREG-0578 (Reference 1) to recommend that programs be developed and executed which would reexamine the performance capabilities of BWR safety and relief valves for unusual but credible events. These programs were deemed necessary to reconfirm that General Design Criteria (GDC) 14, 15 and 30 of Appendix A to 10 CFR Part 50 are met.

1.2 General Design Criteria and NUREG-0737 Requirements

General Design Criteria 14, 15, and 30 require that: (1) the reactor primary coolant pressure boundary be designed, fabricated and tested so as to have an extremely low probability of abnormal leakage; (2) the reactor coolant system and associated auxiliary, control and protection systems be designed with sufficient margin to assure that the design conditions are not exceeded during normal operation or anticipated transient events; and (3) the components which are part of the reactor coolant pressure boundary shall be constructed to the highest quality standards practical.

To reconfirm the integrity of relief and safety valve systems and thereby assure that the applicable General Design Criteria are met, the NRC staff position in NUREG-0578 was issued as a requirement in a letter dated September 13, 1979, by the Division of Licensing (DL), Office of Nuclear Reactor Regulation (NRR) to all operating nuclear power plants. This requirement has been subsequently incorporated as Item II.D.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements" (Reference 2) which was issued on October 31, 1980. As stated in NUREG-0737, each boiling water reactor applicant shall:

a. Conduct testing to qualify reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents.

b. Determine anticipated valve operating conditions by analyses of the postulated accidents and anticipated operational occurrences discussed in Regulatory Guide 1.70, Rev. 2.

c. Determine which single failure will maximize the dynamic forces on the safety/relief valves.

d. Use the highest test pressures predicted by conventional safety analysis procedures.

e. Include in the relief and safety valve qualification program, the qualification of the associated control circuitry, piping and supports.

f. Test data including criteria for success or failure of the valves which were tested, must be provided for NRC staff review and evaluation. These test data should include data which would permit plant-specific evaluation of discharge piping and supports which were not directly tested.

g. Each applicant must submit a correlation or other evidence to establish that the valves tested in a generic test program demonstrate the functional capability of the primary relief and safety valves installed in its facility. This correlation must demonstrate that the test conditions used are equivalent to the expected operating and postulated accident conditions as described in the applicant's Final Safety Analysis Report (FSAR). The effect on valve operability of the valve discharge piping installed in each nuclear facility must also be considered if it is different from the generic test loop piping.

2. BWR OWNERS GROUP RELIEF AND SAFETY VALVE PROGRAM

To respond to the NUREG-0737 requirements cited above, the BWR Owners Group contracted with the General Electric Company (GE) to design and conduct a Safety/Relief Valve Test Program (Reference 3). This program identifies the safety/relief valves to be tested; the test facility requirements; the test sequence; the valve test acceptance criteria; and the procedure for obtaining, analyzing and reporting the test data. Prior to its acceptance, the test program received extensive NRC staff review and comment followed by responses from the GE/BWR Owners Group. Six NRC questions and Owners Group responses dealing with the justification of the applicability of the test results to the in-plant safety/relief valves are contained in the enclosure to Reference 4. The NRC review of the responses to these questions is contained in Reference 5. Based on our review, the NRC staff concerns expressed in the questions were resolved in an acceptable manner.

The BWRs which were first constructed contain a combination of dual function safety/relief valves (SRV), power actuated relief valves (PARV) and single function safety valves (SV). Nearly all of the problems encountered with these valves have been with the dual function or power actuated valves whose purpose is to limit anticipated operational transients and prevent the safety valves from discharging into the dry well. The single function safety valves, designed and set to comply with the over-pressure protection requirements of the ASME Boiler and Pressure Vessel Code, have been essentially failure free. The safety valves used in these early BWRs were of the same size and configuration of those used for many years in fossil fuel plants and, therefore, were backed by many years of experience. Because of this, direct-acting, single-function safety valves were not included in the test program. The valves included in the test program were direct-acting, dual-function safety/relief valves; power-actuated relief valves; and two and three-stage pilot operated safety/relief valves.

The qualification of the SRVs for steam discharge under expected operating and postulated accident conditions has been demonstrated by vendor production tests and is confirmed routinely by in-plant startup and operability tests. Based on this, the NRC staff agreed that the valves should be tested for those events which result in either a liquid or a two-phase flow through the SRV.

The test sequence and conditions established in the test program were based on an evaluation of expected operating conditions determined by analyses of postulated accidents and anticipated operational occurrences discussed in Regulatory Guide 1.70, Rev. 2. Enclosure 2 to Reference 3 provides this evaluation which indicates that there is one event which is significantly likely to occur and can lead to the discharge of liquid or two-phase flow from the SRVs. This event combined with the single failure assumptions required by NUREG-0737 results in the conclusion that a test should be performed simulating the alternate shutdown cooling mode which utilizes the SRVs as a return flow path for low pressure liquid to the suppression pool.

At a meeting with the NRC staff on March 10, 1981, (Reference 6) the BWR Owners Group presented results of a study by Science Applications, Inc. (SAI) which showed that the probability of getting liquid to the steamline, and then to the SRVs is about 10^{-2} per reactor year. However, even if the water level increases to the mid-plane of the steam line nozzle on the vessel, the fluid quality at the valve was calculated by GE to be greater than 20 percent (Reference 3). It is unlikely that the water level would get this high since the feedwater pumps would be tripped prior to the water level reaching the mid-plane by: (1) the L8 high level trip; (2) the turbine vibration trip; or (3) by operator action. Because the steam lines typically drop about 45 feet vertically from the vessel nozzles to the horizontal runs on which the SRVs are mounted, much of the liquid which gets to the steam lines would be entrained as droplets. Therefore, the two-phase mixture upstream of the SRVs, should liquid reach the level of the steam lines, would exist as a froth, droplet, annular or stratified flow regime. Slug flow or subcooled liquid flow would be unlikely.

Even if a two-phase flow discharge through a SRV should result in a stuck open valve, the effects of the resulting blowdown are not severe. As discussed in Reference 7, historically there have been a total of 53 inadvertent blowdown events due to valve malfunctions in pressure relief systems from 1969 through April 1978. These events varied in consequences from a short-duration pressure transient to a rapid depressurization and cooldown of the primary coolant system from about 1100 psig to a few hundred psig. No fuel failures due to these transients have been reported.

In Reference 8, the BWR Owners Group discusses whether the consequences of the worst case transient (i.e., loss of feedwater) combined with the worst single failure (i.e., failure of the high pressure injection system) and one stuck open relief valve would uncover the core. Analyses for a reference BWR/4 plant and a reference BWR/5 plant demonstrates that the reactor core isolation cooling (RCIC) system can automatically provide sufficient inventory to keep the core covered. This capability is not a design basis for the RCIC system and not all plants have been analyzed to demonstrate this capability. If a plant should not have this capability, manual depressurization to low pressure core cooling systems would prevent uncovering of the core for the case of loss of feedwater plus worst single failure plus a stuck open relief valve. Therefore, even for the loss of feedwater transient with the worst single failure, a stuck open relief valve does not uncover the fuel.

At the meeting held on March 10, 1981 (Reference 6), the BWR Owners Group presented an analysis which demonstrated that even if a slug of subcooled water existed upstream of the SRVs, the probability of rupturing the discharge line is 7×10^{-4} per event. We have not reviewed the supporting analysis for this value. However, even if the failure probability is as high as 10^{-2} per event, the combined probability is no greater than that for a steam line break inside containment. GE states that the steam line break, which has been analyzed and found to be acceptable, would cause more severe effects on the core and containment than a break in a SRV discharge line with a stuck open SRV because the assumed area is larger.

In summary, based on the BWR operating history of inadvertent SRV blowdowns, the likelihood of severe consequences, and the bounding design basis steam line break, we have decided not to require high pressure testing with saturated liquid or subcooled water.

Based on the above considerations, the applicant has complied with the requirements (a) through (d) of NUREG-0578 cited in Section 1.2 of this appendix. That is, an acceptable test program was established which adhered to our guidelines on the selection of test conditions and the maximization of system loads. That portion of Item (e) dealing with the qualification of the associated control circuitry is considered to be satisfied by our review of compliance with Section 50.49 of 10 CFR Part 50. Although that review does not specifically address the associated control circuitry, it does address the adequacy of the programmatic approach to qualification of electrical equipment.

3. BWR OWNERS GROUP TEST RESULTS AND ANALYSIS

In October 1981, the BWR Owners Group published a technical report (Reference 9) documenting the results of the prototypical safety/relief valve tests conducted in accordance with the accepted Test Program (Reference 3). These tests were performed by GE for the BWR Owners Group at the Wyle Laboratory in Huntsville, Alabama. The test report which we reviewed describes the test facility, the basis for the test conditions and the valve selection, the instrumentation and its accuracy, and analyzes the results with respect to valve operability, piping and support loads and the applicability of the test results to the in-plant safety and relief valves.

With the completion of the testing and the submittal of the test report, the applicant has complied with Item (f) of Section 1.2 of this appendix. However, our subsequent review of the test results generated six questions specific to the Fermi-2 facility which required resolution; these six questions are contained in Reference 10. The applicant's response to these six plant specific questions, was submitted for review on November 9, 1982 (Reference 11)

4. NRC REVIEW AND EVALUATION

4.1 Review of Test Results and Analysis

An extensive review (References 12 and 13) of the test results cited above was conducted by our consultant on this matter, EG&G Idaho, Inc., at the Idaho National Engineering Laboratory. This review addressed the generic test results. We reviewed the applicability of the generic test results and the equipment which was tested to the Fermi-2 safety/relief valve systems. The applicant's responses to the six plant specific questions are discussed in Section 4.4 below.

4.2 Valves Tested

The generic test program required the testing of six different safety/relief valves. Included was a Target Rock 6 x 10 two-stage pilot operated safety/relief valve, Model No. 7567F. This valve, with minor differences, is similar to the valve used in the Fermi-2 facility. The tested valve differs from the Fermi-2 valves in the following areas: (1) the top work design; (2) the seat bore diameter; and (3) the main disc lift position.

The only significant differences between the tested valve and the Fermi-2 valve in the top works are dimensional and would not affect the operability of the valve or the piping reaction loads caused by water discharge. While the exact dimensions for the Fermi-2 valves were not provided in the test report, the BWR Owners Group in-plant valves have seat bore diameters and disc lift values which range from 4.27 inches and 2.58 inches, to 5.125 inches and 2.63 inches, respectively. Since the two-stage Target Rock test valve has a 5.125 inch diameter seat bore and a 2.63 inch lift, it thereby bounds the maximum flow capacity.

Although the Fermi-2 facility does not employ the three-stage Target Rock valve, it was also included in the test program. The three-stage test valve has a bore diameter of 4.27 inches and was considered bounding from an operational standpoint since flashing under the water test conditions would be more likely to occur with the smallest bore diameter.

The two-stage test valve bounds the maximum flow capacity and discharge line loads which could be expected for the in-plant valves, and the three-stage test valve verified the operability of the Fermi-2 in-plant valves. Accordingly, the tested valves were considered to be applicable to the in-plant valves at the Fermi-2 facility.

4.3 Test Conditions

As discussed in Section 2.0 of this appendix, test conditions to envelop the expected BWR safety/relief valve events were developed in accordance with our guidelines; they were accepted and are presented in Reference 3. The review of the test results indicates that the actual test conditions were in accordance with the established test program.

4.4 Evaluation of Detroit Edison's Responses to Plant Specific Questions

The applicant's response to Question No. 1 indicates that there are differences in the valve discharge line between the test configuration and the Fermi-2 configuration. However, it is the applicant's position that these differences result in the bounding loads on the SRVs. The first segment of test piping downstream of the test valve is longer than the comparable segment in the Fermi-2 facility (12 ft. vs. 5 ft.) which would result in a higher moment at the test valve. Discharge from the tee quencher at the end of the Fermi-2 SRV discharge line cannot transmit loads to the valve as the test system could since the Fermi-2 line is anchored between the quencher and the valve. Accordingly, we find this portion of the response to be acceptable. The second part of the applicant's response addressed the back pressure (dynamic, hydraulic) loads on the valve which was tested and the Fermi-2 valves. The applicant has addressed both transient and steady-state back-pressure loads. The steady-state back pressure for the test valve was forced to be greater than that expected in-plant by installing a predetermined orifice plate in the discharge line before the ram's head discharge nozzle and above the water line. The applicant's response also indicates that the high pressure steam test preceding the low pressure water test would produce the greater transient back pressures between the two tests. This is due to the higher pressure upstream of the SRV and the shorter valve opening time.

Based on these considerations, we find the applicant's response to the first question to be acceptable.

The applicant's response to the second question described the support system components in the Fermi-2 discharge lines which indicates that spring hangers are installed at the Fermi-2 facility whereas the test facility piping did not include spring hangers. The basic argument defending the adequacy of the spring hangers and all other piping supports is that they were designed for the much larger, high steam pressure relief valve opening loads. In this case, therefore, sufficient margin is available in the Fermi-2 spring hangers to account for the additional load due to the dead weight in the water-filled, low pressure event. The test results indicated significantly lower dynamic loads during the water discharge event than during the high pressure steam discharge case. The point made in this response as well as in the response to Question No. 1, is that the test program was designed primarily to demonstrate valve and system adequacy during the prototypical water discharge events (i.e., the alternate shutdown cooling mode).

Thus, with the in-plant safety/relief valve discharge piping and support system designed for the high pressure steam discharge event and with the satisfactory response of the test valves, the discharge piping and support system to the low pressure water blowdown, we consider the reply to the second question to be acceptable.

The third question expressed our concern that during testing, there may have been valve functional deficiencies or anomalies encountered which invalidated test runs and were not reported in the test results because subsequent valid test runs were obtained. The applicant's response to this question states: "All the valves subjected to test runs, valid or invalid, opened and closed without loss of pressure integrity or damage." This statement is supported with the submittal of the Wyle Laboratory test log sheets for the Target Rock valve tests. Accordingly, we find the response to Question No. 3 to be acceptable.

Question No. 4 asked the applicant to describe and compare anticipated events at the Fermi-2 facility with the test conditions of the generic test program. The applicant summarized the analysis procedure using Regulatory Guide 1.70 which arrived at 13 events that would result in liquid or two-phase flow through the SRVs and maximize the dynamic forces on the valve. As indicated in Section 2.0 of this appendix, this analysis concluded that the alternate shutdown cooling mode is the only anticipated event which could result in liquid at the valve inlet. To simulate this event, the test program used a 15-50°F subcooled liquid at 20-250 psig at the SRV inlet prior to valve opening. The applicant states that the test fluid flow conditions conservatively bound the anticipated Fermi-2 conditions for the alternate shutdown cooling mode. We find the applicant's response to the fourth question to be acceptable.

The fifth question addresses the effect on valve performance of steam flow cycling of the valves prior to the low pressure liquid flow event. The actual tests included only one steam cycle, the purpose of which was to bring the valve up to the proper service temperature prior to the low pressure liquid test. Thus, any adverse effect of several high pressure steam cycles on valve performance during the liquid test was not included. At the Fermi-2 facility, there is no requirement to cycle the SRV to maintain the alternate shutdown cooling modes. In addition, the applicant's response indicates that the valve vendors subject their valves to steam flow cycling and that no loss of valve performance has been noted. We find the applicant's response to this question to be acceptable.

The applicant's response to the sixth question addresses the determination and future use of the valve flow coefficient. The applicant's response indicates that it used this experimentally determined value to confirm that the liquid discharge flow capacity of the Fermi-2 SRVs is sufficient to remove core decay heat when injecting into the reactor pressure vessel (RPV) in the alternate shutdown cooling mode. The value of the valve flow coefficient determined in the SRV test demonstrates that the Fermi-2 SRVs are capable of returning the flow injected by the reactor heat removal (RHR) or core spray (CS) pump to the suppression pool. The test conditions and test configuration were representative of Fermi-2 plant conditions (e.g., pressure upstream of the valve, fluid temperature, friction losses and liquid flow rate) in the alternate shutdown cooling mode. Therefore, the experimentally determined values of valve flow coefficient are appropriate for application to the Fermi-2 plant. We find the applicant's response to this question to be acceptable.

Based on our review and evaluations, we find that the applicant has provided an acceptable response to Item (g) and to the piping and support concerns of Item (e) of Section 1.2 of this appendix.

4.5 Supporting Information

4.5.1 Steam Flow Cycling

The two-stage Target Rock valve has been in service on operating BWRs for only a relatively short period of time (i.e., several years). Set pressure in-service test data compiled to date for this valve indicates that after initial or subsequent setpoint adjustment, the valve set-point tends to drift in an upward direction after some period of operation in a BWR plant.

The Technical Specifications for BWR plants require that safety/relief valves be adjusted to open within one percent of their required set pressure. As found in prior adjustments, two-stage valve data indicate that most valves have been opening in a range of one to four percent above nominal set pressure, with a few valves opening at a considerably higher value. Additionally, during a plant transient at one BWR in mid-1982, all two-stage valves exhibited set-point drift greater than four percent, but on subsequent in-service bench testing, opened in the more typical range of one to four percent.

In response to the NRC and industry concern about the high set-point drift exhibited by the two-stage valves, a BWR Owners Group SRV Drift Committee has been formed consisting of at least some of the utilities that use or plan to use the two-stage valve. The BWR Owners Group has contracted with GE to determine the exact nature of the set-point drift phenomenon. The BWR Owners Group has submitted a report which presents the program test results, conclusions and recommendations.

Resolution of the two-stage Target Rock valve high set-point drift issue will be addressed by the NRC Staff as a separate action and not as part of Item II.D.1 of NUREG-0737.

4.5.2 High Pressure Steam Flow/Discharge Piping Response

We have found the applicability of the response of the safety/relief valve discharge piping system to the response of the in-plant piping system to be acceptable in our preceding discussion. In the test report (Reference 9), it is indicated that: (1) the analytically predicted response of the test piping and supports was comparable to the measured values; and (2) the maximum test piping response to liquid flow was generally less than 30 percent of that due to test steam flow conditions. Further, as part of the initial review, we found the loads on the in-plant piping and supports due to steam discharge to be in an acceptable range. Our on-going review of the Mark-I Containment Long Term Program includes a review of the methods of analysis, computer code adequacy and design criteria for SRV discharge piping and supports for high pressure steam discharge conditions.

5. EVALUATION SUMMARY

The applicant has provided an acceptable response to the requirements of NUREG-0737 and thereby reconfirmed that General Design Criteria 14, 15 and 30 of Appendix A to 10 CFR Part 50 have been met. The basis for this conclusion is given below.

The applicant with our concurrence, developed an acceptable relief and safety valve test program designed to qualify the operability of the prototypical valves and to demonstrate that their operation would not invalidate the integrity of the associated equipment and piping. The subsequent tests were successfully completed under operating conditions which by analysis, bounded the most probable maximum forces expected from anticipated design basis events. The generic test results showed that the valves tested functioned correctly and safely for all steam and water discharge events specified in the test program and that the pressure boundary component design criteria were not exceeded. Analysis and review of the test results and the justifications submitted by the applicant indicate the direct applicability of prototypical valve and valve system performances to the Fermi-2 valves and systems intended to be covered by the generic test program.

Accordingly, the requirements of Item II.D.1 of NUREG-0737 have been met (Items (a) through (g) in Section 1.2 of this appendix) and thereby assure that the reactor primary coolant pressure boundary will have, by testing, a low probability of abnormal leakage (General Design Criterion 14) and that the reactor primary coolant pressure boundary and its associated components (i.e., piping, valves and supports) have been designed with sufficient margin such that design conditions are not exceeded during relief/safety valve events (General Design Criterion 15).

Further, the prototypical tests and the successful performance of the valves and associated components demonstrated that this equipment has been constructed in accordance with high quality standards (General Design Criterion 30).

REFERENCES

1. TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations, NUREG-0578, July 1979.

2. Clarification of TMI Action Plan Requirements, NUREG-0737, November 1980.

3. Letter, D. B. Waters (BWR Owners' Group) to Richard H. Vollmer (NRC) "NUREG-0578 Requirement 2.1.2 - Performance Testing of BWR and PWR Relief and Safety Valves," September 17, 1980.

4. Letter, D. B. Waters, Chairman BWR Owners Group to D. G. Eisenhut, Director, Division of Licensing, NRR, USNRC, "Responses to NRC Questions on the BWR S/RV Test Program," BWR06-8135, March 31, 1981.

5. Letter, B. F. Saffell to R. E. Tiller, Comments on BWR Owners Group Responses to NRC Questions on Safety/Relief Valve Low Pressure Program -Saff-95-81, April 23, 1981.

6. Memorandum to Themis P. Speis from Wayne Hodges, "Summary of March 10 Meeting with General Electric to Discuss BWR Liquid Overfill Events," May 1981.

7. Technical Report on Operating Experience with BWR Pressure Relief Systems, NUREG-0462, July 1978.

8. Letter to Darrell G. Eisenhut (NRC) from David B. Waters (BWR Owners Group), BWROG-80-12, "BWR Owners Group Evaluation of NUREG-0737 Requirements," December 29, 1980.

9. Analysis of Generic BWR Safety Relief Valve Operability Test Results, General Electric NEDE-24988-P, October 1981.

10. Letter from USNRC to Detroit Edison on NUREG-0737, Item II.D.1, "Operability of Safety Relief Valves for Fermi-2," May 27, 1982.

11. Letter, Harry Tauber (Detroit Edison to L. L. Kintner), "Additional Information to Verify the Applicability of Generic SRV Test Results to Fermi 2," November 9, 1982, EF2 60,291.

12. Letter, B. F. Saffell to R. E. Tiller, "Review of BWR/GE Safety Relief Valve Test Report (A6356)" Saff-14-82, January 13, 1982.

13. Letter, B. F. Saffell to D. E. Solecki, "Open Questions-BWR/GE/Safety/Relief Valve Test Report, BWR Owners Safety/Relief Submittals (A6356)"-Saff-178-82, May 4, 1982.

APPENDIX M

SAFETY EVALUATION REPORT

ON THE

TORUS ATTACHED PIPING ANALYSIS

(LOAD AUDIT)

FOR THE FERMI-2 FACILITY

The review contained in this Appendix was prepared with substantial assistance from the Division of Reactor Safety Licensing Assistance of the Department of Nuclear Energy, Brookhaven National Laboratory under contract to the U.S. Nuclear Regulatory Commission.

ABSTRACT

The objective of this report is to document the post-implementation audit which compared the Fermi-2 plant-unique analysis report for torus attached piping against the hydrodynamic load acceptance criteria presented in NUREG-0661. A summary of the audit findings, as well as an overview of the various issues or exceptions to the acceptance criteria identified during the audit, is included. In addition, a table highlighting each issue is provided, along with an indication of the type and status of each issue.

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1 INTRODUCTION

In Appendix I of Supplement No. 3 to the SER, we provided a detailed introduction of the load audit portion of our review regarding the torus attached piping of the Fermi-2 facility. That appendix identified two outstanding issues which were to be reviewed and evaluated following the submittal of additional information by the applicant. This appendix (i.e., Appendix M to Supplement No.5 to the SER) contains our evaluation of these two issues which are: (1) the applicant's proposed model to estimate the suppression pool bulk temperature; and (2) the proposed single vent lateral chugging load.

As discussed in Appendix I of Supplement No.3, a long-term program was proposed and instituted by the BWR Mark I Owners Group to develop generic methods for the definition of suppression pool hydrodynamic loads and the associated structural assessment techniques for the Mark I suppression pools (i.e., the torus). Based on our review of the two generic reports submitted by the Mark I Owners Group (References 1 and 2), we presented our evaluation of the long-term program in NUREG-0661. We concluded that the load definition procedures utilized by the Mark I Owners Group, as modified by NRC requirements, provide conservative estimates of these loading conditions and that the structural acceptance criteria are consistent with the requirements of the applicable codes and standards.

The generic analysis techniques cited above are intended to be used to perform a plant-unique analysis for each Mark I facility to verify compliance with the acceptance criteria of Appendix A to NUREG-0661. The results of the post-implementation audit of the major modification portion of the Fermi-2 plant-unique analysis report (Reference 3) is contained in Reference 4. The two outstanding issues previously identified during our audit (i.e., the use of an unapproved analytical model to estimate local pool temperature during transients involving discharge of the safety/relief valves (SRVs) and the proposed single vent lateral chugging load for each downcomer), have now been resolved. The documentation of the review of the lateral load definition is contained in Reference 5.

This appendix provides our evaluation of the compliance of the Fermi-2 plant unique analysis report for the torus attached piping (Reference 6) with the hydrodynamic load criteria in NUREG-0661.

2 SUMMARY OF THE FERMI-2 POST-IMPLEMENTATION AUDIT

The purpose of this post-implementation audit is to evaluate the hydrodynamic loading methodologies used for the torus attached piping (TAP) portion of the Fermi-2 plant-unique analysis with regard to the acceptance criteria in NUREG-0661. The audit procedure consists primarily of a moderately detailed review of the TAP plant-unique analysis report (PUAR) for the Fermi-2 facility to verify both its completeness and its compliance with the acceptance criteria.

During the prior post-implementation audit of the Fermi-2 PUAR, issues were identified as either exceptions to the acceptance criteria or as areas where additional information was required. (Refer to Appendix I of Supplement No. 3.) In order to resolve these issues, we requested the applicant to submit additional information regarding its PUAR. An overview of this request (Reference 7) is presented in Table M-1, including an indication of the type and status of each item.

Based on our review of the applicant's responses, all issues have been resolved. However, the resolution of Item 1 of Table M-1 is contingent on the applicant performing in-plant SRV tests to support its proposed reduction factor for the SRV water jet impingement load and the air bubble drag load. A description of this exception to the required acceptance criteria is provided in the following section for completeness. In addition, a brief description of our request for additional information (Item 2) concerning the random phasing used in the condensation oscillation torus motion load case is provided in Section 2.2 of this appendix.

2.1 Discussion of Item 1

Fermi-2 TAP SRV Water Jet and Air Bubble Loads.

The applicant has proposed applying a load reduction factor of 0.8 to selected TAP responses to the calculated torus shell motions and the submerged structure/hydrodynamic loads resulting from SRV actuations. These loads were calculated using the methodology in the load definition report (LDR). This reduction factor was obtained by comparing measured responses with responses based on LDR predictions in other in-plant tests. The justification for the 0.8 reduction is based on both torus accelerations and shell pressure measurements. Differences in frequency content between the measured and predicted results is incorporated through the different DLF's for the two cases.

The conservatism in the LDR specification of the SRV bubble pressure in conjunction with the conservatism in the application of the resulting loads to the torus attached piping could lead to a substantial conservatism in the calculated responses. The use the of reduction factor of 0.8 is consistent with a conservative interpretation of the data which was used. This data does not, however, precisely duplicate the Fermi-2 geometry and conditions, nor does it directly compare measured to predicted loads on torus attached piping.

Since the in-plant tests which were used to derive the proposed reduction factor do not appear to be designed primarily for measuring either submerged structural hydrodynamic loads or TAP responses, the proposed reduction factor will have to be justified on the basis of shell motions and stresses, and shell and bubble pressures which will be measured in the Fermi-2 facility. (The proposed instrumentation, especially for measuring the pressures, is adequate to provide both redundancy and estimates of variability and asymmetry.) However, the uncertainties in the application of those previously measured results to obtain the predicted TAP responses, preclude us from reaching a final conclusion at this time as to whether the proposed reduction factor for the Fermi-2 facility is realistic. Accordingly, we find the use of the proposed 0.8 reduction factor conditionally acceptable, subject to confirmation by the SRV tests in the Fermi-2 facility and our approval of the methodology used to derive the proposed reduction factor.

2.2 Discussion of Item 2

The condensation oscillation load described in the load definition report is based on taking the absolute sum of the one Hertz components of a pressure amplitude-frequency spectrum from 0 to 50 Hz. The torus motion resulting from the condensation oscillation load for the torus attached piping, used a random phasing technique to sum the harmonic responses in contrast to the absolute sum combination method required by our acceptance criteria in NUREG-0661. The proposed technique used the harmonic amplitudes corresponding to Test M-12 in the full-scale test facility (FSTF) in conjunction with a set of random phase angles. This proposed procedure is one of several variations for implementing phasing in the condensation oscillation load definition. Based on our review of these various methodologies, we find the proposed alternate procedure to be acceptable for the Fermi-2 facility.

3 CONCLUSIONS

The purpose of the post-implementation pool dynamic load audit of the Fermi-2 plant unique analysis report (PUAR) for torus attached piping was to verify compliance with our acceptance criteria in NUREG-0661. As a result of this audit, several aspects of the Fermi-2 plant unique analysis required additional information. The applicant's responses indicate that the pool dynamic load methodologies used in the Fermi-2 PUAR are in general conformance with our acceptance criteria. While we find that these issues are now resolved, this finding is contingent on the confirmation by the applicant of the SRV water jet impingement load and the air bubble drag load from the in-plant SRV tests. These tests will be a condition of the Fermi-2 operating license.

4 REFERENCES

(1) General Electric Company, "Mark I Containment Program Load Definition Report", General Electric Topical Report NEDO-21888, Revision 2, November 1981.

(2) Mark I Owners Group, "Mark I Containment Program Structural Acceptance Criteria Plant-Unique Analysis Applications Guide, Task Number 3.1.3", General Electric Topical Report NEDO-24583, Revision 1, July 1979.

(3) "Enrico Fermi Atomic Power Plant, Unit 2, Plant-Unique Analysis Report", Volumes 1-5, Detroit Edison Company, DET-04-028-1, Revision 0 (prepared by NUTECH Engineers, Inc.), April 1982.

(4) Attachment to Letter from J. D. Ranlet, BNL, to B. Siegel, NRC, Subject: Fermi-2 Technical Evaluation Report, September 21, 1982.

(5) Attachment to Lettter from J. R. Lehner, BNL, to B. Siegel, NRC, Subject: Adequacy of the Existing Mark I Downcomer Chugging Lateral Load Specification, October 1983.

(6) "Enrico Fermi Atomic Power Plant, Unit 2, Plant-Unique Analysis Report for Torus Attached Piping and Suppression Chamber Penetrations", Detroit Edison Company, DET-19-076-6, Revision 0 (prepared by NUTECH Engineers, Inc.), June 1983.

(7) Attachment to Letter from J. D. Ranlet, BNL, to B. Siegel, NRC, Subject: Request For Information on Fermi 2 Plant-Unique Analysis Report (Torus Attached Piping), August 26, 1983.

TABLE M-1

Issues Previously Identified in the Post-Implementation Audit

Item	Description	Exception to the NUREG-0661 Acceptance Criteria	Status
1.	Justify the reduction factors used for the SRV water jet impingement and air bubble drag loads,	yes	Resolved*
2.	Describe the random phasing technique used for the condensation oscillation torus motion load case.	yes	Resolved

*This item is considered resolved contingent on confirmation of the proposed reduction factor by the Fermi-2 in-plant SRV tests.

APPENDIX N

SAFETY EVALUATION REPORT

ON THE CONTROL OF HEAVY LOADS

FOR THE FERMI-2 FACILITY

(PHASE I)

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

ABSTRACT AND EXECUTIVE SUMMARY

The Nuclear Regulatory Commission (NRC) has requested that all nuclear plants either operating or under construction submit responses indicating compliance with our guidelines in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The Detroit Edison Company (the applicant) has submitted data and made commitments which demonstrate that the Fermi-2 facility has been designed and built consistent with Section 5.1.1 of NUREG-0612. This appendix to Supplement No. 5 to the SER contains summaries of the applicant's actions and commitments, including our evaluation of the applicant's responses for the Fermi-2 hoist units with respect to our seven guidelines. This appendix is Phase 1 of our evaluation of this matter.

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1 INTRODUCTION

1.1 Purpose of Review

This appendix contains our review and evaluation of the general load-handling policy and procedures at the Fermi-2 facility and was performed with the objective of assessing the conformance of the applicant with the general load-handling guidelines of Section 5.1.1 to NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

1.2 Background

We established Generic Technical Activity Task A-36 to systematically examine our licensing criteria and the adequacy of measures in effect at operating nuclear power plants to assure the safe handling of heavy loads and to recommend any required changes to these measures. This activity was initiated by our letter dated May 17, 1978, to all power reactor licensees, requesting information concerning the control of heavy loads near spent fuel.

The results of our effort on Task A-36 were reported in NUREG-0612. Our conclusion was that existing measures to control the handling of heavy loads at operating plants, although providing protection from certain potential problems, do not adequately cover the major causes of load-handling accidents and, therefore, should be upgraded.

In order to upgrade measures for the control of heavy loads, we developed a series of guidelines designed to achieve a two-phase objective using an acceptable approach or protection philosophy. The first portion of the objective, achieved through a set of general guidelines identified in Section 5.1.1 of NUREG-0612, is to ensure that all load-handling systems at nuclear power plants are designed and operated such that their probability of failure is uniformly small and appropriate for the critical tasks in which they are employed. The second part of our objective, achieved through guidelines identified in Section 5.1.2 through 5.1.5 of NUREG-0612, is to ensure that for load handling systems in areas where their failure might result in significant consequences, either: (1) features are provided, in addition to those required for all load-handling systems, to ensure that the potential for a load drop is extremely small (e.g., a single failure-proof crane); or (2) conservative evaluations of load-handling accidents indicate that the potential consequences of any load drop are acceptably small. Acceptability of accident consequences is quantified in NUREG-0612 into four accident analysis evaluation criteria.

The approach we used to develop our guidelines for minimizing the potential for a load drop was based on our concept of defense-in-depth and is summarized as follows:

(a) Provide sufficient operator training, handling system design, load-handling instructions, and equipment inspection to assure reliable operation of the handling system.

(b) Define safe load-travel paths through procedures and operator training so that, to the extent practical, heavy loads are not carried over or near irradiated fuel or equipment required to achieve a safe shutdown.

(c) Provide mechanical stops or electrical interlocks to prevent movement of heavy loads over irradiated fuel or in proximity to equipment associated with redundant shutdown paths.

Our guidelines resulting from the foregoing broad criteria are presented in Section 5 of NUREG-0612.

1.3 Plant Specific Background

On December 22, 1980, we issued a letter to the applicant requesting that it review its provisions for handling and control of heavy loads at the Fermi-2 facility, evaluate these provisions with respect to the guidelines of NUREG-0612 and provide certain additional information to be used for an independent determination of its conformance to these guidelines. On December 3, 1981, the applicant provided its initial response to our request. On June 3, 1982, additional information and drawings were submitted. On October 15, 1982, following discussions of our review of the previous responses, a revised report was submitted by the applicant to resolve comments and upgrade its prior submittals. On April 3, 1984, the applicant submitted additional information.

2 EVALUATION AND RECOMMENDATIONS

2.1 Overview

The following sections summarize the applicant's review of heavy load handling at Fermi-2 and our evaluation and conclusions regarding its compliance with the intent of NUREG-0612. While the applicant did not specify the weight of a single spent-fuel element and its handling tool (this is the definition of a heavy load in NUREG-0612), the applicant specifically defined for its use, a heavy load as "greater than one ton."

2.2 Heavy-Load Overhead Handling Systems

This section reviews the applicant's list of overhead handling systems which are subject to the criteria of NUREG-0612 and a summary of its justification for excluding overhead handling systems from this list.

2.2.1 Scope of Review

The scope of our review is summarized by the question we asked regarding heavy loads. This question is:

"Report the results of your review of plant arrangements to identify all overhead handling systems from which a load drop may result in damage to any system required for plant shutdown or decay heat removal (taking no credit for any interlocks, technical specifications, operating procedures, or detailed structural analysis) and justify the exclusion of any overhead handling system from your list by verifying that there is sufficient physical separation from any load-impact point and any safety-related component to permit a determination by inspection that no heavy load drop can result in damage to any system or component required for plant shutdown or decay heat removal."

2.2.2 Summary of the Applicant's Responses

The applicant's review of overhead handling systems identified cranes and hoists and provided separate tables entitled "Hoists Capable of Handling Loads Over Spent Fuel or Shutdown Safety Systems Components" and "Overhead Hoists Exempt From Further Analysis Because They Cannot Handle Heavy Loads Over Spent Fuel or Shutdown Safety Systems Components." These tables identified and excepted certain cranes and hoists located in buildings which do not contain safety-related equipment required to achieve safe shutdown. In its submittal dated October 15, 1982, the applicant supplemented its prior submittals by adding data on additional hoists and provided information on six hoists not then purchased. The applicant stated that these hoists and any other cranes or hoists which must meet the guidelines of NUREG-0612, will be purchased to specifications consistent with NUREG-0612.

2.2.3 Evaluation and Conclusions

We conclude that the applicant has included all applicable hoists and cranes in its list of handling systems which must comply with the requirements of the general guidelines or NUREG-0612. On this basis, we find that the applicant's responses indicate that comprehensive reviews have been made and the required identification is complete for the hoists presently installed and to be purchased.

2.3 General Guidelines

This section addresses the extent to which the applicable handling systems comply with the general guidelines of Section 5.1.1 of NUREG-0612. Our evaluation and conclusions are provided for each guideline.

We have established seven general guidelines which must be met in order to comply with our defense-in-depth approach for the handling of heavy loads. These guidelines consist of the following criteria from Section 5.1.1 of NUREG-0612:

(a) Guideline 1, Safe Load Paths
(b) Guideline 2, Load-Handling Procedures
(c) Guideline 3, Crane Operator Training
(d) Guideline 4, Special Lifting Devices
(e) Guideline 5, Lifting Devices (not specially designed)
(f) Guideline 6, Cranes (Inspection, Testing, and Maintenance)
(g) Guideline 7, Crane Design

These seven guidelines must be satisfied for all overhead handling systems and programs in order to handle heavy loads in the vicinity of the reactor vessel, near spent fuel in the spent-fuel pool or in other areas where a load drop may damage systems required to achieve a safe shutdown. The following sections address the guidelines individually.

2.3.1 Safe Load Paths (Guideline 1)

"Safe load paths should be defined for the movement of heavy loads to minimize the potential for heavy loads, if dropped, to impact irradiated fuel in the reactor vessel and in the spent-fuel pool, or to impact safe shutdown equipment. The path should follow, to the extent practical, structural floor members, beams, etc., such that if the load is dropped, the structure is more likely to withstand the impact. These load paths should be defined in procedures, shown on equipment layout drawings, and clearly marked on the floor in the area where the load is to be handled. Deviation from defined load paths should require written alternative procedures approved by the plant safety review committee."

(a) Summary of the Applicant's Responses

The applicant has revised its refueling equipment laydown location drawings. The laydown locations and travel paths to them for the reactor crane and the main and auxiliary hoists utilize the criteria of NUREG-0612. Also, the Fermi-2 facility has completed 16 procedures and has one under development which addresses the administrative, rigging, and load-handling concerns in NUREG-0612. The procedures include definitions of "Safe Load Paths" and prior to refueling will, wherever practical, require floor lines to show the heavy load paths.

Because of the high strength integrity of the fueling floor (i.e., the fifth-floor) based on its heavily reinforced, 24-inch thick concrete construction, very little added strength is achieved along the building column lines. However, travel paths along these column lines have been established where it is practicable, to keep the travel and placement as simple as possible so as not to confuse operators and supervisors. The established travel paths now on the Fermi-2 drawings will be included in specific maintenance procedures developed prior to criticality with the exception of the procedure for the spent-fuel cask which will be developed prior to handling after initial criticality. An initial step in the procedure will require the person responsible for performing the lift to verify the safe load path is free of obstructions which would interfere with the movement of the load. Because of the high strength integrity of the fueling floor at all locations and the physical separation of redundant safety-related systems located below the fueling floor, deviation from the travel paths shown on the drawings do not notably increase the consequences of any potential accidents, as long as these deviations do not traverse over the reactor, the fuel storage pool, and the equipment hatch areas. Therefore, the placement of painted travel path lines for each heavy load offers very little advantage and could cause confusion. However, painted barrier lines and signs will be established around the reactor, the fuel pool, and the equipment hatch areas. Additionally, painted travel paths will be provided for the five major loads handled over the fifth-floor deck. These include the reactor shield plugs, the reactor vessel head, the drywell head, the spent-fuel cask, and the equipment storage pool slot plugs. For significant loads, placement of temporary markers will identify the load path.

(b) Evaluation and Conclusions

We conclude that the actions and commitments of the applicant are consistent with Guideline 1 of NUREG-0612. On this basis, we find that the actions reported and planned by the applicant should fully satisfy the safe load path guidelines.

2.3.2 Load-Handling Procedures (Guideline 2)

"Procedures should be developed to cover load-handling operations for heavy loads that are or could be handled over or in proximity to irradiated fuel or safe shutdown equipment. At a minimum, procedures should cover handling of those loads listed in Table 3.1-1 of NUREG-0612. These procedures should include: identification of required equipment; inspections and acceptance criteria required before movement of load; the steps and proper sequence to be followed in handling the load; defining the safe path; and other special precautions."

(a) Summary of the Applicant's Responses

The applicant has committed to complete its written procedures prior to initial criticality. Unanticipated load-handling procedures will be written prior to handling of the load. The procedures will meet the guidelines in NUREG-0612. A table was provided which lists the heavy loads carried by each crane along with its designated lifting devices. In order to control future heavy loads which may be handled over or near spent fuel or required safe shutdown equipment, the procedures governing the operation of the reactor building crane, the monorails, and the portable hoists will require the guidelines of NUREG-0612 be followed either by specific maintenance procedures or by attachment to maintenance orders/work packages prior to movement of heavy loads in these areas.

(b) Evaluation and Conclusions

The applicant has committed to complete load handling procedures prior to initial criticality. On this basis, we find that the applicant's commitments are consistent with Guideline 2 of NUREG-0612.

2.3.3 Crane Operator Training (Guideline 3)

"Crane operators should be trained, qualified, and conduct themselves in accordance with Chapter 2-3 of ANSI B30.2-1976, 'Overhead and Gantry Cranes'."

(a) Summary of the Applicant's Responses

Operator training, qualification, and conduct will be in compliance with the requirements of ANSI B30.2-1976 for operation of overhead traveling cranes. Operators of various types of cranes will be trained and qualified to the appropriate standard for the specific type of equipment which will be used. Records of personnel training and qualification will be retained. This training will be administered by the Nuclear Operations Training Group and will be implemented prior to fuel loading. Those individuals operating cranes/hoists will be qualified prior to involvement with any post-criticality heavy-load handling event.

(b) Evaluation and Conclusions

We find that the standards in Chapters 2-3 of ANSI B30.2 and the appropriate standards for other specific types of equipment, meet the general requirements for Guideline 3 of NUREG-0612. On this basis, we find that the applicant's commitments for crane operator training, qualification, and conduct are acceptable.

2.3.4 Special Lifting Devices (Guideline 4)

"Special lifting devices should satisfy the guidelines of ANSI N14.6-1978, 'Standard for Special Lifting Devices for Shipping Containers Weighing 10,000

Pounds (4500 kg) or more for Nuclear Materials. This standard should apply to all special lifting devices which carry heavy loads in areas as defined above. For operating plants, certain inspections and load tests may be accepted in lieu of certain material requirements in the standard. In addition, the stress design factor stated in Section 3.2.1.1 of ANSI N14.6 should be based on the combined maximum static and dynamic loads that could be imparted on the handling device based on characteristics of the crane which will be used. This is in lieu of the guideline in Section 3.2.1.1 of ANSI N14.6 which bases the stress design factor on only the weight (static load) of the load and of the intervening components of the special handling device."

(a) Summary of Applicant's Responses

To date, there are only three lifting devices provided for the handling of heavy loads which would fall within the guidelines of ANSI N14.6-1978, as defined in NUREG-0612. These are the RPV head strongback, the dryer/separator lifting device and the vessel head insulation spreader beam. A design review conducted by the designer, the General Electric Company, finds that the RPV head strongback and the dryer/separator lifting device are in full compliance with the criteria for strength contained in Section 3.2 of ANSI N14.6-1978, taking into account the combined static and dynamic loads. However, certain components in these lifting devices do not meet the additional strength criteria of Section 6.2 for single failure-proof systems. Therefore, the RPV strongback and dryer/separator strongback will be upgraded to meet the single failure-proof criteria of Section 6.2 of ANSI N14.6. After initial criticality, modifications will be completed prior to the use of these devices.

The vessel head insulation spreader beam is being designed to achieve compliance with the ANSI N14.6 strength criteria for combined static and dynamic load forces.

The spent-fuel cask handling system is accepted as single failure-proof and is addressed in detail in Section 9.1.4.2.1 of the Fermi-2 FSAR.

All other special lifting devices and slings will be purchased to ensure that the requirements of ANSI N14.6-1978 and ANSI B30.9-1971 are satisfied.

(b) Evaluation and Conclusions

We find that the applicant has complied with the requirements of Guideline 4, except for components failing to meet Section 6.2 of ANSI N14.6-1978. We also find that the commitments made by the applicant provide assurance that the Fermi-2 facility will meet Guideline 4 of NUREG-0612. On this basis, we find that the status of the special lifting devices at the Fermi-2 facility and the commitments made by the applicant to resolve the deficiencies cited above, are acceptable.

2.3.5 Lifting Devices (Not Specially Designed) (Guideline 5)

"Lifting devices that are not specially designed should be installed and used in accordance with the guidelines of ANSI B30.9-1971, 'Slings'. However, in selecting the proper sling, the load used should be the sum of the static and maximum dynamic load. The rating identified on the sling should be in terms of the 'static load' which produces the maximum static and dynamic load.

Where this restricts slings to use on only certain cranes, the slings should be clearly marked as to the cranes with which they may be used."

(a) Summary of the Applicant's Responses

Slings which were used for construction will not be retained for handling of heavy loads around critical equipment after the plant is operational.

The requirements of the design stress factor will include the maximum static and dynamic loads as defined in NUREG-0612. Any single failure-proof handling system will also meet the requirements of Section 5.1.6 of NUREG-0612.

Additionally, the static rating of each sling will be clearly marked on the sling as well as any information which might restrict the sling to only certain cranes and loads.

(b) Evaluation and Conclusions

We find that the planned actions for lifting devices will, upon completion, be consistent with Guideline 5 of NUREG-0612. On this basis, we find that the commitments made by the applicant for lifting devices not specially designed, are acceptable.

2.3.6 Cranes (Inspection, Testing and Maintenance) (Guideline 6)

"The cranes should be inspected, tested, and maintained in accordance with Chapter 2-2 of ANSI B30.2-1976, 'Overhead and Gantry Cranes,' with the exception that tests and inspections should be performed prior to use where it is not practical to meet the frequencies of ANSI B30.2 for periodic inspection and test, or where frequency of crane use is less than the specified inspection and test frequency (e.g., the polar crane inside a PWR containment may only be used every 12 to 18 months during refueling operations, and is generally not accessible during power operation. ANSI B30.2, however, calls for certain inspections to be performed daily or monthly. For such cranes having limited usage, the inspections, test, and maintenance should be performed prior to their use)."

(a) Summary of the Applicant's Responses

The reactor building crane, main and auxiliary hoists, crane inspection, testing, and maintenance procedures will comply with the guidelines in Chapter 2-2 of ANSI B30.2-1976. Should any deviations from this standard be required, they will be made subject to the requirements of ANSI B30.2-1976. The requirements of this standard will be incorporated into the Reactor Building Crane General Maintenance Procedures No. 35.000.120. This procedure will be written prior to fuel loading.

For all other overhead hoists, the inspection, testing, and maintenance procedures will comply with Chapter 1.2 and 2.2 of ANSI B30.16-1973.

(b) Evaluation and Conclusions

We find that since procedures are to be written using either Chapter 2-2 of ANSI B30.2 or ANSI B30.16-1973 for conformance guides, the specified guidelines can be met. The conformance requirements in both of these ANSI documents include inspection, testing, and maintenance. On the basis, we find that the commitments made by the applicant for crane inspection, testing and maintenance are consistent with the intent of Guideline 6 of NUREG-0612. On this basis, we find them acceptable.

2.3.7 Crane Design (Guideline 7)

"The crane should be designed to meet the applicable criteria and guidelines of Chapter 2-1 of ANSI B30.2-1976, 'Overhead and Gantry Cranes,' and of CMAA-70, 'Specifications for Electric Overhead Traveling Cranes.' An alternative to a specification in ANSI B30.2 or CMAA-70 may be accepted in lieu of specific compliance if the intent of the specification is satisfied."

(a) Summary of the Applicant's Responses

The reactor building main crane is the only single failure-proof crane at the plant site. In Section 9.1.4.2.1 of the FSAR, the applicant describes the single failure-proof design features incorporated in this 125-ton crane.

The Fermi-2 reactor building crane was designed in accordance with EOCI No. 61, "Specifications for Electric Traveling Cranes." However, additional, upgraded criteria included in the later CMAA 70-1976 specification was already a part of the manufacturer's design practices. As part of the applicant's submittal, analyses were included to show that CMAA criteria which differ from EOCI-61 have been satisfied concerning the following 16 items:

(1) The design stress shall not exceed 20 percent of the published average ultimate strength of the material.

(2) The welding design and procedures conform to AWS D14.1 and the material conforms to ASTM A-36.

(3) The impact allowance minimum is 15 percent of rated capacity of the hoist for speeds up to 30 feet per minute.

(4) The twisting moments due to overhanging loads and lateral forces acting eccentric to the horizontal neutral axis of a girder, are calculated based on the distance between the load center of gravity and the girder section shear center.

(5) The longitudinal stiffener is to be located at 0.4 times the distance from the compression flange inner surface to the neutral axis.

(6) The ratios governing the girder allowable compressive stress value are less than the criteria required by EOCI-61.

(7) The diaphragm plate thickness is sufficient to keep the trolley wheel load-bearing stress within 26,000 pounds per square inch (psi).

(8) The allowable vertical stresses without impact will be 14,400 psi in either tension or compression.

(9) The rated load capacity plus the bottom block weight divided by the number of strands of rope must not exceed 20 percent of the published rope breaking strength.

(10) The drum shall be designed to withstand the combined crushing and bending loads.

(11) The minimum drum groove depth is 3/8 x the rope diameter. (The minimum drum groove pitch is 1.14 + the rope diameter or the rope diameter + 1/8 in., whichever is less.)

(12) The horsepower rating of the gearing in the gearbox will be based upon AGMA Standards.

(13) The hoist motion holding brakes meet minimum specified torque requirements.

(14) The bridge and trolley bumpers will be rigidly mounted and capable of stopping the crane within specified acceleration limits.

(15) Criteria have been provided which address static control.

(16) Protection has been provided to prevent motors from restarting upon restoration of power following a loss of power, until the control handles are brought to the off position.

For all other overhead hoists listed for the Fermi-2 facility, the applicant stated that CMAA 70 and ANSI B30.2 are not the applicable standards for these hoists; ANSI B30.16-1973 is stated to be the applicable standard. The design of the recirculation pump hoists conforms to this standard. The remaining hoists which were not yet purchased at that time, have been specified to conform to the requirements of ANSI B30.16-1973.

(b) Evaluation and Conclusions

We find that the applicant has provided sufficient information to show that it meets Guideline 7 of NUREG-0612. On the basis, we find that the design criteria of the cranes at the Fermi-2 facility are acceptable.

2.4 Interim Protection Measures

We have established six measures in Section 5.3 of NUREG-0612 which should be initiated to provide reasonable assurance that handling of heavy loads will be performed in a safe manner until final implementation of the general guidelines in Section 5.1 of NUREG-0612 is complete. Four of these six interim measures are: Guideline 1 (Safe Load paths); Guideline 2 (Load-Handling Procedures); Guideline 3 (Crane Operator Training); and Guideline 6 (Cranes Inspections, Testing, and Maintenance). The two remaining interim measures cover the following criteria: (1) the heavy load technical specifications; and (2) the special review for heavy loads handled over the core.

Our review of the applicant's implementation and evaluation of one of these interim protection measures is contained in the following sections.

2.4.1 Interim Protection Measure (Technical Specifications)

"Licenses for all operating reactors not having a single failure-proof overhead crane in the fuel-storage pool area should be revised to include a specification comparable to Standard Technical Specification 3.9.7, 'Crane Travel - Spent-Fuel-Storage Pool Building,' for PWRs and Standard Technical Specification 3.9.6.2, 'Crane Travel,' for BWRs, to prohibit handling of heavy loads over fuel in the storage pool until implementation of measures which satisfy the guidelines of Section 5.1."

2.4.2 Summary of the Applicant's Statements

The applicant has not made the Fermi-2 facility operational. Accordingly, we did not review interim protection.

2.4.3 Evaluation and Conclusions

None required

3 SUMMARY OF CONCLUSIONS

3.1 Applicable Load-Handling Systems

Based on the information supplied by the applicant, we conclude that the list of cranes and hoists which are subject to the provisions of NUREG-0612, is acceptable. (Refer to Section 2.2.1 of this appendix.) However, we require that the justification for excluding cranes not included in this list, should be available in the event of an audit.

3.2 Guideline Recommendations and Findings

We find that the applicant's program for handling heavy loads is consistent with our seven guidelines for handling of heavy loads. The applicant's commitment to meet all our requirements in Section 5.1.1 of NUREG-0612 are presented in Table 3.1 of this appendix.

3.3 Interim Protection

Our evaluation of the information provided by the applicant indicates that the no action is required to ensure that our six measures for interim protection at Fermi-2 are met since the Fermi-2 facility is not in operation.

TABLE 3.1. ENRICO FERMI ATOMIC POWER PLANT UNIT 2, NUREG 0612 COMPLIANCE MATRIX

Equipment Designation	Heavy Loads (tons)	Weight or Capacity (tons)	Guideline 1 Safe Load Paths	Guideline 2 Procedures	Guideline 3 Crane Operator Training	Guideline 4 Special Lifting Devices	Guideline 5 Slings	Guideline 6 Crane-Test and Inspection	Guideline 7 Crane Design
Rea. Bldg Crane Main Hoist	Shield Plugs & 24 more	125	C	C	C	C	C	C	C
Rea. Bldg. Crane Aux Hoist	Maint Tools & 3 more	5	C	C	C	C	C	C	C
N&S Torus Hatch Hoist	N&S Floor Hatches	5	C	C	C	C	C	C	C
HPCI Hoist	HPCI Floor Shield Plug	12	C	C	C	C	C	C	C
HCIC Hoist	Floor Hatches & Turbine	10	C	C	C	C	C	C	C
RHR Pumps-Div. I B'smt Hoist	Hatches & Motors	16	C	C	C	C	C	C	C
RHR Pumps-Div. B'smt Hoist	Hatches & Motors	16	C	C	C	C	C	C	C
RHR Pumps-Div. I 1st Fl. Hoist	Hatches & Motors	16	C	C	C	C	C	C	C
RHR Pumps-Div. II 1st Fl. Hoist	Hatches & Motors	16	C	C	C	C	C	C	C
N&S Recir. Pump Hoists	MG Set Fluid Dr.	25	C	C	C	C	C	C	C
MG Stes, N, C, & S Hoists	MG Set Fluid Dr.	12	C	C	C	C	C	C	C
MG Set Fluid Dr. N&S Hoist	MG Set Fluid Dr.	20	C	C	C	C	C	C	C
CRD Repair Hoist	CRD Transfer Cask	3	C	C	C	C	C	C	C
Core Spray Div I. Hoist	1st Fl. Hatches	16	C	C	C	C	C	C	C
Core Spray Div II. Hoist	Basement Hatches	16	C	C	C	C	C	C	C

TABLE 3.1. (continued)

Equipment Designation	Heavy Loads (tons)	Weight or Capacity (tons)	Guideline 1 Safe Load Paths	Guideline 2 Procedures	Guideline 3 Crane Operator Training	Guideline 4 Special Lifting Devices	Guideline 5 Slings	Guideline 6 Crane-Test and Inspection	Guideline 7 Crane Design
Diesel Gen. Div. I N&S Hoist (2 each)	*	2	C	C	C	C	C	C	C
Diesel Gen. Div. II N&S Hoist (2 each)	*	2	C	C	C	C	C	C	C
Diesel Gen. MCC Div. I Hoist (2 each)	*	4	C	C	C	C	C	C	C
Diesel Gen. MCC Div. II Hoist (2 each)	*	4	C	C	C	C	C	C	C
Vent. Eqpt. Rm. Hoist	*	8	C	C	C	C	C	C	C
Cranes listed as exempt from analysis because they cannot handle heavy loads are not included here.									

C = Licensee action or commitment complies with NUREG-0612 Guideline.
NC = Licensee action does not comply with NUREG-0612 Guideline
R = Licensee has proposed revisions/modifications designed to comply with NUREG-0612 Guideline.
I = Insufficient information provided by the Licensee.
* To Be Determined by Utility

APPENDIX P

SAFETY EVALUATION REPORT
ON THE CONTROL OF HEAVY LOADS
FOR THE FERMI-2 FACILITY
(PHASE II)

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

ABSTRACT AND EXECUTIVE SUMMARY

The Nuclear Regulatory Commission (NRC) has requested that all nuclear plants, either operating or under construction, submit a response indicating compliance with NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." This report contains our evaluation and recommendations for the Fermi-2 facility with respect to the requirements of Sections 5.1.4, 5.1.5 and 5.1.6 of NUREG-0612 (Phase II). Our evaluation of the applicant's response to Section 5.1.1 (Phase I) of NUREG-0612 is contained in Appendix N of this supplement.

Based on our review, we find that the Detroit Edison Company (the applicant) has through equipment design and upgrade, risk analysis and commitments, shown that the Fermi-2 facility is consistent with the guidelines in Sections 5.1.4, 5.1.5 and 5.1.6 of NUREG-0612. We conclude that there are not outstanding items requiring recommendations or further resolution. We find that this matter is now resolved.

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TABLES

1. INTRODUCTION

1.1 Purpose of Review

This appendix contains our review and evaluation of the general load-handling policy and procedures at the Fermi-2 facility. We performed this evaluation with the objective of assessing the conformance of the Fermi-2 facility with the general load-handling guidelines in Sections 5.1.4, 5.1.5 and 5.1.6 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." This evaluation constitutes Phase II of a two-phase evaluation. Phase I assesses the conformance of the Fermi-2 facility with the guidelines in Section 5.1.1 of NUREG-0612; our evaluation of Phase I is contained in Appendix N to this supplement.

1.2 Background

Generic Technical Activity Task A-36 was established by the U.S. Nuclear Regulatory Commission (NRC) staff to systematically examine our licensing criteria and to determine the adequacy of measures in effect at operating nuclear power plants to assure the safe handling of heavy loads and to recommend any required changes to these measures. We initiated this activity in our letter dated May 17, 1978, to all applicants for an operating license in which we requested information concerning the control of heavy loads near spent fuel.

The results of Task A-36 were reported in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." Our conclusion from this evaluation was that existing measures to control the handling of heavy loads at operating plants, although providing protection from certain potential problems, do not adequately cover the major causes of load-handling accidents and, therefore, should be upgraded.

In order to upgrade measures for the control of heavy loads, we developed a series of guidelines designed to achieve a two-phase objective using an accepted approach or protection philosophy. The first phase of the objective, achieved through a set of general guidelines identified in Section 5.1.1 of NUREG-0612, is to ensure that all load-handling systems at nuclear power plants are designed and operated such that their probability of failure is uniformly small and appropriate for the critical tasks in which they are employed. The second phase of our objective, achieved through the guidelines contained in Sections 5.1.2 through 5.1.5 of NUREG-0612, is to ensure that for load-handling systems in areas where their failure might result in significant consequences, either: (a) features are provided, in addition to those required for all load-handling systems, to ensure that the potential for a load drop is extremely small (e.g., a single failure-proof system); or (b) conservative evaluations of load-handling accidents indicate that the potential consequences of any load drop are acceptably small. Acceptability of accident consequences is quantified in NUREG-0612 into four accident analysis evaluation criteria as follows:

(1) Releases of radioactive material which may result from damage to spent fuel based on calculations involving the accidental dropping of a postulated heavy load, produce doses which are well within the limits specified in 10 CFR Part 100. These limits are a maximum of 300 rem to the thyroid and 25 rem whole body. (Analyses should show that doses are equal to or less than 1/4 of Part 100 limits).

(2) Damage to fuel and fuel storage racks based on calculations involving accidental dropping of a postulated heavy load, does not result in a configuration of the fuel such that K_{eff} is larger than 0.95.

(3) Damage to the reactor vessel or the spent-fuel pool based on calculations of damage following accidental dropping of a postulated heavy load, is limited so as not to result in water leakage which could uncover the fuel. (Makeup water provided to overcome leakage should be from a borated source of adequate concentration if the water being lost is borated.)

(4) Damage to equipment in redundant or dual safe shutdown paths, based on calculations assuming the accidental dropping of a postulated heavy load, will be limited so as not to result in a loss of required safe shutdown functions.

The approach used to develop our guidelines for minimizing the potential for a load drop was based on our concept of defense-in-depth. This approach includes proper operator training, equipment design and maintenance coupled with safe load paths and crane interlock devices restricting movement over critical areas. Our guidelines resulting from the foregoing considerations are tabulated in Section 5 of NUREG-0612.

1.3 Plant-Specific Background

On December 22, 1980, we issued a generic letter to all applicants for an operating license including the Detroit Edison Company, requesting that the applicants review their provisions for handling and control of heavy loads. These provisions with respect to the guidelines of NUREG-0612 were to be evaluated and certain additional information was to be submitted to be used for an independent determination of conformance to these guidelines. Detroit Edison provided its responses to this request in its letters dated December 3, 1981; April 15, 1982; October 15, 1982; and April 3, 1984.

2. EVALUATION AND RECOMMENDATIONS

2.1 Overview

The following sections summarize the applicant's review of heavy load handling at the Fermi-2 facility including our evaluation, conclusions and recommendations to the applicant for making the facilities more consistent with the intent of NUREG-0612.

2.2 Heavy Load Overhead Handling Systems

Table 2.1 presents the applicant's list of overhead handling systems which are subject to the criteria in NUREG-0612. The applicant has defined the weight of a heavy load for the Fermi-2 facility as greater than one ton which is consistent with the NUREG-0612 definition.

2.3 Guidelines

The basic guidelines of NUREG-0612 for our Phase II evaluations are summarized in the following sections and include the applicant's responses and our evaluation and recommendations. The guidelines in Sections 5.1.2 and 5.1.3 apply only to pressurized water reactors. They are not addressed for the Fermi-2 facility since this a boiling water reactor. However, we do require that the Fermi-2 facility conform with the guidelines in Sections 5.1.4, 5.1.5 and as appropriate, the alternative for upgrading in Section 5.1.6 of NUREG-0612.

2.3.1 Reactor Building (Section 5.1.4 of NUREG-0612)

The objectives stated in Section 5.1.4 of NUREG-0612 are:

(1) The reactor building crane, and associated lifting devices used for handling the heavy loads, should satisfy the single failure-proof guidelines of Section 5.1.6 of this report.

OR

(2) The effects of heavy load drops in the reactor building should be analyzed to show that the evaluation criteria of Section 5.1 of NUREG-0612 are satisfied. The postulated dropped loads which were to be analyzed should include: the shield plugs, the drywell head, the reactor vessel head; steam dryers and separators; the refueling canal plugs and gates; the shielded spent-fuel shipping casks; the vessel inspection platform; and any other heavy loads which may be brought over or near safe shutdown equipment as well as fuel in either the reactor vessel or the spent fuel pool. Credit may be taken in this analysis for operation of the standby gas treatment system if the facility Technical Specifications require its operation during periods when the load being analyzed would be handled. The analysis should also conform to the guidelines of Appendix A.

(a) Summary of the Applicant's Responses

The reactor building crane main hoist is rated at 125 tons and is single failure-proof. This is verified in Section 9.1.4.2.1 of the Fermi-2 FSAR. The original design was based on EOCI 61 and analyses have been made to verify that it also conforms with CMAA 70 and ANSI B30.2.

The auxiliary hoist supplementing the main crane is rated at 5 tons. It is not a single failure-proof design. In order to comply with this guideline, the auxiliary hoist is limited to handling a maximum load of 1500 pounds over spent fuel; this is less than the defined heavy load. The crane is equipped with a load limiting device which restricts this hoist from handling heavy loads over the spent fuel pool and the reactor pressure vessel when its head is removed. A load limit switch controlled by the operator must set the device into either a normal or a by-pass mode. In the normal position, there is a hoist limit of 2000 pounds due to a load sensing device in the control circuitry. This device prevents the lifting motors of the hoist from operating if the lift exceeds 2000 pounds. Additionally, a red light is switched on outside the hoist control cab and the operator control panel. In the switch by-pass mode, the hoist may lift its rated load which is 5 tons; the indicating lights will be green.

A maintenance procdure provides rigid administrative control for the auxiliary hoist to ensure compliance with the load restrictions, the painted barrier lines and the signs posted in the spent fuel pool and reactor area.

The other 17 hoists listed as capable of handling heavy loads over spent fuel or over safety-related equipment in the reactor building needed to achieve a safe shutdown, are designed to meet the requirements of ANSI B30.16.

In addition to these hoists, there are four associated handling systems which are also used.

(1) The RPV strongback is a 5-ton unit to facilitate handling of the 81-ton RPV head. The design review of this strongback has been conducted by the General Electric Company (GE) to determine compliance with ANSI N14.6. GE has reported that the lifting lugs meet the criteria in Section 5.1.6 of NUREG-0612. GE also states that the RPV head strongback lifting device has several components which do not meet the additional design strength criteria of Section 6.2 of ANSI N14.6. This strongback will be upgraded to meet the single failure criteria of Section 6.2 of ANSI N14.6. Modifications will be completed prior to the use of this device after initial criticality.

(2) The dryer/separator strongback is a specially built sling to handle either the 42-ton steam dryer or the 73-ton dryer separator. GE states that the strength allowances used for the design of this device does not provide adequate strength to meet the criteria in Section 6.2 of ANSI N14.6 for the maximum combined static and dynamic forces during handling of the separator in air. The dryer/separator strongback will be upgraded to meet the single failure criteria of Section 6.2 of ANSI N14.6. Modifications will be completed prior to the use of the device after initial criticality.

(3) The vessel head insulation spreader beam is a 5-ton device which connects directly to the reactor building crane main hook. It meets single failure-proof criteria. It was designed to achieve compliance with the strength requirements in ANSI N14.6 for static and dynamic loads.

(4) The spent fuel cask handling system is single failure-proof and is addressed in detail in Section 9.1.4.2.1 of the Fermi-2 FSAR. The load capacity is 100 tons and the handling system consists of a redundant sling system.

(b) <u>Evaluation</u>

The reactor building crane main hoist, the vessel head spreader beam, and spent fuel cask handling system as reported, are consistent with the guideline requirement to be single failure-proof. In its letter dated April 15, 1982, the applicant discussed the reactor building crane design in detail to show that it also has been upgraded from the original design of EOCI 61 to comply with CMAA 70.

The commitments by the applicant for resolution of the recognized deficiencies of the RPV strongback and the dryer/separator strongback will bring these devices into compliance with the guidelines of Section 5.1.1 and the guidelines in Section 5.1.4 of NUREG-0612.

The auxiliary hoist associated with the reactor building crane and the 17 monorail hoists in the reactor building are listed as capable of handling heavy loads over spent fuel or shutdown safety systems. The auxiliary hoist and monorail hoists are not single failure-proof so the second option of our guidelines cited in Section 2.3.1 of this appendix applies. The applicant used the matrix analysis technique described in Enclosure 3, (Sections 2.1-3, 2.3-2(a) and 2.3-2(b)) of our generic letter dated December 22, 1980, to identify locations involving risk from the loads of these hoists. The proposed hazard elimination categories are consistent with those which we recommend in our guidelines. We find that the special controls, the load sensor, the signal lights and the operational restrictions applied to the reactor building crane auxiliary 5-ton hoist constitutes an acceptable system which is consistent with the first alternative approach of Section 5.1 of NUREG-0612 whose objective is to show that the potential for a load drop is extremely small.

(c) Conclusions

We conclude that the reactor building crane main hoist which has been upgraded from the standards of EOCI 61 to CMAA 70; the spent fuel cask handling system; and the vessel head insulation spreader beam, all are "single failure-proof" in design and that they are consistent with the guidelines in Section 5.1.4 of NUREG-0612. We also conclude that the commitments by the applicant to upgrade the RPV strongback and the dryer/separator strongback will make them consistent with the guidelines in Section 5.1.4. We find that the risk control plans for the reactor building crane auxiliary 5-ton hoist are consistent with the first objective of the alternative approach in Section 5.1 of NUREG-0612. Finally, we find that the matrix system used by the applicant for the 17 additional hoists in the reactor building established acceptable hazard elimination categories.

2.3.2 Other Areas (Section 5.1.5 of NUREG-0612)

The objectives stated in Section 5.1.5 of NUREG-0612 are:

(1) If safe shutdown equipment are beneath of directly adjacent to a potential travel load path of the overhead handling systems (i.e., a path not restricted by any limits on crane travel or by mechanical stops or electrical interlocks), one of the following criteria should be satisfied in addition to satisfying the general guidelines of Section 5.1.1:

  (a) The crane and associated lifting devices should conform to the single failure-proof guidelines of Section 5.1.6 of this report.

  OR

  (b) If the load drop could impair the operation of equipment or cabling associated with redundant or dual safe shutdown paths, mechanical stops or electrical interlocks should be provided to prevent movement of loads in proximity to these redundant or dual safe shutdown equipment. (In this case, credit shou_ not be taken for intervening floors unless justified by analysis.)

<u>OR</u>

(c) The effects of load drops have been analyzed and the results indicate that damage to safe shutdown equipment would not preclude operation of sufficient equipment of achieve safe shutdown. Analyses should conform to the guidelines of Appendix A, as applicable.

(2) Where the safe shutdown equipment has a ceiling separating it from an overhead handling system, an alternative to Section 5.1.5(1) above would be to show by analysis that the largest postulated load handled by the handling system would not penetrate the ceiling or cause spalling that could cause failure of the safe shutdown equipment.

(a) <u>Summary of the Applicant's Statement</u>

The applicant states that the other area hoists are not single failure-proof. Therefore, the alternative option of (1)(a) cited above was used. Five hoists located in the auxiliary and RHR complex listed in Table 2.1 required review. The matrix analysis shows that the ventilation equipment room hoist in the auxiliary building has a second floor risk involving division 1 and division 2 cable trays. Their hazard elimination category is based on: "site-specific considerations eliminating the need to consider load/equipment combinations." All of the other hoist hazard elimination categories are: "system redundancy and separation preclude loss of capability of the system to perform its safety-related function following a load drop."

(b) <u>Evaluation</u>

The scope of the load drop analyses are presented in the appropriate matrix format using hazard elimination categories as recommended in Enclosure 3 to our generic letter dated December 22, 1980. These indicate that the applicant has satisfied Section 5.1.5 of NUREG-0612.

(c) <u>Conclusions</u>

We find that the coverage and applicability of the analyses used for reviewing load drops in other areas of the Fermi-2 facility are consistent with Section 5.1.5 of NUREG-0612.

2.3.3 <u>Single Failure-Proof Handling Systems (Section 5.1.6 of NUREG-0612)</u>

The objectives of Section 5.1.6 of NUREG-0612 are:

(1) Lifting Devices:

(a) <u>Special lifting devices</u> that are used for heavy loads in the area where the crane is to be upgraded should meet ANSI N14.6-1978, "Standard For Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More For Nuclear Materials," as specified in Section 5.1.1(4) of this report except that the handling device should also comply with Section 6 of ANSI N14.6-1978. If only a single lifting device is provided instead of dual devices, the special lifting device should have twice the design safety factor as required to satisfy the guidelines of Section 5.1.1(4). However, loads that have been evaluated and shown to satisfy the evaluation criteria of Section 5.1 need not have lifting devices that also comply with Section 6 of ANSI N14.6.

(b) Lifting devices that are not specially designed and that are used for handling heavy loads in the area where the crane is to be upgraded should meet ANSI B30.9-1971, "Slings," as specified in Section 5.1.1(5) of this report, except that one of the following should also be satisfied unless the effects of a drop of the particular load have been analyzed and shown to satisfy the evaluation criteria of Section 5.1:

(i) Provide dual or redundant slings or lifting devices such that a single component failure or malfunction in the sling will not result in uncontrolled lowering of the load;

OR

(ii) In selecting the proper sling, the load used should be twice what is called for in meeting Section 5.1.1(5) of this report.

(2) New cranes should be designed to meet NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants." For operating plants or plants under construction, the crane should be upgraded in accordance with the implementation guidelines of Appendix C of this report.

(3) "Interfacing lift points such as lifting lugs or cask trunions should also meet one of the following for heavy loads handled in the area where the crane is to be upgraded unless the effects of a drop of the particular load have been evaluated and shown to satisfy the evaluation criteria of Section 5.1:

(a) Provide redundancy or duality such that a single lift point failure will not result in uncontrolled lowering of the load; lift points should have a design safety factor with respect to ultimate strength of five (5) times the maximum combined concurrent static and dynamic load after taking the single lift point failure.

OR

(b) A non-redundant or non-dual lift point system should have a design safety factor of ten (10) times the maximum combined concurrent static and dynamic load."

(a) Summary of the Applicant's Statements

Special Lifting Devices (Item (1)(a))

The special lifting devices have been discussed in detail in Section 2.3.1 of this appendix, in relation to Section 5.1.4 of NUREG-0612. The loads handled by the special lifting devices are identified in Table 2.2 by reactor building load numbers 1, 4, 5, 6, 10, 11, 22 and 23.

Lifting Devices Not Specially Designed (Item (1)(b))

The loads and hoists are presented in Tables 2.2 and 2.1, respectively. Information on the loads, where they are handled, the safety-related equipment at risk, and the appropriate hazard elimination category were presented in matrix sheets. The presentations cover all hoists listed in Table 2.1. A number of hoists and their loads which are less than the defined "heavy load" for the Fermi-2 facility, have been omitted from consideration.

Special lifting devices and slings will be purchased to ensure that the requirements of ANSI N14.6-1978 and ANSI B30.9-1971 are satisfied. Existing slings used for construction will not be retained for handling of heavy loads around critical equipment after the Fermi-2 facility is operational.

The requirements of the stress design factor will include the maximum static and dynamic loads as defined in NUREG-0612. Any single failure-proof handling systems will also meet the requirements of Section 5.1.6 of NUREG-0612. All slings which fall within the areas of concern in NUREG-0612, will be clearly marked to identify their qualification for that application. This includes a load-rating which will account for static and dynamic loads for hoist speeds up to 30 ft/min. for these slings, as well as any information which might restrict certain specific slings to specific hoist/load applications.

Additional analyses for specific reactor building loads resulted in a commitment by the applicant to handle them under single failure-proof guidelines. Single failure-proof slings have been purchased for their handling. These loads are: (1) No. 13 fuel pool gates A and B; (2) Nos. 16 and 17 RWCU floor plugs and plugs at column E-1/2-10-1/2 (handling is to be by a portable gantry crane purchased for this purpose); and (3) No. 18 equipment hatch plugs.

Descriptive information on the auxiliary building ventilation equipment hoist shows that it is to be an 8-ton capacity hoist designed to meet the requirements of ANSI B30.16. The trolly beam and the trolly are presently installed. Analysis shows the floor strength below this hoist will withstand a load drop of 10 tons from the fifth to the third floor, without impacting critical systems below the third floor.

New Cranes (Item (2))

Table 2.1 which is based on the latest submittal (October 10, 1982), shows six hoists which have not yet been specified for purchase. The trolly support for these hoist locations has been installed. The specified design of these hoists will include conformance with ANSI B30.16 criteria for overhead hoists.

Future cranes or hoists which fall within our concerns in NUREG-0612 will meet the design guidelines of NUREG-0612.

At the time of the applicant's latest submittal on April 3, 1984, the reactor building main hoist was the only one which was stated to need a single failure-proof handling capability. However, any other single failure-proof handling needs which cannot be handled by the reactor building main hoist, will be resolved by upgrading an existing hoist or by procurement of a new hoist meeting the guidelines in NUREG-0554.

Interfacing Lift Points (Item (3))

The applicant's submittals do not address this subject specifically since they are included in the preceeding discussions.

(b) Evaluation

From our review of the submittals for Item (1)(a), we find that the lifting devices identified in Table 2.2 for fifth floor load No.'s 1, 4, 5, 6, 10, 11, 22 and 23 of the reactor building crane main hoist are intended to meet the criteria for single failure-proof devices.

Load No. 14 (the crane block), being an integral part of the single failure-proof crane, also meets this requirement. The commitment made by the applicant on April 3, 1984, for the reactor building crane main hoist load handling equipment upgrade for Loads No. 13, 16, 17 and 18, is also intended to bring them into compliance with the single failure-proof criteria.

The other loads (Item (1)(b)) handled by the reactor building main hoist at various floor levels and the loads handled by other hoists listed in Table 2.1, have been analyzed. The results were presented in matrix form using the format and hazard elimination category codes recommended in Section 2.3.2 of Enclosure 3 to our generic letter. We find that the applicant's statement that devices and slings will be purchased to meet the requirements of ANSI B30.9 and ANSI N14.6 and the guidelines in Section 5.1.6 of NUREG-0612, is a commitment by the applicant to provide lifting devices "not specially designed" which are consistent with our guidelines on this matter.

With respect to Item (2), the new crane design which will meet the requirements of ANSI B30.16 and the applicant's plans to meet our guidelines in NUREG-0612, are acceptable. Since the guidelines in Section 5.1.6 of NUREG-0612 for those areas required to meet the single failure-proof guideline specifies that a new crane be designed to the criteria in NUREG-0554, any new cranes for the Fermi-2 facility will be consistent with our criteria in NUREG-0612.

The omission by the applicant of a discussion on interfacing lift points (Item (3)) is acceptable since there is no alternative upgrading of cranes involved and the analysis and hazard elimination categories are consistent with our guidelines in NUREG-0612. Furthermore, we consider the applicant's statement discussed in Section 2.3.3(a) of this appendix regarding Item (1)(b), that any single failure-proof handling system will also meet the guidelines in Section 5.1.6 of NUREG-0612, is a commitment by the applicant that the system includes the slings, shackles and interface lift points required to constitute a complete handling system. Additionally, we interpret the references in the applicant's submittals to single failure-proof slings to be in accord with the commitment cited above. Accordingly, we find that the applicant's submittals are in agreement with our guidelines in Section 5.1.6(c) of NUREG-0612.

(c) Conclusions

Based on our review of the applicant's submittals, we find with respect to the guidelines in Section 5.1.6 of NUREG-0612, that:

(1) The design and commitments by the applicant for additional upgrade of special lifting devices for the Fermi-2 facility is consistent with our guidelines in Section 5.1.6(1)(a).

(2) Lifting devices not specially designed have been demonstrated by the applicant through its hazard elimination analyses and by its commitments for subsequent purchase, to be consistent with our guidelines in Section 5.1.6(a)(b).

(3) The present status of the existing cranes and the applicant's commitment regarding new cranes at the Fermi-2 facility, are consistent with our guidelines in Section 5.1.6(2).

(4) As evaluated above, appropriate interface lift points are consistent with our guidelines in Section 5.1.6(3).

3. SUMMARY OF CONCLUSIONS

3.1 Guideline Conclusions

3.1.1 The reactor building crane main hoist is single failure-proof. The special lifting devices used with it are being upgraded to provide full compliance with the criteria to achieve the objective of being single failure-proof.

The load handling control system for the reactor building crane auxiliary hoist has special electrical load sensors and operator controlled interlocks with light signals to assure that its intended uses will be safe.

The additional hoists used in the reactor building have load analysis matrix sheets presented which show hazard elimination categories for all loads and potentially affected safety-related equipment.

3.1.2 The scope and coverage of the load drop analyses used with the matrices on other area hoists, is consistent with the guidelines in Section 5.1.5 of NUREG-0612.

Special lifting devices have been shown to be consistent with Section 5.1.6(1)(a) of NUREG-0612.

Lifting devices not specially designed, are shown to be consistent with Section 5.1.6(1)(b) of NUREG-0612.

The status and the applicant's commitments with respect to new cranes, are consistent with Section 5.1.6(2) of NUREG-0612.

Interfacing lift points which are a part of single failure-proof systems, are consistent with Section 5.1.6(3) of NUREG-0612.

3.2 Recommendations

None.

3.3 Overall Summary

We find that the reported progress and the commitments made by the applicant for the control of heavy loads at the Fermi-2 facility demonstrate conformance with our guidelines in Sections 5.1.4, 5.1.5 and as appropriate 5.1.6 of NUREG-0612. On this basis, we find that the applicant has satisfactorily satisfied our requirements for Phase II.

TABLE 2.1. OVERHEAD HOISTS CAPABLE OF HANDLING HEAVY LOADS OVER SPENT FUEL OR SHUTDOWN SAFETY SYSTEM COMPONENTS

Hoist	Hoist Identification Number	Type	Capacity	Hoist Location
Reactor Building Crane Main	T3100E002	(1)	125 Ton	RB-5the Floor
Reactor Building Crane Auxiliary Hoist	T3100E002	(1)	5 Ton	RB-5the Floor
N&S Torus Hatch Hoists	T3100E032&3	(2)	5 Ton/ea	RB-1st Floor
HPCI Hoist	T3100E030	(2)	12 Ton	AB-1st Floor
RCIC Hoist	T3100E031	(2)	10 Ton	RB-1st Floor
RHR Pumps-Division I Basement Hoist	T3100E024	(2)	16 Ton	RB-Basement
RHR Pumps-Division II Basement Hoist	T3100F025	(2)	16 Ton	RB-Basement
RHR Pumps-Division I 1st Floor Hoist	T3100E026	(2)	16 Ton	RB-1st Floor
RHR Pumps-Division II 1st Floor Hoist	T3100E027	(2)	16 Ton	RB-1st Floor
N&S Recirculation Pump Hoists	T3100E015A&16A	(2)	25 Ton/ea	RB-1st Floor
MG Sets, N, C, & S Hoists	T3100E035,6&7	(2)	12 Ton/ea	RB-4the Floor
MG Set Fluid Drive N&S Hoist	T3100E038&9	(2)	20 Ton/ea	RB-4the Floor
CRD Repair Hoist	T3100E019	(2)	3 Ton	RB-3rd Floor
Core Spray Division I Hoist	T3100E028	(2)	16 Ton	RB-1st Floor
Core Spray Division II Hoist	T3100E029	(2)	16 Ton	RB-1st Floor
Diesel Generator Division I N&S Hoist	--	(2)	2 Ton/ea	RHR-Ground Floor
Diesel Generator Division II N&S Hoist	--	(2)	2 Ton/ea	RHR-Ground Floor
Diesel Generator Motor Control Central Division I N&S Hoists	--	(2)	4 Ton/ea	RHR-Upper Floor
Diesel Generator Motor Control Central Division I	--	(2)	4 Ton/ea	RHR-Upper Floor
Ventilation Equipment Room Hoist	--	(2)	8 Ton	AB-5the Floor
NE Equipment Hatch Hoist	--	(2)	12 Ton	RB-1st Floor
Portable Gantry Hoist	--	(3)	16 Ton	RB-5the Floor

a. (1) Overhead Traveling Crane, (2) Monorail Hoist.

b. RB-Reactor Building, AB-Auxiliary Building, RHR-RHR Building.

c. These hoists have not yet been specified for purchase, although the trolley support for these hoist locations has been installed. The specified design of these hoists will include conformance with ANSI B30.16 criteria for "Overhead Hoists."

TABLE 2.2 TABULATION OF HEAVY LOADS OVERHEAD HOIST; REACTOR BUILDING CRANE MAIN HOIST AREA; REACTOR BUILDING FIFTH FLOOR

Load Number	Load	Load Weight	Lifting Device
1	Drywell Head (T2301A001A)	67 Ton	Head Strongback (F1300E009)
2	Reactor Shield Plugs (6)	100 Ton/ea	3-Leg Sling,[a]
3	Reactor Pressure Vessel Service Platform (F1300E010)	6 Ton	Service Platform Lifting Device (CEX-33240A)
4	Vessel Head Insulation (B1151H001)	5 Ton	Spreader Beam,[a]
5	Reactor Pressure Vessel Head	81 Ton	Head Strongback (F1300E009)
6	Reactor Pressure Vessel Head Strongback (F1300E009)	5 Ton	Connects directly to Main Hoist Hook
7	Storage Pool Slot Plugs (4)	43 Ton/ea	2-Leg Sling,[a]
8	Fuel Pool Slot Plugs (4)	9 Ton/ea	1-Leg Sling,[a]
9	Stud Tensioner (F1300E007)	6 Ton	Connects directly to Main Hoist Hook
10	Steam Dryer (B1107D041)	42 Ton	Dryer/Separator Sling (F1300E008)
11	Steam Separator (B1112D002)	73 Ton	Dryer/Separator Sling (F1300E008)
12	Storage Pool Gate	14 Ton	2-Leg Sling,[a]
13	Fuel Pool Gates (A & B)	4.3 Ton & 2.5 Ton	2-Leg Sling,[a]
14	Crane Load Block	5 Ton	None
15	Spent Fuel Cask (F1600E001)	100 Ton	Redundant Cask Slings

TABLE 2.2 (continued)

Load Number	Load	Load Weight	Lifting Device
16	Concrete Floor Hatch (E1/2-10 Column Location)	5 Ton	4-Leg Sling,[a]
17	RWCU Demin Floor Plug	14 Ton	4-Leg Sling,[a]
18	Equipment Hatch Plugs	17 Ton	4-Leg Sling,[a]
19	Debris Shipping Cask	--[b]	--[a]
20	Fuel Storage Racks	--[b]	--[a]
21	Refueling Bridge (T2502D001)	14 Ton	4-Leg Sling,[a]
22	Dryer/Separator Sling (F1300E008)	2.4 Ton	Attaches directly to Main Hook
23	Vessel Head Insulation Spreader Beam	1.2 Ton	Attaches directly to Main Hook

OVERHEAD HOIST: REACTOR BUILDING AUXILIARY HOIST (5 TON)
AREA: REACTOR BUILDING FIFTH FLOOR

Load Number	Load	Load Weight	Lifting Device
1	Maintenance Tools	5 Ton	--[b,a]
2	Lifting Fixtures	5 Ton	--[b,a]
3	New Fuel Vault Plugs	2 1/2 Ton	4-Leg Sling,[a]

OVERHEAD HOIST: NORTH, CENTER AND SOUTH MOTOR GENERATOR SET HOISTS (3 12-TON EACH)
AREA: REACTOR BUILDING FOURTH FLOOR

Load Number	Load	Load Weight	Lifting Device
1	North and South Motor Generator Set Generator (Without Rotors) B31035001A,B	11 Ton	--[b,a]

TABLE 2.2 (continued)

Load Number	Load	Load Weight	Lifting Device
2	North and South Motor Generator Set Generator Rotors	8 Ton	--[b,a]
3	North and South Motor Generator Set Motors (Without Rotors) B31035001A,B	11 Ton	--[b,a]
4	North and South Motor Generator Set Motor Rotors	8 Ton	--[b,a]
	OVERHEAD HOIST: NORTH AND SOUTH MOTOR GENERATOR SET FLUID DRIVE HOISTS (2 20-TON EACH) AREA: REACTOR BUILDING FOURTH FLOOR		
1	North and South Motor Generator Set Fluid Drives B31035001A,B	Ton[a]	--[b,a]
	OVERHEAD HOIST: CONTROL ROD DRIVE REPAIR HOIST (3-TON) AREA: REACTOR BUILDING THIRD FLOOR		
1	CRD Transfer Cask (C1102E001)	2 Ton	--[b,a]
	OVERHEAD HOIST: NORTH AND SOUTH TORUS MATCH HOISTS (2 5-TON) AREA: REACTOR BUILDING FIRST FLOOR		
1	North and South Torus Hatches	5 Ton	--[b,a]
	OVERHEAD HOIST: RCIC HOIST (10-TON) AREA: REACTOR BUILDING FIRST FLOOR		
1	RCIC First Floor Hatch	9 Ton	--[b,a]
2	RCIC Basement Floor Hatch	9 Ton	--[b,a]

TABLE 2.2 (continued)

Load Number	Load	Load Weight	Lifting Device
3	RCIC Pump (E5101C001)	Ton	--[b,a]
4	RCIC Turbine (E5101C002)	Ton	--[b,a]
	OVERHEAD HOIST: HPCI HOIST (12-TON) AREA: REACTOR BUILDING FIRST FLOOR		
1	HPCI Floor Shield Plugs (3)	10 Ton	--[b,a]
2	HPCI Pump (E4101C001)	Ton	--
3	HPCI Turbine (E4101C2)	Ton	--[b,a]
	OVERHEAD HOIST: RECIRCULATION PUMP GEARED HOISTS (2 24-TON) AREA: REACTOR BUILDING FIRST FLOOR		
1	Recirculation Motor (2) (B3101C001A,B)	20 Ton/ea	--[b,a]
2	Recirculation Drive Mounts (2)	4 Ton/ea	--[b,a]
3	Recirculation Pump Covers (2)	2.5 Ton/ea	--[b,a]
4	Rotating Pump Assembly (2) (B3101C001A,B)	1.5 Ton/ea	--[b,a]
	OVERHEAD HOIST: RHR DIVISION 1 AND DIVISION 2 BASEMENT HOISTS (2 16-TON) AREA: REACTOR BUILDING BASEMENT		
1	North and South Floor Hatch	7 Ton	--[b,a]
2	Division 1 and 2 RHR Pumps (4) (E1102C002A-D)	2 Ton	--[b,a]
3	Division 1 and 2 RHR Motors (4) (E1101C001A-D)	2 Ton	--[b,a]

TABLE 2.2 (continued)

Load Number	Load	Load Weight	Lifting Device
	CVERHEAD HOIST: RHR DIVISION 1 AND DIVISION 2 AREA: FIRST FLOOR HOIST (2 16-TON)		
1	North and South Floor Hatch	8 Ton	--[b,a]
2	Division 1 and 2 RHR Pumps (4) (E1102C002A-D)	2 Ton	--[b,a]
3	Division 1 and 2 RHR Motors (4) (E1102C001A-D)	2 Ton	--[b,a]

TABULATION OF HEAVY LOADS

Load Number	Load	Load Weight	Lifting Device
	OVERHEAD HOIST: CORE SPRAY DIVISION 1 AND 2 HOISTS (2 16-TON) AREA: REACTOR BUILDING FIRST FLOOR		
1	Core Spray First Floor Hatch (2)	8 1/2 Ton	--[b,a]
2	Core Spray Basement Floor Hatch (2)	8 1/2 Ton	--[b,a]
3	Core Spray Pump Motors (4) E2101C00A-D	Ton	--[b,a]
	OVERHEAD HOIST: NE EQUIPMENT DOOR HOIST (12 TON) AREA: REACTOR BUILDING FIRST FLOOR		
1	NE Equipment Door T2301A001B	11.3 Ton	--[b]
	OVERHEAD HOIST: DIESEL GENERATOR MOTOR CONTROL CENTER DIVISION 1 AND 2 HOISTS (4 4-TON) AREA: RHR BUILDING GRADE FLOOR		
	--[b]	--[b]	--[b,a]

TABLE 2.2 (continued)

Load Number	Load	Load Weight	Lifting Device
	OVERHEAD HOIST: DIESEL GENERATOR NORTH AND SOUTH DIVISION 1 AND 2 HOISTS (4 2-TON) AREA: RHR BUILDING GRADE FLOOR		
1	Diesel Generator Components (i.e. cylinders, cylinder liners)	--[b]	--[b,a]
	OVERHEAD HOIST: VENTILATION EQUIPMENT HOIST (8-TON) AREA: AUXILIARY BUILDING		
	--[b]	--[b]	--[b,a]

a. Not yet purchased.

b. To be determined later.

APPENDIX Q

SAFETY EVALUATION REPORT

ON THE

CONTAINMENT PURGE AND VENT VALVE OPERABILITY

(Item II.E.4.2 of NUREG-0737)

FOR THE FERMI-2 FACILITY

The review contained in this Appendix was prepared with substantial assistance from Brookhaven National Laboratory under contract to the U.S. Nuclear Regulatory Commission.

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LIST OF TABLES

1 DESIGN REQUIREMENTS

Demonstration of operability of the containment purge and vent valves, particularly the ability of these valves to close during a design basis accident (DBA), is necessary to assure containment isolation. The limiting DBA for this review is the postulated loss-of-coolant accident (LOCA). This demonstration of operability is required by our Branch Technical Position CSB 6-4 and by Section 3.10 of the Standard Review Plan (NUREG-0800) for containment purge and vent valves which are not sealed closed during operational conditions 1, 2, 3, and 4. This matter must also be resolved in accordance with our requirements in Item II.E.4.2 of NUREG-0737.

2 DESCRIPTION OF PURGE AND VENT VALVES

The valves identified as the containment isolation valves in the purge and vent system are as follows:

Valve Number	Size (Inches)	Use	Location (Containment)
VR3-3011	24	Purge Inlet	Inside Drywell
VR3-3012	24	Purge Inlet Pressure Equalizing	Outside Drywell
VR3-3013	20	Purge Inlet	Outside Wetwell
VR3-3014	20	Purge Inlet Pressure Equalizing	Outside Wetwell
VR3-3015	20	Purge Outlet	Outside Wetwell
VR3-3016	20	Purge Outlet	Outside Wetwell
VR3-3019	6	Bypass (VR3-3016)	Outside Wetwell
VR3-3023	24	Purge Outlet	Outside Drywell
VR3-3024	24	Purge Outlet	Inside Drywell
VR3-3026	6	Bypass (VR3-3023)	Outside Drywell
V4-2060	10	Purge Inlet	Outside Drywell
V4-2061	6	Purge Inlet	Outside Wetwell
V4-2063	6	Purge Inlet Pressure Equalizing	Outside Wetwell

The valve and operator manufacturers, model numbers, and operator types are shown in Table Q-1.

The three-way solenoid valves used with the Jamesbury and Bettis pneumatic operators are manufactured by ASCO (Model Number NP 8321A6E).

Table Q-1
Valve/Operator Data

Wafer Sphere Valve Number	Size (in.)	Valve Manufacturer/ Model	Operator Type	Operator Manufacturer/ Model Number
VR3-3011	24	Jamesbury 8222-EX Mod. A	1	Limitorque SMB1-60/H3BC
VR3-3012	24	Jamesbury 8922-EX Mod. A	2	Bettis T-416-B-SR2
VR3-3013	20	Jamesbury 8922-EX Mod. A	2	Bettis T-316-B-SR1
VR3-3014	20	Jamesbury 8922-EX Mod. A	2	Bettis T-316-B-SR1
VR3-3015	20	Jamesbury 8922-EX Mod. A	2	Bettis T-316-B-SR1
VR3-3016	20	Jamesbury 8922-EX Mod. A	2	Bettis T-316-B-SR1
VR3-3019	6	Jamesbury 8126-EX Mod. B	2	Jamesbury ST-290 MS
VR3-3023	24	Jamesbury 8922-EX Mod. A	2	Bettis T-416-B-SR2
VR3-3024	24	Jamesbury 8222-EX Mod. A	1	Limitorque SMB1-60/H3BC
VR3-3026	6	Jamesbury 8126-EX Mod. B	2	Jamesbury ST-290 MS
V4-2060	10	Jamesbury 8926-EX Mod. A	2	Jamesbury ST-290 MS
V4-2061	6	Jamesbury 8926-EX Mod. A	2	Jamesbury ST-290 MS
V4-2063	6	Jamesbury 8926-EX Mod. A	2	Jamesbury ST-290 MS

1 - Electric motor operator.
2 - Pneumatic operator, air to open, spring to close.

3 DEMONSTRATION OF OPERABILITY

The applicant provided operability demonstration information for the purge and vent valves in the following letters and meetings:

Reference A - Detroit Edison letter dated October 11, 1982.

Reference B - Detroit Edison letter dated January 4, 1982.

Reference C - Detroit Edison letter dated November 18, 1981.

Reference D - Meeting, "Purge Valve Audit" at the Fermi-2 site on December 2, 1981.

Reference E - Wyle Laboratories Report, 55210 - 18-inch Jamesbury Valve Test (shown to staff at audit meeting).

Reference F - Allis Chalmer's test report VER-0209 dated December 17, 1979.

Reference G - Fermi-2 Final Safety Analysis Report, Figure 6.2-11 "Recirculation Line Break - Primary Containment Initial Pressure Transient" (handout at audit meeting).

Reference H - Jamesbury Corporation letter dated November 12, 1981, B. C. Zannini (Jamesbury) to J. Green (DECo) (handout at audit meeting).

Reference I - "Aerodynamic Model Test on Butterfly Valves," by Dr. Ing. C. Keller and Dr. Ing. F. Salzmann, published in ESCHER, Wyss News, Vol. IX, No. 1, January/March 1936.

Reference J - "Tests on Streamlines Butterfly Valves," by H. Netsch and F. Schulz published in the Engineers Digest, Vol. II, No. 8, August 1950 (from Maschinenbau and Warmewirtschaft, Vol. 4, No. 9, September 1949).

Reference K - "Supportive Data Relating to Torque Coefficient Selection for Jamesbury Wafer-Sphere Valve with 90° Elbow Directly Upstream of Valve" (handout at audit meeting).

Reference L - Allis Chalmer's letter dated April 30, 1981. R. H. Zeiders to M. Haughey (NRC) Subject: Butterfly valves for containment isolation, Allis Chalmer's Valve Division Tests.

Reference M - Detroit Edison letter dated January 4, 1981 (with attachments: (a) "Combined Loading Stress Analysis on Shaft for Purge Valves"; (b) "Seismic Qualification of 6-inch Purge Valve Based on Report JHA-76-34 (PI-2406).")

Reference N - Detroit Edison letter dated January 10, 1985 - Response to request by NRC for additional information.

Reference O - Detroit Edison letter dated March 6, 1985 - Purge Valves - Followup Information.

The applicant's approach to predicting torque loads for their containment purge and vent valves under LOCA conditions is shown in the analysis (Reference A) performed for them by the Multiple Dynamics Corporation (MDC). A constant peak containment pressure of 56 psig was assumed to act across the valve during closure. The drywell LOCA containment pressure ramps up to 48 psig after 5 seconds by which time the valve is closed. The worst case torque coefficient at a 90° valve opening was used irrespective of valve position. Valve closure time is stated to be under 2 seconds including instrumentation delay time.

The dynamic torque coefficients used for the torque analysis of the Jamesbury valves were formulated from the 18-inch Jamesbury valve test program conducted by Wyle Laboratories (Reference E). The test setup at Wyle was basically a straight pipe approach flow configuration, and the test valve was installed and tested with only the disc's flat side facing the flow. The 18-inch test valve was the same design (wafer sphere) as the Fermi-2 valves, with the same aspect ratio as the Fermi 20-inch and 24-inch valves. The inlet pressure during these tests was higher at all disc angles than the drywell pressure profile given in Reference G. Single valve operation was postulated (i.e., the redundant in-series valve is assumed to have failed open) and Jamesbury assumed a 150 percent increase in the dynamic torque coefficients developed from the Wyle tests to account for the effects of the upstream piping elbow.

The air-operated actuator torque output curves associated with the Fermi-2 valves were presented by the applicant in Reference C. From these curves, it was concluded that the operators can provide the torques necessary to close and seat the 6, 10 and 20-inch valves from their 90° (full open) initial position. These same curves indicated that the operator torque ratings are not exceeded.

Reference C also provided information concerning the sizing of the motor operator for the 24-inch valve. The information demonstrated that the operators are sized to stroke the valves under the postulated loads at a reduced voltage of 80 percent.

The structural capability of the valves is addressed by MDC for the applicant in its Report Number DECO-04-2468 contained in Reference A. Stress amplification factors are applied to the stress values determined by John Henry Associates in a previous seismic qualification report (Reference C).

The applicant in its letter dated January 10, 1985, (Reference N) verifies that the bracket bolt material for the 10 and 24-inch valves is SA-193B7. The applicant also states that the allowable stress limits were taken as the yield point of each valve component. Since the yield point in shear for structural steel is considerably lower than the yield point in tension, only 60 percent of the tensile yield stress is used as the allowable shear stress.

4 EVALUATION

The applicant committed in its letter dated January 4, 1982, (Reference B) to reorient and maintain all of the purge valves with an in-plane orientation with the shaft relative to an upstream elbow; the valves have been installed accordingly. All valve discs were oriented with the flat face upstream, with the exception of the inboard 20-inch torus isolation valves VR3-3013 and VR3-3015 on the two torus purge lines which were not changed for in-service inspection and maintenance reasons. The applicant's decision not to reorient the VR3-3013 and VR3-3015 valve discs with the flat face upstream is acceptable since these valves have a large stress margin which offsets the increase torque predicted for the configuration in which the curved disc faces upstream. In addition, the LOCA-induced dynamic torques tend to assist closure with this curved disc configuration.

MDC in their approach to predicting dynamic torque loads for the Fermi-2 containment purge and vent valves, uses a constant peak containment pressure corresponding to the containment design pressure of 56 psig as compared to the 48 psig from the LOCA containment pressure response curve at 5 seconds after LOCA initiation. In addition, the worst case dynamic torque coefficient at the 90° valve opening is applied to all increments of the valve closure angle. We agree with these assumptions and find them acceptably conservative.

Reference C provides actuator torque output curves for the air-operated actuators and information concerning the sizing of the motor operators for the

24-inch valves. Our review of this information shows that the actuators are capable of closing the valves under DBA/LOCA conditions without exceeding their structural capability.

Table 2 in the MDC stress analysis report submitted as an attachment to Reference A, compares the combined loading stresses for the critical valve components amplified for dynamic torque with the allowable shear and normal stresses. The allowable shear stress used in Table 2 of the MDC report for the 10 and 24-inch valve bracket bolt material (SA-193B7) is 0.6 of the tensile yield for the bolt material (0.6 x 105,000 = 63,000 psi). With regard to the 63,000 psi allowable shear stress, we have determined from both the AISC code and the ASME Section III code, that the allowable shear stress for the bracket bolt material should not exceed 42,000 psi. This is based on an allowable shear stress equal to 0.4 times the tensile yield stress. The calculated value for bolt shear stress contained in the applicant's submittal dated October 11, 1984, for the 24-inch valve is 44,000 psi which exceeds the 42,000 psi allowable. On this basis, we found it unacceptable. The shear stress for the 10-inch valve bracket bolts is less than the 42,000 psi allowable shear stress.

Subsequently, the applicant indicated in its letter dated March 6, 1985, (Reference O) that the methodology used by MDC to calculate the shear stresses in the bolts is overly conservative. For the 24-inch valves, the ratio of dynamic torque to static torque (static disc pressure) is nominally 2. This factor of two was used by MDC to estimate the bolt shear stress for the dynamic case by multiplying it times the combined seismic and static loads. A less conservative approach, but still acceptable, would be to estimate the bolt shear stress by combining the seismic load with twice the static load. The resulting bolt shear stress is 28,000 psi which is less than the allowable shear stress of 42,000 psi.

In Reference O, the applicant identified the bracket materials in Table 1 on page 13 of the MDC stress analysis (Reference A) for the 10-inch and 24-inch valves. The ductile iron material is grade 604010 and the carbon steel material is SA-36. These materials have yield stresses of 40,000 psi and 36,000 psi, respectively. Thus, the calculated stresses are less than the allowable stresses. We find the 6-inch, 10-inch, 20-inch and 24-inch valves have acceptable margins with regard to stresses resulting from valve closure during a DBA/LOCA.

The applicant has addressed the seismic qualification of the valves in the John Henry Associate's Report Number JHA-76-34 entitled, "Sesimic Qualification of Valves" which is contained in Reference C.

5 SUMMARY OF CONCLUSIONS

We have completed our review of the additional information submitted by the applicant concerning the operability of the 6-inch, 10-inch, 20-inch and 24-inch containment purge and vent valves for the Fermi-2 facility. We find that the additional information demonstrates the ability of the 6-inch, 10-inch, 20-inch and 24-inch valves to close against the buildup of pressure in containment in the event of a DBA/LOCA. On this basis, we find that the applicant has demonstrated the operability of the purge and vent valves for the Fermi-2 facility in the event of a design basis accident combined with seismic loads. We find, therefore, that the applicant has satisfactorily resolved Item II.E.4.2 of NUREG-0737.

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14. ABSTRACT (200 words or less)
Supplement No. 5 to the Safety Evaluation Report (SER) related to the operation of the Fermi-2 facility, provides the NRC staff's evaluation of additional information submitted by the applicant regarding the outstanding review issues identified in Supplement No. 4 to the SER dated September 1984. This supplement contains the staff's conclusion that there are no outstanding issues which must be resolved prior to issuance of a low-power operating license (i.e., less than five percent of full rated power) for the Fermi-2 facility. Supplement No. 5 to the SER also summarizes the conditions which are placed in the Fermi-2 operating license. The Fermi-2 facility is located on Lake Erie in Monroe County, almost 8 miles east-northeast of Monroe, Michigan.

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