UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 25, 2001

Mr. William T. O'Connor, Jr.
Vice President - Nuclear Generation
Detroit Edison Company
6400 North Dixie Highway
Newport, MI 48166

SUBJECT: FERMI 2 - ISSUANCE OF AMENDMENT RE: SPENT FUEL POOL RERACK (TAC NO. MA7233)

Dear Mr. O'Connor:

The Commission has issued the enclosed Amendment No. 141 to Facility Operating License No. NPF-43 for the Fermi 2 facility. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated November 19, 1999, as supplemented May 31, August 2, October 19, and November 21, 2000.

The amendment revises the TSs by changing (1) the design features description of the fuel storage equipment and configuration to allow an increase in the spent fuel pool (SFP) storage capacity and (2) the description of the high-density spent fuel racks program to clarify that the surveillance program is applicable only to racks containing Boraflex as a neutron absorber. Specifically, the amendment revises the TSs to increase the capacity of the SFP from 2,414 to 4,608 fuel assemblies.

A copy of our safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

John G. Lamb

John G. Lamb, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-341

Enclosures: 1. Amendment No. 141 to NPF-43
2. Safety Evaluation

cc w/encls: See next page

NRR-058

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DETROIT EDISON COMPANY

DOCKET NO. 50-341

FERMI 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 141
License No. NPF-43

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Detroit Edison Company (the licensee) dated November 19, 1999, as supplemented May 31, August 2, October 19, and November 21, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-43 is hereby amended to read as follows:

   Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 141 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. Detroit Edison Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M Craig

Claudia M. Craig, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: January 25, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 141

FACILITY OPERATING LICENSE NO. NPF-43

DOCKET NO. 50-341

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
4.0-2	4.0-2
5.0-19	5.0-19

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR; and

c. The following nominal center to center distances between fuel assemblies placed in the various storage rack types, as applicable

Spacing (inches)	Rack Type
6.22	High density storage racks that contain Boraflex as the neutron absorbing material
6.23	High density storage racks that contain Boral as the neutron absorbing material
11.9 x 6.6	Low density storage racks
10.5	Defective fuel assembly storage rack

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 660 ft 11.5 inches.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 4608 fuel assemblies.

5.5 Programs and Manuals

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

e. The provisions of SR 3.0.2 do not apply to the test frequencies in the Primary Containment Leakage Rate Testing Program.

f. The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

5.5.13 High Density Spent Fuel Racks

A program shall be provided, for the high density storage racks containing Boraflex as the neutron absorber, which will ensure that any unanticipated degradation of the Boraflex will be detected and will not compromise the integrity of the racks.

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 141 FACILITY OPERATING LICENSE NO. NPF-43

DETROIT EDISON COMPANY

FERMI 2

DOCKET NO. 50-341

1.0 INTRODUCTION

By application dated November 19, 1999, as supplemented May 31, August 2, October 19, and November 21, 2000, the Detroit Edison Company (DECo or the licensee) requested changes to the Technical Specifications (TSs) for Fermi 2. The proposed changes would revise the TSs by changing (1) the design features description of the fuel storage equipment and configuration to allow an increase in the spent fuel pool (SFP) storage capacity and (2) the description of the high-density spent fuel racks program to clarify that the surveillance program is applicable only to racks containing Boraflex as a neutron absorber. Specifically, the proposed amendment would revise the TSs to increase the capacity of the SFP from 2,414 to 4,608 fuel assemblies.

The May 31, August 2, October 19, and November 21, 2000, supplemental letters provided clarifying information that was within the scope of the original application and did not change the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

Fermi 2 is a boiling-water reactor which commenced commercial operation on January 23, 1988. Its current Operating License will expire in March 2025. The Fermi 2 reactor core contains 764 fuel assemblies. The SFP currently contains 14 high-density racks and four low density racks for a total storage capacity of 2,385 spent fuel assemblies (SFAs). There is an additional rack currently installed that is designed to accommodate 35 defective SFAs. There are 2,383 useable storage cells with two locations designated for the neutron absorber material surveillance program. Based upon the current inventory of SFAs and projected discharge estimates by the licensee, full core discharge capability will be lost in June 2001. At that time, the SFP inventory (both new and spent fuel) will be 1,744 assemblies. The proposed reracking of the SFP will increase the total storage capacity to 4,608 SFAs. The completed configuration represents a storage capacity increase of 2,194 SFAs. The new spent fuel storage racks will contain Boral as the active fixed neutron absorbing poison for primary reactivity control.

The proposed reracking will be accomplished in three campaigns. The first campaign will include the placement of four General Electric (GE) high density storage racks (with 763 additional storage locations; Holtec International (Holtec) Racks A, B, C1, and C2). The second campaign will consist of the removal of two existing racks (the defective fuel storage rack and four low density racks) and the installation of five new high density racks (with 630 additional

storage locations; Holtec Racks D, E, F, G, and H). The final campaign involves a significant construction effort consisting of the removal of the remaining 13 racks and the installation of 14 new racks (with 3,215 additional storage locations).

The proposed new spent fuel racks also contain two additional design features. Holtec racks E and F contain a total of 10 dual purpose cells, which are designed to store items of larger cross-sections (e.g., control blades and defective fuel containers). Each dual purpose cell can be converted into four normal fuel storage cells with the installation of a cruciform insert. Two spent fuel racks (Racks B and G) are also designed to accommodate overhead platforms with a 5-ton storage capacity. These platforms are movable and can be installed by inserting the four support legs into empty storage cells.

To accommodate this proposed modification, the following three TS changes have been proposed:

(1) TS 4.3.1.c will be modified to include the four storage rack types (i.e., high density storage racks with Boral, high density storage racks with Boraflex, low density storage racks, and defective fuel assembly storage racks) that will be in the pool,

(2) TS 4.3.3 will be modified to reflect the increase in storage capacity from 2,414 SFAs to 4,608 SFAs, and

(3) TS 5.5.13 will be modified to state that a program will be provided for the high density storage racks that contain Boraflex to ensure no unanticipated degradation.

This safety evaluation presents the results of the review of the proposed amendment in the areas of criticality, occupational radiation exposure, radioactive waste, structural integrity and adequacy, fuel-handling accidents, safe handling of heavy loads, and thermal hydraulics.

3.0 EVALUATION

3.1 Criticality

The proposed amendment by DECo will include the installation of new storage racks and the replacement of existing storage racks with high density racks in a three-phased approach. The initial phase will add up to four racks to the SFP in open spaces to increase the storage capacity to 3,146 assemblies. The second phase will remove the four GE racks, the existing defective fuel storage rack, and install five high density racks. This modification will increase the storage capacity to 3,588 assemblies. The third phase will replace the remaining 13 existing Boraflex racks with 14 new high density racks to increase the storage capacity to 4,608 assemblies. The complete configuration represents a storage capacity increase of 2,194 assemblies.

The boron for the Fermi 2 spent fuel racks is in the form of Boral, which is composed of aluminum and boron carbide. No Boraflex is used in the new design. The use of Boral as the boron containing material has been approved by the NRC staff in many earlier reviews. The Boral is fastened to the fuel cells and provides a high thermal neutron removal cross section, and has proven to be structurally sound in fuel pool applications.

The current, NRC-approved, Fermi 2 analysis approach and the TSs for the SFP and existing racks state that the reactivity status, k-effective, of the SFP shall be less than 0.95 at a 95-percent probability and confidence uncertainty level. This meets the NRC staff's reactivity requirement. The specification further indicates that this k-effective value is satisfied if the maximum k-infinity of each of the stored fuel assemblies is no greater than 1.33. The present submittal does not propose to change this analytical approach or the SFP criterion, which remains at 0.95 for both the old and new racks. The maximum k-infinity for the fuel assemblies has been reduced to 1.31. (k-infinity is calculated with an infinite array of specified, uncontrolled assemblies in a cold, 20 degrees Celsius, reactor core configuration). The fuel assembly chosen for the k-infinity and corresponding pool analyses was the GE 12 fuel assembly configuration with a 5.0 weight percent Uranium-235 (U-235) content. GE 12 was chosen because it has the highest reactivity for a given enrichment and gadolinium loading. The 5.0-percent enrichment should encompass most future loadings, but this is not a requirement since the loading will have to meet the primary k-effective and k-infinity requirements.

The nuclear design and safety analysis was done by Holtec. The criticality analyses for the high density fuel storage racks was performed with the CASMO4 code. CASMO4 is a two-dimensional multi-group transport theory code. The MCNP code, a three-dimensional transport theory code, and the NITAWL-KENO5a code, a three-dimensional code, were used for verification purposes. The 238 group SCALE cross-sections were used. These methodologies and cross sections are well known and have been accepted in past NRC reviews, including previous analyses by Holtec. The use of the two codes, MCNP and KENO5a, provides greater assurance for the analysis accuracy.

The methodologies and cross-sections have been benchmarked by Holtec (and many other groups) against a number of relevant critical experiments simulating parameters related to storage racks. These benchmark calculations have been used to develop methodology bias and uncertainty factors to be added to the nominal k-effective calculations for the racks. Holtec has also determined the potential variation of rack and fuel parameters, which are used in determining the k-effective of the rack and fuel system. These parameters include rack manufacturing tolerances, boron loading variations, Boral width tolerance variation, and cell lattice pitch variation. The variation of k-effective with these parameters (taken at a 95/95 probability/confidence level) was determined. These parameters were statistically combined with the methodology uncertainty to provide a delta k uncertainty which was added to the base k-effective calculation. This treatment of the uncertainties is in conformance with past NRC recommendations and approvals.

Holtec has also investigated abnormal conditions that might be associated with the SFP. These include (1) pool water temperature effects (reference temperature was 20 degrees Celsius, but a worst-case temperature, 4 degrees Celsius, was assumed for the investigation) (the moderator temperature reactivity coefficient is negative so that temperature increases or boiling reduce reactivity) (2) eccentric fuel positioning (the nominal analysis case with the fuel centered in the cell yields maximum reactivity), (3) dropped fuel assembly (no significant reactivity increase), and (4) rack lateral movement (no significant reactivity increase). These analyses have provided a satisfactory demonstration that possible abnormal conditions will not lead to a reactivity problem if the required k-infinity and k-effective limits are met.

For the new (unburned) fuel racks, the TSs retain the currently approved TSs k-effective limits of 0.90 if dry and 0.95 if fully flooded. There is a proposed maximum k-infinity limit of 1.31 for the fuel that can be in the racks. The introduction of a k-infinity limit is an acceptable improvement over the current new fuel rack TSs, which do not provide a k-infinity approach with such a specific criterion. Similar to the review for the SFP, this is an acceptable approach and specification.

3.1.1 Technical Specification Changes

To accommodate the proposed amendment, DECo requested to change the Fermi 2 TSs. More specifically, Section 4.3, "Fuel Storage," discusses the current storage capacity and design features of the existing and new racks, which ensure adequate design margin with respect to criticality.

Sub-Section 5.5.13, "High Density Spent Fuel Racks," is also included to clarify that the surveillance program is only applicable to racks that utilize Boraflex as neutron absorber. The NRC staff has determined that the proposed TS changes are consistent with the technical analyses provided by the licensee and the NRC staff technical evaluation described in Section 3.1 above. The NRC staff finds these changes to the TSs for Fermi 2 to be acceptable.

3.1.2 Summary of Criticality Considerations

The NRC staff reviewed the reports submitted by DECo describing the addition of fuel racks to the SFP, the criticality analyses performed and methods used and the changes to the TSs (for both the SFP and for the new fuel racks) resulting from the analyses. Based on this review, the NRC staff concludes that appropriate documentation was submitted and that the proposed changes satisfy the NRC staff positions and requirements in these areas. The criticality aspects of the spent fuel racks and the new unburned fuel racks are acceptable.

3.2 Occupational Radiation Exposure

The NRC staff has reviewed the licensee's plan for the replacement of the existing SFP storage racks at Fermi 2 with respect to occupational radiation exposure. As stated above, the licensee plans to replace the existing fuel storage racks in the SFP with 23 new high-density racks. A number of facilities have performed similar operations in the past. On the basis of the lessons learned from these operations, the licensee estimates that the proposed fuel rack installation can be performed within a radiological dose of approximately 12 person-rem. This estimate includes the radiation waste processing of the existing contaminated racks, as well as the projected dose to divers in the event they are used consistent with the licensee's contingency plan.

All of the operations involved in the fuel rack installations will utilize detailed procedures prepared with full consideration of as low as is reasonably achievable (ALARA) principles. Workers performing the SFP reracking operation will be given pre-job briefings to ensure that they are aware of their job responsibilities and the precautions associated with the job. The licensee will monitor and control work, personnel traffic, and equipment movement in the SFP area to minimize contamination and to assure that exposures are maintained ALARA. Personnel will wear protective clothing and respiratory protective equipment, if necessary. Alarming dosimeters will be used as needed to confirm exposure and dose rates, while thermal

luminescent dosimeters (TLDs) will be used to officially document the dose received. Additional personnel monitoring equipment (such as extremity TLDs or multiple TLDs) will be issued as required.

As indicated above, the licensee intends to complete the three-phase fuel rack replacement without the use of divers in the pool. Removal of existing racks and installation of the new racks are expected to be completed remotely from the surface of the pool. However, if diving is necessary, the licensee has developed a contingency plan that includes diving procedures that are compliant with Regulatory Guide 8.38, Appendix A, in terms of diver restraint, radiological monitoring, physical monitoring, and standard SFP diving operations.

Prior to any diving operations, the radioactive sources in the pool will be configured to maximize the distance and shielding of the divers. Three dimensional radiation surveys with appropriate TLD devices will be performed. In addition, the divers will be equipped with monitors to survey the work area during each dive. The licensee will utilize underwater TV cameras to maintain visual contact with the divers during all diving operations. The divers will also be physically restrained by a dive tender with a tether contained in the dive umbilical. The SFP water will be continuously filtered through the SFP purification system in order to maintain water clarity. In addition, the licensee will vacuum the SFP floor prior to initiation of the diving operation and will vacuum the pool additional times during the diving operation if it should become necessary to maintain diver doses ALARA. Each diver will be equipped with whole body and extremity dosimetry (including alarming dosimetry) with remote, above surface, readouts which will be continuously monitored by radiation protection personnel.

All items removed from the pool, as well as divers if used, will be monitored for radiation and contamination. This monitoring will be performed in isolated "bull pens" that separate the potentially contaminated areas from the rest of the refueling floor. The bull pens will minimize the possible spread of contamination, including "hot particles" (or discrete radioactive particles (DRPs)). Based on the Fermi 2 operating history and fuel integrity experience, the licensee does not anticipate any significant radiological challenges from DRPs.

The licensee assessed the radiological impact of the proposed SFP design change on areas of the plant during normal operations. Revised shielding calculations indicate that the dose rates through the east and west walls of the pool showed only a modest increase (to 0.6 mrem/hr compared to the previous maximum of 0.5 mrem/hr). The maximum dose rates in the equipment storage room, adjacent to the north wall of the pool, increased to 400 mrem/hr.

These calculations are based on the conservative assumption that all assemblies in the storage array have cooled for only 60 hours. The actual operational dose rates in this area will be dependent on the age of the fuel stored in the north end of the pool. In addition, this area is not a normally occupied room and can be controlled as a "High Radiation Area" consistent with the requirement in 10 CFR Part 20. The licensee has provided marked up radiation zoning maps from the Fermi 2 Updated Final Safety Analysis Report (UFSAR) to reflect these design changes.

3.2.1 Summary of Occupational Radiation Exposure

On the basis of our review of the proposed license amendment, the NRC staff concludes that the proposed increase in spent fuel storage capacity at Fermi 2 can be performed in a manner that will ensure that doses to the workers will be maintained ALARA. The NRC staff finds that the projected occupational radiation dose for the project of approximately 12 person-rem is in the range of doses for similar modifications at other plants and is, therefore, acceptable.

3.3 Radioactive Waste

The existing contaminated fuel storage racks will be the main source of radioactive waste for the proposed modification. They will be washed prior to being removed from the pool to remove as much contamination as possible. They will then be shipped, using a special Department of Transportation approved container, to a volume reduction facility for processing and subsequent disposal at a burial site.

In order to maintain the SFP water as clean as possible, underwater vacuuming of the SFP will be used to remove radioactive crud, sediment, and other debris generated in the rack replacement. Filters from use of this underwater vacuum system will also be a source of solid radwaste. Overall, however, the licensee does not expect that the replacement of the spent fuel storage racks in the SFP will result in a significant change in the generation of solid radwaste at Fermi 2.

The impact of the expanded fuel storage capacity on the production and release of radioactive waste during normal operations is expected to be insignificant. The level of radioactive contamination in the pool water impacts the amount of solid waste produced as pool purification system resins, as well as the liquid effluents originating from SFP water. Radioactive gasses that evolve from the surface of the pool also contribute to the plant's gaseous effluents. However, the levels of gaseous and particulate radioactivity in the pool water are dominated by the most recent reactor core offload to the SFP, not the older cooled fuel stored in the pool. Therefore, the storage of additional aged SFAs resulting from this proposed design change will have a minimal contribution to radioactivity in the pool water, and related radioactive waste generation.

3.3.1 Summary of Radioactive Waste

On the basis of our review of the proposed license amendment, the NRC staff concludes that the proposed increase in spent fuel storage capacity at Fermi 2 can be performed in a manner that will ensure that doses to the workers will be maintained ALARA and the generation of additional solid radioactive wastes will be minimized. The NRC staff, therefore, finds the proposed increase in spent fuel storage capacity at Fermi 2 to be acceptable.

3.4 Structural and Seismic Considerations

This evaluation summarizes the results of the NRC staff's review of the procedures and results of the structural analyses that were performed by the licensee to demonstrate the structural adequacy of the new SFP racks under the postulated design loads (Appendix D of Standard Review Plan (SRP) Section 3.8.4) for normal, seismic, and accident conditions at Fermi 2 (Section 6.0 of the November 19, 1999, application).

3.4.1 Spent Fuel Pool Racks

SFP racks are seismic, Category I equipment and are required to remain functional during and after a safe shutdown earthquake (SSE) under all applicable loading conditions pursuant to 10 CFR Part 50, Appendix A, Design Criterion 62. The licensee's consultant, Holtec, performed the design, fabrication, and safety analysis of the new high density SFP storage racks.

The principal construction materials for the new racks are made of American Society of Mechanical Engineers (ASME) SA240-Type 304L stainless steel. The neutron absorber material is Boral. The overall design of the new racks at Fermi 2 is similar to Holtec racks that NRC has approved for service at many other nuclear power plants. The key design criteria are based on NRC memorandum entitled "Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978, as modified by amendment dated January 18, 1979.

The key design criteria of the Fermi 2 SFP racks are described in Section 2.1 of the November 19, 1999, application. The following criteria are applicable from the structural safety point of view: (1) all new rack modules are required to be free-standing; (2) all free-standing rack modules are required to be kinematically stable (against overturning) when subjected to a seismic event, with safety factors of 1.5 and 1.1 for operating basis earthquake (OBE) and SSE conditions, respectively; (3) all primary stresses in the rack modules must satisfy the limits postulated in Section III, Subsection NF of the ASME *Boiler and Pressure Vessel Code*; (4) the spatial average bulk pool temperature is required to remain under 150 degrees following a normal refueling; for a full core off-load, it should be demonstrated that bulk pool boiling does not occur with single active failure; and (5) the ability of the reinforced concrete structure of the SFP to withstand the effects of the load combinations set forth in the Fermi 2 UFSAR must be demonstrated.

At the time of the original rack installation in the Fermi 2 SFP, the seismic evaluation of the racks was performed using single-rack (SR), three-dimensional (3-D) simulations. However, for the current SFP expansion, both SR and whole pool multi-rack (WPMR) analyses were performed to simulate the dynamic behavior of the high density rack structures. Holtec used a computer program, DYNARACK, for the dynamic analysis to demonstrate the structural adequacy of the spent fuel rack design under the earthquake loading conditions. The DYNARACK program (which can perform simultaneous simulation of all racks in the pool for the WPMR analysis) has been accepted by the NRC in previous rerack analyses for several nuclear power plants. The DYNARACK program utilizes a nonlinear analytical model consisting of inertial mass elements, spring elements, gap elements, and friction elements to simulate the three-dimensional dynamic behavior of the rack and the fuel assemblies, including the frictional and hydrodynamic effects. The DYNARACK computer code simulates the friction, impact, and other nonlinear dynamic events accurately. The code models the beam characteristics of the rack, including shear, flexibility, and torsion effects appropriately by modeling each rack as a three-dimensional structure having the support pedestals and the fuel assemblies in proper locations. The potential rattling between the fuel and storage cells is simulated by permitting the impact at any of the four facing walls followed by rebound and impact at the opposite wall. Further, the rack pedestals can lift off, or slide, to satisfy the instantaneous dynamic equilibrium

of the system throughout the seismic event. The rack structure can undergo overturning, bending, twisting, and other dynamic motion modes as dictated by the interaction between the seismic inertia, impact, friction, and fluid coupling forces. The DYNARACK code calculates the nodal forces and displacements at the nodes, and then obtains the detailed stress field in the rack elements from the calculated nodal forces.

The lateral motion of the rack due to earthquake ground motion is resisted by the pedestal-to-pool slab interface friction, and is amplified or retarded by the fluid coupling forces produced by the close position of the rack to other structures. The seismic analyses of the racks were performed utilizing the direct integration, time-history method. One set of three artificial time-histories (two horizontal and one vertical acceleration time-histories) was generated in accordance with the provisions of SRP 3.7.1. A preferred criterion for the time-history generation given in SRP 3.7.1 calls for both the response spectrum and the power spectral density (PSD) corresponding to the generated acceleration time-history to envelope their target (design basis) counterparts with only finite enveloping infractions. In response to an NRC staff question, the licensee explained in the May 31, 2000, supplement letter that the target (design basis) response spectra were obtained from Section 3.7 of the Fermi 2 UFSAR. In particular, the response spectra for the fifth floor elevation 684 feet - 6 inches (which is 38 feet - 9 inches above the SFP floor slab) were conservatively used to generate bounding acceleration time histories. The licensee generated the time-histories to satisfy the preferred criterion stated above. This procedure is acceptable to the NRC staff.

The licensee considered the applicable loads and their combinations in the seismic analysis of the rack modules and performed parametric simulations for both the SR and WPMR analyses. The parameters, which were varied in the various computer runs, consisted of the rack/pool interface coefficient of friction, the extent of storage locations occupied by spent fuel (ranging from nearly empty to full), and the type of seismic input (SSE or OBE). For the parametric simulations, the licensee performed a total of 14 3-D SR model analyses and 18 WPMR model analyses. The results of these analyses show the maximum rack displacement to be 0.83 inches (for a WPMR analysis under the SSE condition). For this case, a rack-overturning evaluation indicated the factor of safety against overturning to be 45, which is much higher than the prescribed limit of 1.1 for the SSE condition. These results show that there are large safety margins against overturning of the racks as evidenced by the small rack movements and, thereby, the structural integrity and stability of the racks and fuel assemblies are maintained.

The licensee computed the maximum values of pedestal vertical forces, pedestal friction forces (i.e., horizontal loads), pedestal thread shear stresses, rack displacements, and rack stress factors. Using these data, the licensee performed the rack impact evaluation, as well as the stress limit evaluation of the rack structure, satisfying ASME Code, Section III, Subsection NF, for normal and upset conditions (Level A or Level B), and Section F-1334 (ASME Section III, Appendix F) for Level D condition. The calculated results show that there are no rack-to-wall impacts, and no rack-to-rack impacts at the top of the rack or baseplate during any of the seismic events. The results, however, indicated that there was fuel-to-cell wall impacts. The licensee's evaluation of the effect of such impacts showed that the impacts did not affect the structural integrity of the cell walls, since the limiting impact load (2151 lbf, including a safety factor of 2.0) is much greater than the highest calculated impact load (of 449 lbf).

The licensee calculated the weld stresses at the rack connections (e.g., baseplate-to-cell welds, and baseplate-to-pedestal welds, cell-to-cell welds) under the SSE and OBE loading conditions and found that all the calculated weld stresses are well below the corresponding allowable stresses specified in ASME Code, Section III, Subsection NF, indicating that the weld connection design of the rack is adequate.

In summary, the licensee's parametric study (e.g., varying coefficients of friction, different geometries and fuel loading conditions of the rack) involving both SR and WPMR analyses showed that (1) all stresses are well below their corresponding "NF" limits, (2) there are no rack-to-wall or rack-to-rack impacts, and that (3) the rack overturning is not a concern. Therefore, the NRC staff concludes that the rack modules will perform their safety function and maintain their structural integrity under postulated loading conditions and are, therefore, acceptable.

3.4.2 Spent Fuel Pool Structure

The SFP is a safety-related, seismic Category I, reinforced concrete (RC) structure, supported by a two-way RC slab. The inside (plan) dimensions of the SFP are 34 feet - 0 inches by 40 feet - 0 inches and the pool depth is 39 feet - 1 inches. The minimum thickness of the slab is 72 inches. The east and the west walls of the SFP are 6-feet-thick, and the north wall thickness is reduced from 72 inches to 48 inches above elevation 659 feet - 6 inches, where the new fuel storage pool is located. The south wall of the SFP is an integral part of the concrete reactor shield and it has a minimum thickness of 4 feet. In response to an NRC staff question related to the pool liner, the licensee stated that the pool liner consists of an array of 1/4-inch stainless steel plates. The plates are spliced together by 1-1/2 inch by 1/2 inch rectangular bars, which also provide the backing surface for the liner seam welds. The liner anchorage consists of an array of 3/8-inch diameter by 4-inch long bolts, which are fastened to the rectangular bar. These bolts are embedded in the concrete on 12-inch spacing along the plate splices.

The pool structure was analyzed using the finite element computer program, ANSYS, and the results for individual load components were combined using factored load combinations per SRP 3.8.4, and American Concrete Institute (ACI) 349-85. In addition to the dead and live loads, the analysis considered the seismic, thermal, and hydrodynamic loadings. Tables 8.4 and 8.5 in the November 19, 1999, application show the minimum safety factors for the bending strength evaluation and shear strength evaluation of the slab and walls. These tables show all the predicted safety factors to be greater than 1.0, thus demonstrating the structural integrity of the SFP under the increased loads due to the additional racks.

The NRC staff has reviewed the effect on the SFP structure of the increase of 15 degrees above the allowable value of 150 degrees in the maximum pool water temperature. The NRC staff notes that the higher pool water temperature slightly reduces the safety factors for bending of the SFP concrete pool slab, but the reduced safety factors are still within acceptable limits. Also, the staff noted that the safety factors for bending in the north wall lower portion actually increases slightly because the higher temperature increases the compressive stresses in the reinforced concrete walls.

The NRC staff further agrees with the licensee's judgment that no reduction in concrete design strength was required due to the temporary and short-lived phenomenon of the water temperature increase of 15 degrees above the allowable value of 150 degrees. Such temporary increases in pool water temperature have been previously approved by the NRC staff in several nuclear plant sites.

The NRC staff has reviewed the licensee's analytical procedures and the summary of the results, and concluded that the licensee's structural analysis demonstrates the structural integrity of the SFP structure under full fuel loading and SSE loading conditions. Thus, the SFP design is acceptable.

3.4.3 Fuel-Handling Accidents

The systems aspects of the fuel handling accidents are described elsewhere in the safety evaluation prepared by the NRC staff. The structural aspects of accidental fuel drop events only are discussed here. Several fuel-handling-accident cases were evaluated by the licensee: (1) one case for the drop of a fuel assembly (with its handling tool) impacting the top of a rack ("shallow drop" scenario), and (2) two cases for the drop of a fuel assembly (with its handling tool) falling through an empty storage cell and impacting the rack baseplate ("deep drop" scenarios). The licensee also analyzed another type of fuel drop event involving the dropping of a rack from the top of the water level in the pool and hitting the liner plate.

The "shallow drop" event was analyzed by the finite element method, which predicted localized plastic deformation at the top of the impacted region. The maximum depth of this plastic deformation is limited to 14 inches, which is below the design limit of 19 inches.

The "deep drop" event wherein the impact region is located above the support pedestal produces a maximum stress of 25 thousand pounds per square inch (ksi) in the liner which is less than the failure limit of 71 ksi for the liner material. However, the maximum compressive stress of 8.3 ksi produced in the concrete slab is greater than the concrete compressive strength of 5.9 ksi. In response to an NRC staff question addressing this issue, the licensee cited supporting technical references in the May 31, 2000, supplemental letter to show that the assumption of 5.9 ksi as the compressive strength of concrete is conservative for a dynamic event such as the "deep drop". Furthermore, its analysis using this conservative assumption indicated that this high stress region is limited to a circular area of the concrete slab less than 4.0 inches in diameter and that the estimated depth of the localized concrete crushing is 0.032 inches which, in the judgment of the NRC staff, will not endanger the structural integrity of the 72-inch-thick concrete slab.

The second "deep drop" event through an exterior cell produces some deformation of the baseplate and localized severing of the baseplate/cell wall welds. However, the licensee's analysis indicates that this "deep drop" event lowers the fuel assembly support surface by a maximum of 1.37 inches, which is less than the distance of 5.5 inches from the baseplate to the liner.

In response to an NRC staff question related to the heaviest rack drop accident, the licensee responded that the liner plate deforms plastically during such an event, and that the maximum vertical deflection of the liner plate is 0.641 inches. The licensee further stated that the maximum calculated stress of 46.8 ksi in the liner plate is less than the failure stress of 71 ksi, thus indicating that the plastically deformed liner transfers the impact load to the concrete slab through the contact interface between the liner and the concrete slab without breaching the liner.

The NRC staff reviewed the licensee's fuel drop analysis results and concurs with its findings that the postulated fuel drop accident events produce only localized damage well within the design limits for the racks.

3.4.4 Summary of Structural and Seismic Considerations

Based on the review and evaluation of the licensee's November 19, 1999, application and May 31, 2000, supplemental letter, the NRC staff concludes that the structural analyses of the spent fuel storage rack modules and the SFP structure are in compliance with the acceptance criteria set forth in the UFSAR and are consistent with the current licensing basis.

3.5 Radiological Consequences of Fuel-Handling Accidents

The NRC staff reviewed the radiological consequences of the proposed changes with respect to the design-basis fuel-handling accident. In the submittal, the licensee stated that a rack drop involving radiological consequences is precluded since all rack movement during the removal and installation phases will follow safe load paths that prevent heavy loads from being transported over the stored spent fuel. Any movement of fuel assemblies required to support the modification will be performed in the same manner and under the same administrative controls as normal refueling operations that are currently performed. Additionally, the licensee's analysis of the fuel-handling accident event shows that the racks remain intact and the resulting fuel damage is within levels previously determined. Therefore, the proposed reracking of the SFP will not affect any of the assumptions or inputs used in evaluating the dose consequences of the design-basis fuel-handling accident in the SFP. The radiological consequences of the current design-basis fuel-handling accident remain bounding.

The licensee performed an analysis of the drop of a pool gate into racks containing irradiated fuel assemblies. The analysis demonstrates that the number of fuel rods damaged (81) by a dropped pool gate is below the Fermi 2 fuel-handling accident assumption of 140 rods. The licensee's analysis also showed the kinetic energy associated with the drop of the heaviest overhead platform is enveloped by the kinetic energy associated with the gate drop. Therefore, the potential structural damage to fuel would be bounded by the results for the gate drop. Since the resulting fuel damage for these events is bounded by the fuel damage associated with the current design-basis fuel-handling accident, the radiological consequences are also bounded.

3.5.1 Summary of Radiological Consequences of Fuel-Handling Accidents

The NRC staff has determined that the radiological consequences of the current design-basis fuel-handling accident remain bounding for the proposed changes. Therefore, the NRC staff finds the proposed installation of spent fuel racks at Fermi 2 to be acceptable with regard to potential radiological consequences of a design-basis fuel-handling accident in the SFP.

3.6 Safe Handling of Heavy Loads

In support of the NRC Office of Nuclear Reactor Regulation, Brookhaven National Laboratory (BNL) performed an evaluation of the licensee's application and its supplements relative to the control of heavy loads.

3.6.1 Background

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," dated July 1980, provides guidelines and recommendations to assure the safe handling of heavy loads by prohibiting, to the extent practicable, heavy load travel over stored SFAs, fuel in the reactor core, safety-related equipment, and equipment needed for decay heat removal. The NUREG defines a heavy load as any load carried in a given area during the operation of the plant that weighs more than the combined weight of a single SFA and its associated handling tool.

Phase I of NUREG-0612 provides guidelines for reducing the likelihood of dropping heavy loads and limiting the resulting potential consequence of a drop. The guidelines are focused on establishing safe load paths, procedures for load handling operations, training of crane operators, the design of lifting devices and the design, testing, inspection, and maintenance of cranes. Phase II of NUREG-0612 provides guidelines for mitigating the consequences of dropped loads, including the use of single-failure proof cranes, use of electrical interlocks and mechanical stops to restrict crane travel, or performance of load drop and consequence analyses to assess the impact of dropped loads on plant safety. Generic Letter (GL) 85-11, "Completion of Phase II of Control of Heavy Loads at Nuclear Power Plants, NUREG-0612," dated June 25, 1985, dismissed the need for licensees to implement the requirements of NUREG-0612, Phase II. However, GL 85-11 encouraged licensees to implement actions they perceive to be appropriate to provide adequate safety. Based upon specific instances of heavy load handling concerns, the NRC requested licensees, in NRC Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor, or Over Safety-Related Equipment," to provide specific information detailing their extent of compliance with these guidelines. In response to this request, the licensee stated by letter dated May 10, 1996, that with one exception, the lifting equipment and associated administrative controls were in compliance with the provisions of NUREG-0612. The one discrepancy identified involved the use of the reactor building overhead crane (RBOC) auxiliary hoist, which was not single-failure proof, for lifts between 2000 and 5000 lbs. over the reactor building equipment hatch. Because of this, the licensee committed to using the approved main (single-failure proof) RBOC hoist for these lifts as well.

This section evaluates the licensee's compliance with these guidelines relating to the safe handling of heavy loads.

3.6.2 Evaluation

3.6.2.1 Hoisting System

NUREG-0612 recommends that when licensees handle heavy loads in the proximity of safe shutdown equipment or irradiated fuel in the SFP, specific actions be implemented to minimize the potential for an accidental drop. These actions include: the use of cranes and special lifting devices that are inspected, tested, and maintained to specific guidelines; the development of specific procedures to cover the load handling operations; and the use of trained and qualified crane operators and other personnel.

The new spent fuel modules will be delivered to the first floor of the reactor building. As stated by the licensee, the maximum rack weight for the proposed new high density racks is 37,905 lbs. (Rack B, Campaign I). The 117-ton single-failure proof cask handling crane (same as the RBOC main hoist) will be used to lift the racks to the refueling deck. The handling and installation of the racks will be done with the cask handling crane (RBOC). As stated by the licensee, the cask handling crane has been designed, fabricated, and qualified in accordance with the guidelines of Sections 5.1.1(6) and (7) of NUREG-0612, the American National Standard Institute (ANSI) Standard B30.2-1976, and the Crane Manufacturers Association of America Specification CMMA-70.

The 5-ton auxiliary hook of the RBOC may be utilized for the removal of the lift rig following rack installation, or handling the long-handled rack leveling tool, which is used for final rack leveling following installation. The licensee has also discussed the use of an additional temporary hoist in conjunction with the lift rig, which may be used during the installation process. Though the licensee states that the use of the main hook should preclude the need for this arrangement, the use of temporary cranes is permissible per NUREG-0612, provided the design is redundant or rated for twice the load (static plus dynamic). The licensee states that the temporary hoist will comply with the latter provision, and be rated for a minimum of 37.5 metric tons (37,500 kilograms or 82,673.25 pounds mass or 41.34 tons).

The licensee states that the remotely engageable lifting rig complies with all the provisions of Section 5.1.6 of NUREG-0612 and ANSI 14.6-1978, and is similar to that used at numerous other plants. The lift rig consists of independently loaded lift rods which engage the underside of the spent fuel rack baseplate. The redundancy provided ensures that a failure of one lift rig will not result in the uncontrolled lowering of the rack module. In order to address the safe lifting of the asymmetric racks, the licensee states that large turnbuckles will be used in each of the four loadpaths leading from the four lift rig eyepads. These will aid in leveling the hanging racks in which the lift rig is offset from the racks' center of gravity.

NUREG-0612 recommends that licensees provide an adequate defense-in-depth approach to maintain safety during the handling of heavy loads near spent fuel and cites four major causes of accidents: (1) operator errors, (2) rigging failures, (3) lack of adequate inspection, and (4) inadequate procedures. The licensee plans to implement measures using procedures and administrative controls in each of these areas. The licensee states that the training of crane operators will be in accordance with Chapter 2-3 of ANSI B30.2-1976 as well as specialized training provided by the rack vendor. Additional licensee personnel who will provide hand signals to the crane operator during rack movement and others who will serve as spotters will be similarly trained. As previously discussed, the specially-designed redundant lifting rig

complies with all the provisions of ANSI 14.6-1978. The spent fuel cask handling crane is tested, maintained, and inspected in a manner which satisfies Chapter 2-2 of ANSI B30.2-1976. The licensee states that a complete inspection and preventive maintenance program will be performed on the cranes prior to the start of the rack installation, as defined in Fermi 2 plant procedures. This includes inspections of the hook surfaces and wire ropes for wear or damage. The licensee plans to implement a multitude of procedures to cover the entire rack installation process (e.g., mobilization; rack handling, upending, lifting, installation, verticality, alignment; site safety; and ALARA compliance).

As previously discussed, the maximum weight of the proposed new spent fuel rack modules is 37,905 lbs. (Holtec 19x19 Rack B). The licensee has evaluated the impact of dropping the rack to the liner. Though such an accident is not a postulated event (due to the defense-in-depth approach per NUREG-0612), the analysis was shown to be acceptable since there was no compromise of the integrity of the liner, and no loss of SFP coolant. Since the liner integrity is maintained, and significant loss of coolant will not occur, the licensee concluded that a rack drop is not a new kind of accident. The proposed reracking at Fermi 2 involves the replacement of all existing racks to increase the spent fuel storage capacity, and does not utilize any new or unproven technology that has not been used successfully at other plants.

Based upon this evaluation, BNL believes that the cask handling crane, coupled with the lifting rig and other lifting devices~~, will~~ enable the licensee to handle the ~~heavy~~ loads with little or no risk to the safety of the proposed reracking operation. Additionally, BNL believes that the licensee's proposed personnel training, equipment inspections, use of redundant lift rigs, and procedural controls provides adequate defense-in-depth to maintain safety during the removal of the existing racks and installation of the new spent fuel racks.

3.6.2.2 Safe Load Paths and Load Handling Accident Analysis

In addition to the guidelines discussed above, NUREG-0612 also discusses the identification of safe load paths for the movement of heavy loads to minimize the potential for heavy loads, if dropped, to impact irradiated fuel in the SFP, or to impact safe shutdown equipment. The licensee states that movements of both the old racks being removed and the new racks to be installed will be designed to minimize potential impact on safe shutdown equipment. Once the racks are raised to the refuel floor, the load paths will follow the safe load paths currently defined for the movement of a spent fuel cask. Because both the new and old racks are located throughout the pool, entry and exit points for both will be over the north, south, or west pool walls. For the racks being removed from the pool, the licensee states that they will be moved to a point along one of the desired pool walls, raised from the pool, and moved to a point on the refuel floor on the west side of the pool to align it with the established safe load path leading to the hatchway.

The licensee states that spent fuel shuffles will be performed in independent phases of the rerack effort in order to transfer spent fuel from the old racks to the new racks in accordance with Fermi 2 TSs. The licensee states that the new racks will not be carried over any region of the pool containing fuel, and that crane stop blocks will be temporarily installed to prevent any such movement. Rack upending will be performed in an area away from the SFP, and will not overlap any safety-related component. For rack movements along the pool floor, the height of the rack above the liner will not exceed 6 inches, except for areas which may have projections that obstruct the path.

As discussed previously, the licensee states the proposed reracking does not present a new or different kind of accident which was not previously analyzed. The licensee states that the removal of the old racks and the installation of the new racks will not traverse any safety-related equipment or stored spent fuel. Because of this, no previously unanalyzed event is postulated to occur that would result in a fuel configuration change, fuel release, or compromise the pool structure leading to the loss of the coolant.

The licensee also evaluated the drop of a pool gate onto a rack, which was considered to be an unlikely event because the new racks will not be placed directly beneath the gates. This analysis modeled the drop of the 9,500 lb. gate onto racks containing irradiated fuel assemblies, estimated that 81 fuels rods would be damaged, and concluded that this remains below the Fermi 2 fuel-handling accident design basis of 140 fuel rods.

As previously discussed, two of the proposed new racks will be designed to accommodate overhead storage platforms. Similar "Holtec Overhead Platforms (HOPs)" have been licensed for use at the Nine Mile Point Nuclear Plant. The licensee evaluated the consequences of dropping the heaviest platform (1,460 lbs.) on irradiated fuel assemblies, and concluded that this event is also bounded by the gate drop analysis. The licensee conservatively assumed that the impact would be sustained by one irradiated fuel assembly with no resulting fuel rod rupture. The licensee states that the placement or removal of items from the HOPs (maximum stored weight 5 tons dry) will be governed by Fermi 2 work control procedures as well as hoisting, rigging, and load handling procedures. These procedures require an engineering analysis when loads are transported over spent fuel. These procedures also require that the work package governing the work activity contain a plant impact statement (which addresses the impact on plant systems, structures, and components), supporting data, and drawings.

BNL concurs with the licensee's evaluation that the proposed reracking will result in little or no risk to the safety of the proposed reracking operation, and does not present any new or unanalyzed accident scenario which could result in damage to the SFP structure or stored spent fuel. Additionally, BNL believes that through the implementation of crane stop blocks and procedural controls, the safe load paths to be used during the removal of the existing racks and the installation of the new racks will adequately minimize the potential for damage to the stored spent fuel or safety-related equipment. In addition, BNL believes that the licensee's proposed work control procedures associated with the placement and use of the overhead platforms will result in no significant risk to the safe storage of the spent fuel.

3.6.2.3 Fuel Handling Considerations

The licensee has evaluated the potential of an accident involving the dropping of an SFA associated with this proposed reracking. As stated by the licensee, the probability of such an accident is not significantly increased by the proposed operations. The probability of a dropped or misplaced fuel assembly is primarily influenced by the methods used to grapple and move the fuel. The methods used to move the fuel will not be changed by the proposed reracking, and will utilize the same equipment and plant procedures. Spent fuel shipping cask movements will not be permitted during the proposed reracking. Accordingly, the licensee has concluded that the proposed reracking will not involve a significant increase in the probability of an accident involving the spent fuel.

The amount of fuel shuffling to be performed during the reracking process will be greater compared to leaving the pool in its current configuration. However, in the event the new racks were not installed, the licensee would have to take alternate actions to provide sufficient spent fuel storage once the current capacity of the SFP was attained. These actions (e.g., use of spent fuel casks) would require approximately the equivalent amount of spent fuel handling plus the additional heavy load handling associated with the spent fuel casks.

Since the proposed reracking will maximize the available space in the pool, the main hook of the refueling bridge will not access the storage cells on the easternmost rows of cells along the east wall (total of 63 storage cells). As stated by the licensee, spent fuel bundles will be inserted in these cells by using an offset tool in conjunction with the refuel bridge, or the refuel bridge monorail-mounted hook will be used in accordance with plant procedures.

BNL concurs with the licensee's evaluation that the increase in spent fuel handling associated with the proposed reracking will not result in an increase in the probability of an accidental fuel drop. All fuel handling will be performed in accordance with the current fuel handling methods, use the same fuel handling equipment, and the same procedures. In addition, the installation of the proposed new spent fuel racks will preclude the need to handle additional spent fuel casks that may have been required to provide adequate on-site spent fuel storage.

3.6.2.4 Summary of Safe Handling of Heavy Loads

Based on the NRC staff's review of the licensee's application, its supplements, and the BNL evaluation, the NRC staff concludes that the control of heavy load aspects associated with the proposed changes to the TSs to increase the spent fuel storage capacity at Fermi 2 are in accordance with NUREG-0612, GL 85-11, and NRC Bulletin 96-02. Compliance with the specified administrative controls and procedures will result in the safe handling of heavy loads associated with this effort. These changes will enable the licensee to maximize the storage capacity of the SFP while not increasing the likelihood of damage to existing stored spent fuel and the pool and plant structures.

3.7 Thermal-Hydraulic Considerations

In support of the NRC Office of Nuclear Reactor Regulation, BNL performed an evaluation of the licensee's submittals relative to the thermal hydraulic analyses of the SFP.

3.7.1 Background

NUREG-0800, "Standard Review Plan" (SRP) provides criteria related to the design and performance of the spent fuel pool. Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," provides methods acceptable for the licensee to implement General Design Criteria 61 of Appendix A to 10 CFR Part 50 which requires that fuel storage and handling systems be designed to assure adequate safety under normal and postulated accident conditions.

This section documents compliance with the thermal hydraulic guidelines relating to the proposed increase in spent fuel storage capacity. This evaluation is performed for the SFP configuration after the final campaign, which represents the maximized configuration, and thereby bounds all intermediate configurations.

3.7.2 Evaluation

3.7.2.1 Spent Fuel Pool Cooling System

The fuel pool cooling and cleanup system (FPCCS) at Fermi 2 cools the SFP by transferring decay heat through heat exchangers to the reactor building closed cooling water system (RBCCWS). The FPCCS is composed of two trains, each containing one fuel pool cooling pump and one heat exchanger. The FPCCS heat exchangers are shell and tube units; the hot water from the SFP is sent to the tube side and the cold cooling water is supplied to the shell side of the heat exchangers from the RBCCWS. Under specific plant and system conditions, backup cooling is provided to the SFP by the residual heat removal (RHR) system. In this configuration, supplemental cooling is provided to the SFP by means of a permanently piped cross-tie to the RHR system. The cross-tie piping and the necessary FPCCS are Seismic Category 1. In this mode of operation, one RHR pump and the corresponding RHR division heat exchanger will provide the means to cool the SFP. The cold tube side flow to the RHR heat exchanger is supplied from the RHR service water (RHRSW) system and the shell side water is supplied from the SFP.

According to the submittals provided by the licensee, two trains of the FPCCS provide cooling for the SFP until the water temperature exceeds the high temperature alarm set point. The alarm set point is 130° F. Should the SFP pool temperature exceed the alarm set point, the operators will take action based on alarm response Plant Operating Procedure 2D9, and align one division of RHR to cool the SFP. The specific prerequisites and steps needed to align one division of RHR to the SFP cooling system are provided by Plant Operating Procedure 23.205 for the RHR system.

Table 1 shows the heat removal capabilities of the FPCCS and RHR from the SFP for various configurations with a SFP temperature of 125° F. The cold shell side water inlet temperature from RBCCWS for the FPCCS heat exchangers is assumed to be 95° F, and the cold tube side water inlet temperature from RHRSW for the RHR heat exchanger is assumed to be 89° F.

Table 1 SFP Heat Removal Capabilities

System Configuration	Heat Removal Capability (10^6 Btu/hr)
1 train of FPCCS (1 FPCCS pump and 1 FPCCS heat exchanger)	4.56
2 trains of FPCCS (2 FPCCS pumps and 2 FPCCS heat exchangers)	9.12
2 trains of FPCCS and one division of RHR (2 FPCCS pump and 2 FPCCS heat exchangers, and 1 RHR pump and 1 RHR heat exchanger)	39.84

Since the proposed increase in SFP storage capacity would result in the increase of SFP heat load for all discharge scenarios, the licensee reevaluated the effects of the increased SFP storage capacity on the SFP heat loads and temperatures.

Three basic discharge scenarios were postulated for bulk pool thermal-hydraulic evaluation:

(1) A normal partial core discharge (260 fuel assemblies). The minimum decay time of the previously discharged fuel assemblies is 18 months.

(2) A full core discharge (764 fuel assemblies). The minimum decay time of the previously discharged fuel assemblies is 18 months.

(3) An emergency full core discharge (764 fuel assemblies). The minimum decay time of the previously discharged fuel assemblies is 12 months.

In all of above scenarios, the minimum in-core hold time was 60 hours, and two trains of FPCCS were assumed to be available. Furthermore, supplemental RHR cooling was available when the bulk pool temperature exceeded the high temperature alarm set point of 130°F.
In addition to the above basic discharge scenarios, the licensee also performed supplemental analyses of two other discharge scenarios in conformance with the recommendations of the NRC SRP guidelines. These additional scenarios were:

(4) A normal partial core discharge assuming a single failure as recommended by NRC SRP 9.1.3, Section III.1.d. The worst-case single active cooling system failure would be the loss of a division of the RHR system aligned to cool the pool.

(5) An emergency (unplanned) full core discharge with minimum decay time of 36 days as recommended by SRP 9.1.3, Section III.1.h. No single failure was assumed.

The licensee calculated the decay heat load from previously discharged fuel assemblies using Holtec's QA validated DECAY program, and the transient decay heat loads and pool bulk temperatures using Holtec's QA validated ONEPOOL program. Both programs incorporated NRC Technical Position ASB 9-2 methodology for the decay heat calculations.

The licensee applied the following conservatisms in evaluating the decay heat load to the SFP and the maximum SFP bulk temperature:

(1) All analyses were performed for the SFP configuration after the final campaign, which has the largest number of fuel storage locations, the highest decay heat load, and lowest SFP thermal capacity.

(2) The decay heat load was based on a discharge schedule with bounding parameters (maximum irradiation time and batch size) for all projected discharges.

(3) The decay heat calculations were performed for a fuel inventory that slightly exceeds the maximum possible pool storage capacity.

(4) The thermal inertia (thermal capacity) of the SFP was based on the net SFP water volume only. This conservatively neglected the thermal inertia of the fuel assemblies, fuel racks, and pool structures.

(5) The cooling effect of evaporation heat losses and all other passive heat removal mechanisms (i.e., conduction through the wall and slab) were neglected.

(6) Design temperatures were used for the coolant water inlet to the FPCCS and RHR system heat exchangers.

(7) The thermal performance of all cooling system heat exchangers was determined incorporating a 5-percent tube plugging allowance.

(8) The once-burned fuel assemblies for full core discharge scenarios are conservatively assumed as twice-burned, thereby increasing their decay heat generation rate.

BNL and the NRC staff concur that the methodology and assumptions the licensee used to calculate the decay heat loads meet the intent of the NRC guidelines. The coincident net decay heat load for each scenario estimated by the licensee is given in Table 2.

Table 2 Net Decay Heat Loads

Scenario Number	Discharge Scenario	Coincident Net Decay Heat Load (10^6 Btu/hr)
1	Normal partial core discharge without a single failure	12.20
2	Full core discharge	41.84
3	Emergency full core discharge with 12 months decay	42.37
4	Normal partial core discharge with a single failure	12.20
5	Emergency full core discharge with minimum decay time of 36 days	43.25

The licensee solved the differential equations representing the transient heat balance and the thermal response of the SFP, using the Holtec QA validated computer program ONEPOOL, to obtain the bulk pool temperature. This program utilizes the above data on heat removal capability of the heat exchangers and the heat exchanger geometric data as well as the temperature effectiveness values estimated for the heat exchangers. The assumptions discussed above were also incorporated into the model. BNL and the NRC staff concur that the methodology and assumptions the licensee used to calculate the SFP bulk temperatures meet the intent of the NRC guidelines. Table 3 shows the maximum pool bulk temperature calculated for each scenario by the licensee.

Table 3 Maximum Pool Bulk Temperature

Scenario Number	Discharge Scenario	Maximum Pool Bulk Temperature (°F)
1	Normal partial core discharge without a single failure	125.4
2	Full core discharge	140.6
3	Emergency full core discharge with 12 months decay	141.1
4	Normal partial core discharge with a single failure	165.0
5	Emergency full core discharge with minimum decay time of 36 days	142.0

Based on the review of the methodology and the information provided by the licensee in the submittals, BNL finds that the peak SFP temperatures for all scenarios have been properly calculated. As shown in the above Table 3, the peak bulk temperatures of the SFP remain below 150 degrees for all scenarios except Scenario 4. The peak SFP temperature is 165 degrees for Scenario 4, which is a planned (normal)[1] partial core discharge postulating a single failure[2]. The NRC staff's evaluation of the effects of the peak SFP calculated temperature of 165 degrees on the SFP structures is addressed in Section 3.4.2 above.

3.7.2.2 Effect of Spent Fuel Pool Boiling

In the unlikely event that there is a complete loss of cooling, the SFP water temperature will begin to rise and eventually reach the boiling temperature.

The licensee performed an analysis to determine the minimum time-to-boil and the maximum boil-off rate. The calculated minimum time from the loss-of-pool cooling at peak pool water temperature until the pool boils based[3] on the heat load for the emergency full core offload is 4.2 hours, and the calculated maximum boil-off rate is 90.8 gpm. These results show that there would be at least 4.2 hours available for corrective actions prior to SFP boiling in the unlikely event of a failure of all forced cooling to the SFP. The maximum boil-off rate of 90.8 gpm is less than the minimum makeup capacity of 100 gpm available from the condensate storage tanks, which is the normal makeup source. Additionally, up to 500 gpm of makeup water can be provided from fire protection systems within a 1-hour period. The licensee also stated that other means of SFP makeup are available to the operators, including up to 3,500 gpm from the RHR system, and up to 2000 gpm from the RHRSW system cross-tie to the RHR system from the ultimate heat sink.

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[1] In the responses, dated October 19, 2000, to the NRC staff's Request for Additional Information, the licensee stated that full-core offload is not a normally practiced refueling procedure at Fermi 2.

[2] The licensee stated that during a normal partial core offload, one RHR division and two trains of FPCCS will be aligned to cool the SFP. The worst-case single active cooling system failure, with respect to the SFP, would be the loss of the RHR division.

[3] As shown in Table 2 above, an emergency (unplanned) full-core offload scenario generates the highest decay heat rate in the SFP; thus, this scenario will result in the shortest time to boil.

Based on this review, BNL and the NRC staff conclude that in the unlikely event that there is a complete loss of cooling, the licensee is capable of aligning the makeup water from various sources to the pool before boiling begins and that makeup water will be supplied at a rate which exceeds the boil-off rate. BNL and the NRC staff also conclude that cooling the SFP at Fermi 2 by adding makeup water during an unlikely event that there is a complete loss of SFP cooling conforms with the guidance described in the SRP and is, therefore, acceptable.

3.7.2.3 Summary of Thermal-Hydraulic Considerations

Based on the NRC staff's review of the licensee's application, its supplements, and the BNL evaluation, the NRC staff found that the thermal-hydraulic aspects of the proposed license amendment request are acceptable. It is noted that a single-active failure of the normal cooling system was assumed only with a normal partial core discharge scenario. Thus, the partial core discharge is the normal core discharge activity, as stated by the licensee.

3.8 Summary of Evaluation

The licensee has proposed revisions to the TSs to maximize the spent fuel storage capacity of the SFP and replace spent fuel racks that use Boraflex with Boral. The proposed reracking will be accomplished in three campaigns. Two of the new spent fuel racks will contain dual purpose cells that can be used to store items of larger cross-sections and two tier racks will be designed to accommodate overhead platforms.

Based upon the NRC staff evaluation covering the areas of criticality, occupational radiation exposure, radioactive waste, structural integrity and adequacy, fuel-handling accidents, safe handling of heavy loads, and thermal hydraulics, the NRC staff concludes that the proposed revisions to TSs 4.3.c, 4.3.3, and 5.5.13 comply with all applicable regulatory documents (i.e., NUREG-0612 and -0800, NRC Bulletin 96-02, Regulatory Guide 1.13 and Office Technical Positions). These proposed revisions will allow the licensee to increase the spent fuel storage capacity.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Michigan State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.32, an environmental assessment and finding of no significant impact has been prepared an published in the *Federal Register* on January 25, 2001 (66 FR 7815). Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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