November 28, 2000

Mr. R. G. Lizotte
Master Process Owner-Assessment
c/o Mr. David A. Smith
Northeast Nuclear Energy Company
P. O. Box 128
Waterford, CT 06385-0128

SUBJECT: MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3 - ISSUANCE OF AMENDMENT RE: INCREASING SPENT FUEL STORAGE CAPACITY (TAC NO. MA5137)

Dear Mr. Lizotte:

The Commission has issued the enclosed Amendment No. 189 to Facility Operating License No. NPF-49 for the Millstone Nuclear Power Station, Unit No. 3 (MNPS3) in response to your application dated March 19, 1999, as supplemented by letters dated April 17, May 5, June 16, July 26, and November 21, 2000.

The amendment changes Technical Specification (TS) 1.40, "Spent Fuel Pool Storage Pattern"; 1.41, "3-OUT-OF-4 AND 4-OUT-OF-4"; 3/4.9.1.2, "Boron Concentration"; 3/4.9.7, "Crane Travel-Spent Fuel Storage Areas"; 3/4.9.13, "Spent Fuel Pool-Reactivity"; 3.9.14, "Spent Fuel Pool-Storage Pattern"; 5.6.1.1, "Design Features - Criticality"; and 5.6.3, "Design Features - Capacity." In addition, the amendment revises INDEX pages xii and xv for new figures and page numbers and replaces Figures 3.9-1 and 3.9-2 with four new figures and makes changes to the TS Bases consistent with changes to their respective TS sections. These changes are being made to support the increase in the capacity of the spent fuel pool at MNPS3 from 756 assemblies to 1,860 assemblies (an increase of 1,104 assemblies).

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/RA/

Victor Nerses, Sr. Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor

Docket No. 50-423

Enclosures: 1. Amendment No. 189 NPF-49
2. Safety Evaluation

cc w/encls: See next page

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Victor Nerses, Sr. Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
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Enclosures: 1. Amendment No. 189 NPF-49
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Accession Number:ML003744387
* See previous concurrence
**SE input provided on 7/5/00, no major changes
***SE input provided on 5/3/00, no major changes

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DATE	10/11/00	11/16/00	9/15/00	9/18/00	9/20/00	7/5/00	5/30/00

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NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

DOCKET NO. 50-423

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 189
License No. NPF-49

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

   A. The application for amendment by Northeast Nuclear Energy Company, et al. (the licensee) dated March 19, 1999, as supplemented by letters dated April 17, May 5, and June 16, and July 26, 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-49 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 189 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

James W. Clifford, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: November 28, 2000

cc:

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Unit 3

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ATTACHMENT TO LICENSE AMENDMENT NO. 189

FACILITY OPERATING LICENSE NO. NPF-49

DOCKET NO. 50-423

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
xii	xii
xv	xv
1-7	1-7
3/4 9-1a	3/4 9-1a
3/4 9-7	3/4 9-7
3/4 9-16	3/4 9-16
3/4 9-17	3/4 9-17
3/4 9-18	3/4 9-18
3/4 9-19	3/4 9-19
-----------	3/4 9-20
-----------	3/4 9-21
B 3/4 9-1	B 3/4 9-1
B 3/4 9-8	B 3/4 9-8
B 3/4 9-9	B 3/4 9-9
5-6	5-6
-----------	5-6a

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 189

TO FACILITY OPERATING LICENSE NO. NPF-49

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

DOCKET NO. 50-423

1.0 INTRODUCTION

1.1 License Amendment Request

By letter dated March 19, 1999, as supplemented by letters dated April 17, May 5, June 16, July 26, and November 21, 2000, the licensee (Northeast Nuclear Energy Company) of Millstone Nuclear Power Station, Unit No. 3 (MNPS3) requested an amendment to Facility Operating License No. NPF-49. The proposed amendment would change Technical Specification (TS) 1.40, "Spent Fuel Pool Storage Pattern"; 1.41, "3-OUT-OF-4 AND 4-OUT-OF-4"; 3/4.9.1.2, "Boron Concentration"; 3/4.9.7, "Crane Travel-Spent Fuel Storage Areas"; 3/4.9.13, "Spent Fuel Pool-Reactivity"; 3.9.14, "Spent Fuel Pool - Storage Pattern"; 5.6.1.1, "Design Features - Criticality"; and 5.6.3, "Design Features - Capacity." In addition, the proposed amendment would revise INDEX pages xii and xv for new figures and page numbers and would replace Figures 3.9-1 and 3.9-2 with 4 new figures and make changes to the TS Bases consistent with changes to their respective TS sections. These changes are being made to support the proposed increase in the capacity of the spent fuel pool (SFP) at MNPS3 from 756 spent fuel assemblies (SFAs) to 1,860 SFAs (an increase of 1,104 SFAs). The letters dated April 17, May 5, June 16, July 26, and November 21, 2000, provided clarifying information and did not change the staff's initial proposed no significant hazards consideration determination or expand the scope of the application as published in the Federal Register.

1.2 Oral Argument Before the Atomic Safety and Licensing Board (ASLB)

In response to a notice of opportunity for hearing (64 FR 48672, September 7, 1999), two organizations -- Connecticut Coalition Against Millstone (CCAM) and Long Island Coalition Against Millstone (CAM), hereinafter designated as CCAM/CAM -- jointly filed a Request for Hearing and Petition to Intervene dated October 6, 1999, in the license amendment proceeding.

In a Supplemental Petition dated November 17, 1999, CCAM/CAM jointly submitted 11 proposed contentions. On February 9, 2000, the ASLB issued its Prehearing Conference Order (Granting Request for Hearing) and ruled that CCAM/CAM had standing and that three contentions were admissible for litigation. All three contentions concern criticality and as submitted the contentions concern: 1) “Undue and Unnecessary Risk to Worker and Public Health and Safety”, 2) “Significant Increase in Probability Of Criticality Accident” and 3) “Proposed Criticality Control Measures Would Violate NRC Regulations.”

The Commission’s regulations in 10 CFR Part 2, Subpart K provide for hybrid hearing procedures that include a form of summary disposition based on oral argument in proceedings concerning the expansion of spent fuel storage capacity. Pursuant to these regulations, a presiding officer must grant a timely request to invoke these procedures. On February 22, 2000, the licensee made such a request, and on April 19, 2000, the ASLB granted it. On July 19, 2000, the ASLB heard oral arguments related to the three admitted contentions. In its Memorandum and Order dated October 26, 2000, the ASLB concluded that with respect to two of the three contentions admitted, there were no significant factual disputes that would warrant a further evidentiary hearing. On the third contention, the ASLB concluded that the "license condition" agreed to by all parties with respect to boron surveillance and concentration should be imposed on the amended license. Having concluded this, the ASLB ordered the hearing terminated. To comply with the ASLB order, the TS was revised to include a requirement that fulfills the agreed upon boron surveillance and concentration. On November 13, 2000, CCAM/CCM petitioned the Commission for review of the Memorandum and Order for “erroneous rulings...regarding administrative controls and criticality prevention issues”.

1.3 Rack Installation

As noted previously, by letter dated March 19, 1999, the licensee requested an amendment for the proposed increase in the capacity of the SFP. To support the increase, the licensee indicated the need to install additional spent fuel storage racks in the SFP and submitted a report, as Attachment 5 (non-proprietary version, Attachment 6, proprietary version) to the March 19, 1999, letter, on the installation of these additional racks for staff review. Some of the topics the report covered are as follows: fabrication; materials and heavy load considerations; criticality safety evaluation; thermal-hydraulic considerations; structural/seismic considerations; fuel handling and construction accidents; fuel pool structure integrity considerations; Boral surveillance program; installation; and radiological evaluation.

Subsequent to the March 19, 1999, submittal, the licensee notified the Nuclear Regulatory Commission (NRC) by letter dated July 26, 2000, that it was necessary for the licensee to proceed with rack installation in order to meet the scheduled work plan for the upcoming MNPS3 refueling outage. Also, the licensee noted that NRC approval of the proposed TSs governing use of the additional racks was not required to support installation activities and that licensee safety evaluations had been prepared in accordance with 10 CFR 50.59 to support installation of the additional racks. The licensee stated that these evaluations concluded that the rack installation could be performed safely and in accordance with the existing NRC-approved licensing and design basis for MNPS3. The licensee also stated that use of the new racks for storage of nuclear fuel is prohibited until the NRC approves the proposed amendment and the proposed changes to the TS are received and implemented.

Although the licensee determined that installation of the additional spent fuel storage racks did not require prior NRC review and approval, the staff’s review of this element of the March 19, 1999, submittal had already been completed before the receipt of the July 26, 2000, letter and is included in this evaluation. However, this evaluation does not include a review of the licensee’s 10 CFR 50.59 evaluation.

2.0 EVALUATION

2.1 Occupational Radiation Exposure

With respect to occupational radiation exposure, the staff reviewed the licensee's application to increase the SFP capacity through the installation of 15 additional spent fuel storage racks at MNPS3. A number of facilities have performed similar installations in the past. On the basis of the lessons learned from these installations, the licensee estimates that the proposed fuel rack installation can be performed for between 2 and 5 person-rem.

All of the operations involved in the fuel rack installation will utilize detailed procedures prepared with full consideration of ALARA (as low as is reasonably achievable) requirements pursuant to 10 CFR 20.1101(b). Workers performing the SFP rack installation will be given pre-job briefings to ensure that they are aware of their job responsibilities and 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. The licensee will issue thermoluminescent dosimeters (TLDs) and self-reading dosimeters to all personnel. Additional personnel monitoring equipment (such as extremity TLDs or multiple TLDs) will be issued as required.

The licensee will use divers for the installation of the spent fuel storage racks in the SFP. The licensee will install 14 of the 15 racks in the SFP during the current rack installation. Using conservative assumptions for the spent fuel in the SFP (i.e. all the SFAs stored in the SFP have a burnup of 60,000 MWD/MTU, all SFAs have decayed only 100 hours, and all SFAs are located at the fuel pool wall), the licensee calculates that the maximum dose rate at the outer surface of the SFP wall due to the storage of the additional SFAs in the SFP will be 2.5 mR/hr. Although the current dose rates at the wall surfaces are negligible, the maximum estimated dose rate of 2.5 mR/hr is well within the design basis for the original SFP design. The addition of the new racks in the SFP will not require radiation zoning changes in any accessible areas surrounding the SFP.

The licensee will conduct radiation surveys of the diving area using two independent survey instruments prior to the scheduled rack installation diving period and will verify these SFP surveys every 24 hours during the scheduled diving period. The licensee has no plans to move fuel or any other high radiation sources in the SFP during the scheduled rack installation diving period. The licensee will terminate the dive if there is any change in the underwater radiation environment. Supplementary lighting will be used to illuminate the work area in the SFP if the divers and health physics (HP) technicians determine that additional lighting is necessary to perform the rack installation. The licensee will utilize underwater television cameras to maintain visual contact with the divers during all diving operations. 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 as many times during the diving operation as it becomes necessary to maintain diver doses ALARA.

In order to minimize the underwater radiation fields where the divers will be working, the licensee will relocate the SFAs in the SFP to move the highest burn-up fuel away from diver work areas. 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. In addition, the divers will be in continuous voice communication with radiation protection personnel. The licensee will use diver tethers with tenders to ensure that divers maintain a safe distance from SFAs or other high radiation sources stored in the SFP.

After each installation of a rack in the SFP, the licensee will wash the lifting device with demineralized water to control the spread of contamination. The licensee does not expect the concentrations of airborne radioactivity in the vicinity of the SFP to increase as a result of the expanded spent fuel storage capacity. However, the licensee will operate continuous air monitors in areas where there is a potential for significant airborne activity during installation of the racks in the SFP.

2.1.1 Conclusion on Occupational Radiation Exposure

On the basis of our review of the licensee’s application, the staff concludes that the proposed increase in spent fuel storage capacity at MNPS3 can be performed in a manner that ensures doses to the workers will be maintained consistent with sound radiation protection principles that ALARA requires pursuant to 10 CFR 20.1101(b) and, therefore, is acceptable.

2.2 Solid Radioactive Waste

Spent resins are generated by the processing of SFP water through the SFP Purification System. These spent resins are changed out about once a year and the licensee does not expect this change-out frequency to be significantly affected by the storage of additional spent fuel in the SFP. In order to maintain the SFP water as clean as possible, and thereby minimize the generation of spent resins, the licensee will vacuum the floor of the SFP to remove any radioactive crud, sediment, and other debris before the new racks are installed. Filters from use of this underwater vacuum system will be a source of solid radwaste. Overall, however, the licensee does not expect that the storage of additional spent fuel in the SFP will result in a significant change in the generation of solid radwaste at MNPS3.

2.2.1 Conclusion on Solid Radioactive Waste

On the basis of our review of the licensee’s application, the staff concludes that the proposed increase in spent fuel storage capacity at MNPS3 can be performed in a manner that ensures doses to the workers will be maintained in full consideration of ALARA requirements (10 CFR 20.1101(b)) and the generation of additional solid radioactive wastes will be minimized. As regards such radioactive waste, the application is, therefore, acceptable.

2.3 Accident Dose Considerations

The staff previously evaluated the MNPS3 design-basis fuel-handling accident in the fuel pool in NUREG-1031, “Safety Evaluation Report Related to the Operation of Millstone Nuclear Power Station, Unit No. 3" (July 1984) for the potential radiological consequences. In that evaluation, the staff assumed that one SFA was dropped in the SFP during refueling operation and that all of the fuel rods in the dropped SFA were damaged, thereby releasing the volatile fission products from the fuel rod gaps. The staff found in that evaluation that the potential radiological consequences of a fuel-handling accident were well within the 10 CFR Part 100 dose guidelines.

Fuel rod damage following an SFA drop is primarily dependent upon the weight and design of the fuel assembly, the drop height, which determines the kinetic energy upon impact, and the orientation of the falling fuel assembly. None of these parameters are changed by this proposed increase in SFP storage capacity. The total number fuel rods damaged remains the same as the number assumed in the previously evaluated fuel handling accident . The new spent fuel pool rack configuration does not change the elevation of the top of stored fuel and the fuel drop height remains the same. Therefore, the finding in NUREG-1031 is still bounding in that the radiological consequences are within the 10 CFR Part 100 dose guidelines.

2.3.1 Conclusion on Accident Dose Considerations

The staff finds that the radiological consequences from the proposed increase in SFP storage capacity continue to meet the 10 CFR Part 100 dose guidelines and, therefore, are acceptable.

2.4 Structural Aspects

2.4.1 Storage Racks

At present, the SFP has 21 storage racks containing a total of 756 storage cells (square holes in the racks into which SFAs are placed for storage) that were installed during the original construction. The proposed TS change involves the installation of 15 additional new high density storage racks (14 racks during the currently planned additional rack installation, with the 15^{th} planned for a yet to be established date) in the existing SFP floor space. These 15 new racks will have a maximum capacity of 1104 storage cells. Existing racks will remain in the pool, but the additional racks will have a closer storage cell to storage cell (and hence a closer SFA to SFA) spacing to maximize the fuel storage capacity.

SFP racks are seismic Category I equipment and are required (in order to satisfy GDC 62) to remain functional during and after a safe shutdown earthquake (SSE) under all applicable loading conditions. The licensee's contractor, Holtec International, performed the design, fabrication, and safety analysis of the new high density SFP storage racks. All storage racks are made of American Society of Mechanical Engineers (ASME) SA240-Type 304L austenitic stainless steel. The only non-stainless steel material utilized in the rack is the neutron absorber material, which is a boron carbide-aluminum alloy called Boral.

The overall design of the new racks at MNPS3 is similar to Holtec racks that the Nuclear Regulatory Commission (NRC) has approved for service at many other nuclear power plants. The key design criteria are based on the NRC letter dated April 14, 1978, to all power reactor licensees transmitting an NRC position entitled "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", dated April 14, 1978, (Accession No. 8105150005), as modified by NRC letter dated January 18, 1979, (Accession No. 800513018).

The key design criteria of the MNPS3 SFP racks are described in Section 2.2 of Attachment 5 of the licensee's letter dated March 19, 1999. Briefly, the following criteria are applicable from the structural safety point of view: (1) all new racks are required to be free-standing; (2) all free-standing racks are required to be kinematically stable (against tipping or overturning) when a seismic event that is 150% of the postulated SSE is imposed; (3) all primary stresses in the racks 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 at or below 150 °F in the wake of a normal refueling with single active failure of one train of spent fuel pool cooling; and (5) the ability of the reinforced concrete structure of the SFP to withstand the effects of the load combinations set forth in Standard Review Plan (SRP) Section 3.8.4 must be demonstrated.

At the time of the original rack installation in the MNPS3 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 an NRC-accepted computer program, DYNARACK, for the dynamic analysis to demonstrate the structural adequacy of the SFP 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 used for other rack 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 3-D dynamic behavior of the rack and the SFAs including the frictional and hydrodynamic effects. The DYNARACK computer code accurately simulates the friction, impact, and other nonlinear dynamic events. The code models the beam characteristics of the rack including shear, flexibility, and torsion effects appropriately, by modeling each rack as a 3-D structure having the support pedestals and the SFAs in proper locations. The potential interaction between the SFA and storage cell walls 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 corresponding to the generated acceleration time-history to envelope their target (design basis) counterparts with only finite enveloping infractions.

In response to a staff request for additional information dated February 25, 2000, the licensee explained in their letter dated May 5, 2000, that the target response spectra were obtained by broadening and smoothing the plant response spectra for the fuel pool floor (Elevation 11' 0" in the Fuel Building) in accordance with Regulatory Guide 1.122 and Table 1.8-1 of the MNPS3 Final Safety Analysis Report (FSAR). In Attachment 6, Section 6.4 of the licensee's submittal dated March 19, 1999, the licensee generated the time-histories to satisfy the preferred criterion stated above. This procedure is acceptable to the staff.

The licensee considered the applicable loads and their combinations in the seismic analysis of the racks, and performed parametric simulations for both the SR and WPMR analyses. The parameters, which were varied in the different 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 operating basis earthquake (OBE)). For the parametric simulations, the licensee performed a total of 26 3-D SR model analyses and 7 WPMR model analyses. The results of these analyses show the maximum rack displacement to be 1.02 inches (for 1.5xSSE condition). For this case, a rack overturning evaluation indicated the factor of safety against overturning to be 58, 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, and that the structural integrity and stability of the racks and SFAs will be maintained.

From the large number of computer runs of parametric evaluations, 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 the 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 during any of the seismic events. However, there are some impacts between adjacent racks at the baseplate level, and some impacts between SFAs and storage cell walls. The licensee evaluated the stresses imposed by the instantaneous impacts on the steel baseplate, and indicated that all stresses were well below the corresponding ASME Code Section III, Subsection NF limits. In addition, the licensee calculated the weld stresses of the rack at the connections (e.g., baseplate-to-storage cell, and baseplate-to-pedestal, connections) under the SSE and OBE loading conditions, and demonstrated that all the calculated weld stresses are well below the corresponding allowable stresses specified in the 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) there are large factors of safety for the induced stresses of the rack when compared to the corresponding allowable values provided in the ASME Code, Section III; and (2) there are no rack-to-wall and rack-to-rack impacts (except some insignificant impact at the rack baseplate level as described above). Therefore, the staff concludes that the racks will perform their safety function and maintain their structural integrity under postulated loading conditions and are, therefore, acceptable.

2.4.2 Spent Fuel Pool (SFP) Structural Integrity

The SFP is a safety-related, seismic Category I, reinforced concrete (RC) structure. The SFP and the adjacent two pools, i.e., the Cask Pit (CP) and the Transfer Canal (TC), are located in the Fuel Building, and are separated by reinforced concrete walls of various thicknesses. The walls are supported at different elevations by a massive on-grade RC slab which extends down to the soil elevation -3' 3". The SFP on-grade mat (RC slab) is 14' 6" thick and its upper elevation is at 11' 3". The walls surrounding the SFP are 6' 0" thick on two sides and 6' 6" on the other two sides.

The licensee analyzed pool regions using the NRC-accepted industry finite element computer program, STARDYNE; and, in conformance with SRP 3.8.4, the results for individual load components were combined using factored load combinations. In addition to the dead and live loads, the analysis considered the seismic, thermal, and hydrodynamic loadings. Tables 8.5.1 and 8.5.2 in Attachment 5 of the licensee's letter dated March 19, 1999, show the analytical results from potentially limiting load combinations for the bending strength evaluation and shear strength evaluation of the slab and walls, respectively. As seen from these tables, the predicted limiting safety margins for the reinforced concrete range from 13.75 to 22.16 for bending strength of the walls, and 1.97 to 4.73 for shear strength of the same structural elements. The licensee also indicated that the maximum design basis bulk pool temperature will be increased from 140 °F to 150 °F. The 150 °F temperature is within the American Concrete Institute Code 349 allowable value and is, therefore, acceptable.

The 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.

2.4.3 <u>Fuel Handling Accident</u>

The licensee evaluated the following fuel handling accident cases: (1) the drop of an SFA (with its handling tool) impacting the top of a rack ("shallow drop" scenarios); and (2) the drop of an SFA (with its handling tool) falling through an empty storage cell and impacting the rack baseplate ("deep drop" scenarios).

Section 2.8.4 "Heavy Load Drop Accident Analysis" provides the results of the staff's review and evaluation on the "shallow drop" and "deep drop" scenarios. The results were that the staff agreed with the licensee's conclusion that, based on the load drop analyses, the integrity of the fuel and the SFP would be maintained if a fuel assembly or a spent fuel storage rack were dropped.

2.4.4 <u>Conclusion on the Structural Aspects</u>

Based on the review and evaluation of the licensee's submittal of March 19, 1999, and its subsequent response, dated May 5, 2000, to the staff's request for additional information (RAI), the staff concludes that the structural analyses of the spent fuel storage racks and the SFP structure under seismic and accident loading conditions are in compliance with the acceptance criteria specified in the FSAR and are consistent with the current licensing basis and, therefore, are acceptable.

2.5 <u>Material Compatibility</u>

2.5.1 <u>Structural Materials</u>

The structural materials used in the fabrication of the new spent fuel racks include: ASME SA240-Type304L austenitic stainless steel for all sheet metal stock and internally threaded support legs, ASME SA564-Type 630 precipitation hardened stainless steel (heat treated to 1100 °F) for externally threaded support spindle, and ASME Type SFA 5.9 R308L stainless steel for weld material.

These materials used in the Holtec racks are consistent with industry practice and have a proven history of compatibility in the spent fuel pool environment. Therefore, the staff considers that they are acceptable for use in this application.

2.5.2 Poison Material

The Holtec racks employ Boral as the neutron absorber material. Boral is a hot-rolled cermet of aluminum and boron carbide, clad in 1100 alloy aluminum. It is chemically inert and has a long history of applications in the spent fuel pool environments where it has maintained its neutron attenuation capability under thermal loads. A strongly adhering film of impervious hydrated aluminum oxide passivates the surface of the aluminum typically within a few days of being placed in water. The corrosion layer only penetrates the surface of the aluminum cladding a few microns during passivation and causes no net loss of aluminum cladding. Hydrogen, a product of the corrosion process, may cause swelling in the rack panels resulting in deformation of the storage cells. To prevent this from occurring, the racks are designed with spot welding to vent the corrosion gases. The neutron absorbing capability of Boral is not affected by this corrosion process.

Boral has a long history of application in nuclear plants and its use has indicated that the material exhibits a high degree of stability to radiation. Based on accelerated test programs, Boral is considered a satisfactory material for reactivity control in spent fuel storage racks and is expected to fulfill its design function over the lifetime (at least until the operating license expiration date in the year 2025) of the racks. However, the licensee has deemed it prudent to establish a Boral surveillance program. The surveillance program includes the use of blackness testing and Boral coupons to obtain physical and chemical properties from which the stability and integrity of the Boral in the storage cells may be inferred.

Based on these desirable material characteristics and its history of effective performance, the staff finds the use of Boral in this application to be acceptable.

2.5.3 Conclusions on Material Compatibility

Based on its evaluation, the staff finds the materials utilized in the fabrication of the storage racks manufactured by Holtec International are compatible with the SFP environment at MNPS3. The type of degradation exhibited by the racks does not affect their neutron absorbing capability and the rack design allows for the venting of the corrosion product gas, hydrogen, to prevent deformation of the rack's storage cells. In addition, the licensee has established a Boral surveillance program to verify the continued, satisfactory performance of Boral throughout the life of the storage racks. Based on this evaluation, the staff concludes that the materials used in the new storage racks are acceptable.

2.6 Criticality Considerations

2.6.1 Criticality Calculations

After installation of the additional storage racks, the pool will contain three distinct administratively controlled storage regions as shown in Figure 1 of Attachment 4 of the licensee's submittal dated March 19, 1999. Each region will be characterized by a nominal center-to-center spacing of the storage cells. By design, the new storage cells will contain a fixed neutron absorber for primary reactivity control. The new racks will be grouped in Region 1 and 2. The 21 existing racks, which will remain in place, will be designated as Region 3.

The Region 1 and 2 racks will contain Boral as the neutron absorbing material. The Boral absorbers are designed to fully shadow the total active length of the fuel assemblies. All Region 3 racks have Boraflex as the neutron absorbing material but no credit is taken for the Boraflex in the criticality analysis. Furthermore, the racks are assumed to be flooded with un-borated water at a temperature within the SFP operating range that results in the highest reactivity.

Region 1 is designed to accommodate the following storage rack configurations: (1) new unirradiated fuel assemblies with maximum nominal enrichment of 5.0 weight percent (w/o) U-235 in a 3-out-of-4 configuration with the fourth cell empty and blocked, and without restriction on burnup; or (2) a 4-out-of-4 configuration with burnup/enrichment restrictions as depicted in Figure 4.1.1 in Attachment 5 of the licensee’s submittal dated March 19, 1999. The licensee’s criticality analysis for Region 1 of the spent fuel pool showed that the maximum k_{eff} is less than 0.93, thus conforming to the criteria of SRP Section 9.1.2 of k_{eff} less than or equal to 0.95. Region 1 has the capacity to store up to 350 fuel assemblies, and is also designed to accommodate an emergency core offload.

The Region 2 racks were designed and analyzed to store 754 fuel assemblies, subject to certain burnup/enrichment limits, utilizing a 4-out-of-4 storage configuration. The burnup/enrichment restrictions imposed in Region 2 are more restrictive than those imposed in Region 1, with the acceptable domain (maximum k_{eff} of 0.945) depicted in Figure 4.1.2 of Attachment 5 of the licensee’s submittal dated March 19, 1999. The calculated maximum reactivity in Region 2 includes the reactivity effect of the axial distribution in burnup, providing an additional margin of uncertainty for the depletion calculations.

The licensee performed calculations to qualify the 21 existing Westinghouse designed racks, referred to herein as Region 3 racks. The Region 3 racks are designed and analyzed to accommodate up to 756 fuel assemblies with a maximum nominal initial enrichment of 5.0 w/o U-235 that have accumulated minimum burnup and cooling times that fall within the acceptable domains depicted in Figure 4.1.3 of Attachment 5 of the licensee’s submittal dated March 19, 1999. The licensee’s criticality analysis for Region 3 of the spent fuel pool storage showed that the maximum k_{eff} is equal to 0.945, which conforms to the SRP criterion. The calculated maximum reactivity in Region 3 also includes the reactivity effect of the axial distribution in burnup, providing an additional margin of uncertainty for the depletion calculations.
The burnup criteria identified above for acceptable storage in the three regions will be implemented by appropriate licensee administrative procedures to ensure verified burnup.

The nuclear design and safety analysis was done by Holtec International. They used primarily the MCNP code, a 3-D Monte Carlo code developed by Los Alamos National Laboratory, using continuous energy cross-sections. Supplemental calculations, for verification, were done with NITAWL-KENO5a, a 3-D code developed by Oak Ridge National Laboratory, also using Monte Carlo. The 238 group SCALE cross-sections were used. The CASM0-3 code version 4.4, a transport theory code, was also used for some calculations of temperature and tolerance reactivity effects. 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 for primary analysis provides greater assurance for the accuracy of the analysis.

The methodologies and cross-sections have been benchmarked by Holtec against a number of relevant critical experiments simulating parameters related to storage racks. Benchmark calculations, presented in Appendix 4A of Attachment 5 of the licensee's submittal dated March 19, 1999, indicate a bias of 0.0009 with an uncertainty of 0.0011 for MCNP4a and a bias of 0.0030 with an uncertainty of 0.0012 for KENO5a, both evaluated with the 95 percent probability at the 95 percent confidence level. These benchmark calculations have been used to develop methodology bias and uncertainty factors to be added to the nominal k_{eff} calculations for the racks. Holtec has also determined the potential variation of rack and fuel parameters that are used in determining the k_{eff} of the rack and fuel system. These independent parameters include rack manufacturing tolerances, boron loading variations, Boral width tolerance variation, and storage cell lattice pitch variation. The variation of k_{eff} 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_{eff} calculation. This treatment of the uncertainties is in conformance with NRC recommendations as provided in the previously mentioned letters (see Section 2.4.1) dated April 14, 1978, and January 18, 1979. In addition, rack calculations were done using a conservative infinite array of fuel assemblies except for the assessment of peripheral effects and certain abnormal conditions.

Holtec also investigated abnormal conditions that might be associated with the SFP. These include: (1) pool water temperature and density effects; (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 reasonably possible abnormal conditions will not violate the k_{eff} TS limits.

Analysis conducted by Holtec indicated that a minimum concentration of 425 ppm of soluble boron in the pool water is sufficient to ensure k_{eff} will remain less than or equal to 0.95 assuming a mis-loaded fuel assembly. However, the proposed TS 3.9.1.2, as modified by the licensee's letter dated April 17, 2000, sets a minimum SFP soluble boron concentration of 800 ppm. This provides an added conservatism to ensure k_{eff} will remain less than or equal to 0.95. Further, the licensee's letter dated May 5, 2000, retains an existing TS applicability requirement, thereby requiring the proposed boron concentration of 800 ppm be maintained whenever fuel is stored in the SFP. Upon implementation of the proposed TS, the MNPS3 SFP boron surveillance frequency will be every 7 days. Also, in a letter dated June 16, 2000, the licensee modified their initial application of March 19, 1999, to retain the existing TS remedial action in the event the soluble boron concentration is reduced below the proposed acceptance limit. The staff finds the results of the analysis satisfactory because acceptable methods and assumptions were used. The staff finds that the proposed revisions to the TS surveillance requirement whenever fuel is stored in the SFP, and the licensee's remedial action when the soluble boron concentration is reduced below the acceptance limit of 800 ppm to be acceptable because these revisions will retain the surveillance requirement and remedial action that are in the current TS.

As part of the proposed increase in spent fuel storage capacity, Holtec also calculated the local peak water temperature and the local peak clad temperature. The local peak water temperature was found to be well below the boiling temperature at the top of the pool. The local peak clad temperature was found to be well below the temperature where clad damage or a zirconium-water reaction would occur. The results show that the temperatures are below any zirconium/fuel material failure criteria; therefore, the staff finds these results acceptable.

2.6.2 Conclusion on Criticality Considerations

The NRC staff has reviewed the information submitted by the licensee for the continued operation of MNPS3 in support of its proposed increased SFP capacity. Based on this review, the staff has concluded that the requested TS changes are acceptable, and satisfy the staff's positions and requirements (pursuant to 10 CFR 50.68 and GDC 62) for the prevention of criticality in fuel storage and handling.

2.7 Thermal-Hydraulic Considerations

2.7.1 Background

Thermal-hydraulic considerations were addressed comprehensively in a separate licensee application requesting full-core offload capability as a routine planned evolution. The licensee's application dated January 18, 1999, as supplemented by letters dated April 5 and December 21, 1999, and May 2, 2000, requested an amendment that proposed changes to the TS and FSAR to allow an entire reactor core to be offloaded to the SFP and an increase in the SFP water temperature limit from 140 °F to 150 °F during routine (planned) refueling outages.

In order to render the SFP thermal-hydraulic analyses independent of a particular operating/refueling cycle, the analyses performed by the licensee's contractor (Holtec International), and submitted with the January 18, 1999, application, assumed that the SFP fuel inventory corresponds to the end of the licensed life of MNPS3. A maximum decay heat load based on a total of 3048 SFAs was assumed versus the decay heat load based on the current assumed 2169 SFAs. As a result, the heat load in the MNPS3 SFP is maximized by assuming this maximum quantity of fuel stored in the SFP. This approach also bounds the case for the potential maximum number (1860) of SFAs that could result from the proposed increase in spent fuel storage capacity at MNPS3 as identified in this license amendment application. As noted previously, the licensee, in its application, proposed to install high density storage racks to increase the SFP storage capacity from 756 to 1860 SFAs.

2.7.2 Conclusions on Thermal-Hydraulic Considerations

Based on the staff's review of the justification, analyses, and other information provided by the licensee in the submittals noted in Section 2.7.1, the staff concluded, in Amendment No. 182 dated September 12, 2000, that the licensee's proposal to offload an entire reactor core to the SFP as a routine, planned evolution is acceptable. Furthermore, the staff concluded that the approved increase to the SFP cooling limit will still allow adequate cooling of the SFP even with the increased SFP capacity as proposed in this amendment request. The safety evaluation of the staff review on thermal-hydraulic considerations was forwarded to the licensee by a letter dated September 12, 2000, (Accession No. ML003744109).

2.8 Control and Handling of Heavy Loads

2.8.1 Background

NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants,” dated July 1980, provides regulatory guidelines for licensees to assure safe handling of heavy loads in areas where a load drop could impact on stored spent fuel, fuel in the reactor core, or equipment that may be required to achieve safe shutdown or permit continued decay heat removal. The objectives of the guidelines are to assure that either: (1) the potential for a load drop is extremely small, or (2) the potential hazards of load drops do not exceed acceptable limits. The NUREG provides guidelines that are implemented in two phases. Phase I guidelines address measures for reducing the likelihood of dropping heavy loads by: (1) providing criteria for establishing safe load paths; (2) procedures for load handling operations; (3) training of crane operators; (4) design, testing, inspection, and maintenance of cranes and lifting devices; and (5) analyses of the impact of heavy load drops.

Phase II guidelines address alternatives for mitigating the consequences of heavy load drops, including using either: (1) a single-failure-proof crane for increased reliability of the load handling system; or (2) electrical interlocks and mechanical stops for restricting crane travel; or (3) load drop and consequence analyses for assessing the impact of dropped loads on plant safety and operations.

Generic Letter (GL) 85-11, “Completion of Phase II of Control of Heavy Loads at Nuclear Power Plants, NUREG-0612,” dated June 28, 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 enhance safety.

In NUREG-1031, “Safety Evaluation Report (SER),” dated July 1984, and subsequent Supplements 1 and 2, dated March and September 1985, respectively, the staff approved the licensee’s measures for controlling and handling heavy loads, including moving concrete floor plugs, new fuel casks, and weights needed for testing the cranes. In addition, the licensee’s response to NRC Bulletin 96-02, “Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment,” dated May 10, 1996, reaffirmed its commitment to apply the guidelines in NUREG-0612.

The proposed amendment addresses heavy load issues, including handling of spent fuel assemblies, spent fuel storage racks, and the gate that isolates the pool from the transfer canal. Licensee considerations are given to the design and operation of the hoisting system, safe load paths, procedures, training of the crane operator and rack installation crew, and analyses of postulated heavy load drop accidents and consequences over fuel in the racks, over the spent fuel pool, and over safety-related equipment. The licensee states that MNPS3 is not currently licensed to transport fuel casks into the spent fuel building. Therefore, analyses of cask drops and consequences are not addressed in this amendment request.

2.8.2 Hoisting Systems

The 10-ton new fuel receiving crane will be used to lift the racks and the fuel assembly shipping containers (containing new fuel) from the receiving bay in the fuel building to the storage location on the operating floor. There, the 10-ton new fuel handling crane will be used to retrieve the racks and the fuel assemblies from the shipping containers. The 10-ton new fuel handling crane will be used for handling all heavy loads during the rack installation, including installing the new racks in the spent fuel pool, lifting the gate from the fuel transfer canal, and moving the fuel assemblies. A 10-ton hoist will be suspended from the bridge of the new fuel handling crane and used in conjunction with the storage rack lifting rig (special lifting device) to lift and move the racks into the spent fuel pool. The use of the 10-ton hoist will allow the licensee to avoid contamination of the 10-ton new fuel handling crane hook during SFP rack lifting operations in the SFP. Therefore, the 10-ton new fuel handling crane will not be used to manipulate the racks while they are in the SFP proper.

In NUREG-1031, the staff approved the fuel receiving and the new fuel handling cranes and the hoisting system and found that the general load handling system and policy and procedures at MNPS3 are consistent with the guidelines in NUREG-0612. However, the storage rack lifting rig is specifically designed to lift the new racks; thus, it is not addressed in NUREG-1031.

As stated in NUREG-1031, the fuel handling crane is designed, fabricated, installed, inspected, tested, and operated in accordance with requirements of the Crane Manufacturers Association of America (CMAA) Specification No. 70, “Specifications for Electric Overhead Traveling Cranes,” and American National Standards Institute (ANSI)/ASME B30.2-1976, “Safety Standards for Overhead and Gantry Cranes (Top Running Bridge and Multiple Girder).”

The licensee states that the new racks will be lifted using a remotely engaged spent fuel rack lifting rig that is specially designed to lift the storage racks. The lifting rig is designed and tested in accordance with the guidelines in NUREG-0612, Sections 5.1.6(1), 5.1.6(3a), and the requirements in ANSI N14.6 (1978), “Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More for Nuclear Materials.” Accordingly, the lifting rig has twice the design safety factor with respect to the yield and ultimate strength, six times and ten times the combined concurrent static and dynamic loads for the yield and ultimate strength, respectively, of its material of construction. The lifting rig also is redundantly designed with four independently loaded lift rods such that failure of a single rod will not result in uncontrolled lowering of the rack. Therefore, the lift rods and lift points of the lifting rig are designed and tested as follows: (1) with a stress design factor (as specified in ANSI N14.6 (1978), Section 3.2.1) of five times the lifted weight without exceeding the ultimate strength of the material; and (2) load tested to 300% of the maximum weight to be lifted, and hoisted and suspended for a minimum of 1 hour. After load testing, examination of the critical weld joints using a liquid penetrant is performed.

Based on the capacity of the 10-ton fuel handling crane coupled with the capacity of the 10-ton hoist and the storage rack lifting rig, the staff concluded that the cranes are adequate to support the weight of the racks and the added rigging loads without risk of a load drop. In addition, the design, inspection and testing of the crane and lifting device will help to assure the reliability of the system for safe handling of the racks and further reduce any potential risk of an accidental rack drop during rack installation.

2.8.3 Load Path

Existing spent fuel storage racks will remain in the spent fuel pool and will not be moved or modified. As stated in the licensee’s June 16, 2000, response to the RAIs, the proposed new racks will be lifted to the fuel building operating level through the equipment hatch using two cranes in tandem: the 10-ton new fuel receiving and the 10-ton new fuel handling cranes. At the operating level, the racks will be moved along a safe load path to the spent fuel pool using the 10-ton new fuel handling crane. The racks will then be lowered into the pool to a minimum height of 6 inches above the pool liner for movement into their permanent location. The 10-ton hoist will be attached to the 10-ton new fuel handling crane in order to lower the new racks into the pool and position them in their location. The licensee states that the racks and lifting rig will not be carried over any fuel or near irradiated fuel in the SFP. The racks and lifting rig also will be procedurally precluded from travel over any safe-shutdown equipment along the safe load path. Fuel will be moved into racks that are not in the safe load path. Movements of the racks along the SFP floor will not exceed 6 inches unless other travel heights are needed to clear pool floor projections. The licensee stated that the transfer canal pool gate will be moved over the fuel racks; however, in accordance with TS 3/4.9.7, “Crane Travel - Spent Fuel Storage Areas,” the gate is restricted from movement over fuel assemblies in the SFP.

The new installed fuel storage racks will not significantly change the method of handling loads during normal plant operations because the same equipment (i.e., the fuel handling system), methods, and procedures used before the new additional rack installation will be used following the installation.

Based on the above discussion, the staff finds that safe load paths for movement of the storage racks do not involve any movement over spent fuel. Additional assurance that this will be the case is provided by the licensee’s actions to implement administrative controls to prevent movement of the new racks over any region of the pool containing fuel or safe shutdown equipment. It is further supported by the licensee’s pattern for storing and moving spent fuel into racks that are not in the safe load paths and by administrative controls implemented by the licensee.

2.8.4 Heavy Load Drop Accidents Analysis

The MNPS3 FSAR does not provide a heavy load drop analysis. However, FSAR Section 9.1.5, “Overhead Heavy Load Handling Systems,” does state that, if a heavy load lift has not been previously evaluated, a 10 CFR 50.59 evaluation will be performed prior to the lift operation to ensure that an unreviewed safety question does not exist. Accordingly, for this license amendment, the licensee analyzed postulated load drop accidents of the spent fuel assemblies, spent fuel storage racks, and the gate that isolates the pool from the transfer canal. Two drops of the fuel assemblies using a bounding impact weight of 2100 lbs. (includes the weight of the heaviest fuel assembly plus the Rod Control Cluster Assembly and the fuel handling tool) from a maximum lift height of 36 inches above the racks were considered: a vertical drop on top of the racks (“shallow-drop”), and a vertical drop to the base plate of the racks (“deep drop”). The fuel assembly drop on top of the storage racks resulted in deformation of the racks to a depth of 6.64 inches below the top of the rack and no damage to the fuel in the racks. The fuel assembly drop onto the base plate of the racks resulted in some deformation of the baseplate and no damage to the spent fuel pool liner.

A vertical drop of the heaviest rack, which weighs 18,050 lbs. (and does not include the lift rig, rigging, and the 10-ton electric hoist) from 40 feet above the SFP floor was evaluated. The results indicated that the SFP liner would be punctured and SFP water would leak. The SFP concrete floor slab would be indented approximately 2.7 inches; however, no structural damage to the concrete would occur. In a telephone conference on May 25, 2000, the NRC staff requested additional information from the licensee pertaining to the handling and control of heavy loads during the rack installation operation. By letter dated June 16, 2000, in response to the staff's RAI, the licensee stated that a rack drop would puncture a hole approximately 5 inches in diameter in the SFP liner. However, the licensee also stated that any leakage through the SFP would be collected by the leak chase system. As stated in the licensee's June 16, 2000, letter, valves in the leak chase system are normally closed; therefore, the normally closed valves will isolate any SFP leakage and minimize any loss of coolant inventory from the SFP. In addition, any significant loss of SFP coolant inventory would trip the low level alarms in the control room to alert plant operators to the conditions and prompt them into the appropriate emergency procedure.

Furthermore, SFP makeup can be obtained from a number of sources to supplement any leakage from the SFP. As noted in FSAR Section 9.1.3.2, emergency makeup can be provided from the service water system. Normal SFP makeup can be provided from the demineralized water system, the refueling water storage tank, the reactor coolant drain tank, and the reactor makeup water storage tank. The licensee also has a contingency procedure whereby, if the spent fuel pool liner is punctured as a result of a rack drop, liner repair could be performed during the rack installation.

Also, in their letter dated May 5, 2000, the licensee stated that the concrete slab experiences a maximum localized (peak normal) compressive stress of 25.2 ksi, which exceeds the failure stress of 20.2 ksi and indicates some localized crushing of the concrete slab. However, the licensee's analysis also indicates that the high stress region is located directly beneath the pedestal and is limited to a circular area less than 5 inches in diameter. The 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 pool. The staff considers this analysis acceptable.

Therefore, because the structural integrity of the concrete slab in the SFP remains unimpaired after a rack drop, and because the licensee has the capability to isolate an SFP leak and provide makeup to supplement any SFP inventory loss, the staff agrees with the licensee's conclusion that neither catastrophic damage of the SFP structure nor rapid loss of pool water would occur.

The licensee did not analyze the potential for a rack drop on spent fuel assemblies or on safety related equipment because: (1) the racks would not be moved over any fuel in the pool or near any irradiated fuel; and (2) rack upending will be staged away from the SFP, safety-related equipment, and in an area that is capable of withstanding a rack drop from at least 6 feet.

The licensee analyzed a drop of the gate (5,000 lbs.) that isolates the pool from the transfer canal. The gate will also be lifted using the 10-ton fuel building auxiliary crane and dual rigging that satisfies NUREG-0612 safety margins. Accordingly, the gate rigging will be load-tested to lift three times the weight of a rack and the other components of the lifting device without exceeding the minimum yield strength of the material. Also, the gate rigging will be designed to lift five times the weight of a rack (about 9-tons) without exceeding the ultimate strength of the rig materials. This is a very conservative design since the crane lifting loads would be much less (less than 10-tons) than 5 times the weight of a rack.

The licensee stated that the current license does not allow fuel to be under the safe load path for movement of the gate. Therefore, the postulated gate drops were analyzed at 36 inches above empty fuel racks and at 40 feet above the SFP liner. The results of the analysis indicated that a gate drop would penetrate the racks to a depth of 5 to 7.45 inches with no impact on the fuel stored in areas adjacent to the load path. The SFP liner would be damaged by a drop of the gate, but the SFP concrete slab would not fail. Since the gate weight is much less than the rack weight, the breach in the SFP liner is expected to be smaller than the breach caused by a rack drop. Therefore, a breach in the SFP liner because of a gate drop would be mitigated by the leak chase system and the SFP makeup capability as discussed above.

NUREG-0612 recommends that licensees provide an adequate defense-in-depth approach to maintaining safety during the handling of heavy loads near spent fuel and cited four major causes of accidents: operator errors, rigging failures, lack of adequate inspection, and inadequate procedures. The licensee stated that they will implement measures using administrative controls and procedures to preclude load drop accidents by these major causes. Accordingly, the licensee plans to provide: (1) comprehensive training in accordance with Occupational Safety and Health Administration (OSHA) 29 CFR 1926, “Rigging Equipment for Material Handling,” OSHA 29 CFR 1910, “Slings,” and ANSI B30.1, “Jacks,” for the rigging operations personnel; (2) redundantly designed and adequately tested lifting rigs in accordance with ANSI N14.6, (3) inspection and maintenance checks on the cranes, lifting devices, and racks themselves prior to and during the rack installation; and (4) specific procedures that cover the entire rack installation effort, including the identification of required equipment, inspection, acceptance criteria prior to load movement, defining safe load paths, and steps and precautions for proper load handling and movement.

Based on the load drop analyses, the staff finds that the integrity of the fuel and the SFP would be maintained if a fuel assembly, a gate, or a spent fuel storage rack were dropped. The design and capability of the crane and lifting devices, in conjunction with the administrative procedures and controls that are focused on, but not limited to, the areas noted above, would enable the licensee to maintain safety during the rack installation.

2.8.5 TS 3/4.9.7 “Crane Travel - Spent Fuel Storage Areas”

FSAR Section 9.1.5 defines heavy load as loads in excess of 1,800 lbs. As noted above, the weight of the heaviest load to be moved during the rack installation is 18,050 lbs. (bounding weight); therefore, rack installation involves the movement of heavy loads.

TS 3.9.7 prohibits loads greater than 2,200 pounds from travel over fuel assemblies in the storage pool.

Technical Specification Surveillance Requirement (SR) 4.9.7 states:

“Crane interlocks and physical stops which prevent crane travel with loads in excess of 2200 pounds over fuel assemblies shall be demonstrated operable within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.”

The licensee proposes to change the Technical Specification SR as follows:

“Crane interlocks and physical stops which prevent crane travel with loads in excess of 2200 pounds over the fuel storage pool shall be demonstrated operable within 7 days prior to crane use and at least once per 7 days thereafter during crane operation. Administrative controls may be used in lieu of crane interlocks and physical stops for handling fuel racks, spent fuel pool gates, or loads less than 2200 pounds."

The proposed changes to SR 4.9.7 require the following: (1) the licensee verifies, within 7 days prior to crane use, and at least once per 7 days of crane operation, that the crane interlocks and physical stops prevent crane travel from over the spent fuel storage pool instead of only over fuel assemblies; and (2) the licensee may use administrative controls in lieu of crane interlocks and physical stops to prevent loads such as the fuel storage racks, the spent fuel pool gate, or loads less than 2,200 pounds from travel over the SFP.

Under the proposed changes, TS 3.9.7 continues to prohibit loads in excess of 2,200 lbs. (heavy loads) from travel over the fuel assemblies in the SFP in accordance with the guidance of NUREG-0612. One of the proposed changes to SR 4.9.7 would require that the licensee verify that the crane interlocks and physical stops are functioning to prevent travel over the SFP instead of only over fuel in the SFP. Therefore, this proposed change in SR 4.9.7 imposes more restrictive requirements for loads less than 2200 lbs.

The other proposed change to the SR supports the requirements of the TS because only the spent fuel storage racks, the gate, and loads weighing less than 2,200 lbs. may be administratively controlled to prevent them from travel over the SFP. Heavy loads would still be restricted from travel over the SFP and fuel in the SFP via crane interlocks and physical stops. Subsequent to the proposed change, the requirements of the TS will continue to satisfy the guidelines in NUREG-0612 and are, therefore, acceptable.

2.8.6 Conclusion on the Control and Handling of Heavy Loads

Based on the preceding discussions, the staff finds that the aforementioned considerations for the movement of heavy loads to support the proposed increase in the spent fuel pool storage capacity and changes to TS 3/4.9.7 are acceptable. The licensee's use of the cranes, the special lifting rig for the spent fuel racks, and the administrative controls and procedures enhance the licensee's capability to reduce the potential for load drops and is in accordance with NUREG-0612 and ANSI N14.6. The design, testing and inspection of the cranes, the hoist, lifting rig, and other lifting devices will assure that the hoisting system is highly reliable and enable the licensee to safely handle the racks and other heavy loads during the rack installation process. The licensee's postulated accident analyses of the spent fuel storage rack and the gate indicate that the spent fuel pool liner could be breached. However, due to the capabilities of the leak chase and SFP makeup systems, the licensee is capable of maintaining the SFP and its contents within the acceptable consequence limits set forth in the guidelines of NUREG-0612. The proposed changes to TS 3/4.9.7 maintain the restriction against the movement of heavy loads over fuel in the SFP in accordance with NUREG-0612. In addition, the proposed change to the TS to allow the licensee to use administrative controls instead of crane interlocks and physical stops to prevent crane travel over the SFP applies to the spent fuel storage racks, the gate and loads less than 2,200 lbs. The crane interlocks and physical stops are still required to prevent heavy loads from travel over the SFP.

3.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

3.1 Introduction

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility, in accordance with the amendment, would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

As required by 10 CFR 50.91(a), the licensee provided its analysis of the issue of no significant hazards consideration in its March 19, 1999, amendment request. The staff reviewed the licensee's analysis and, based on its review, it appeared that the three standards of 10 CFR 50.92(c) were satisfied. Therefore, the NRC staff proposed to determine that the amendment request involves no significant hazards consideration, and published its proposed determination in the Federal Register for public comment on September 7, 1999 (64 FR 48672).

The staff has completed its evaluation of the licensee's proposed amendment as discussed in Section 2.0 above. Based on its evaluation, the staff has determined that the proposed amendment does not significantly increase the probability or consequences of an accident previously evaluated; does not create the possibility of a new or different kind of accident from any accident previously evaluated; and does not involve a significant reduction in a margin of safety. The following staff evaluation in relation to the three standards of 10 CFR 50.92 supports the staff's final no significant hazards consideration determination.

3.2 First Standard

"Involve a significant increase in the probability or consequences of an accident previously evaluated."

The following postulated accidents and events involving spent fuel storage were identified and evaluated by the licensee. The staff likewise evaluated the same accidents and events.

1. dropped spent fuel storage rack and SFP gate
2. a spent fuel assembly drop
3. loss of SFP cooling flow
4. a seismic event
5. mis-loaded fuel assembly

The probability that any of the accidents in the above list can occur is not significantly increased by the proposed changes. The probabilities of a seismic event or loss of SFP cooling flow are not influenced by the proposed changes. The probabilities of accidental spent fuel storage rack, SFP gate and fuel assembly drops or mis-loadings are primarily influenced by procedures and equipment used for handling the fuel. Fuel handling is not a random event; it is strictly controlled using approved procedures, trained personnel, and specialized equipment. Using these methods, any increase in the probability of a fuel handling accident (spent fuel storage rack, SFP gate and fuel assembly drop or mis-loading) is not significant.

In its submittal, the licensee re-evaluated the consequences of the previously postulated scenarios for an accidental drop of a spent fuel storage rack , SFP gate or fuel assembly in the SFP. The licensee found that the structural damage as a result of a drop to the SFP, pool liner, and fuel assembly is primarily dependent on the mass of the falling object and the drop height. Since these two parameters are not changed by the proposed activity from those considered previously, the structural damage to these structures/hardware remains unchanged. Therefore, there is no increase in the consequences of an accident previously evaluated.

Similarly, the radiological dose at the exclusion area boundary will not be increased from those previously considered, since the pertinent fuel parameters remain unchanged. The staff reviewed the licensee's analysis as discussed in Section 2.3 of this safety evaluation. Based on its review, the staff concluded that the bounding scenario for the postulated fuel handling accident does not change due to the addition of storage racks in the SFP. On this basis, the staff concluded that the consequences of this type of previously evaluated accident are not significantly increased by the proposed change.

The staff evaluated the consequences of a loss of SFP cooling event in Section 3.2 of the safety evaluation (in NRC letter dated September 12, 2000, Accession No.: ML003744109) for the full-core offload amendment application dated January 18, 1999. Although the evaluation was done in conjunction with the full-core offload amendment, the evaluation applies to this SFP increased capacity amendment. On the basis of our evaluation, the staff determined that sufficient time is available for plant operators to take mitigating actions to restore cooling prior to the pool boiling. In addition, sufficient makeup capability is available should boiling occur. Thus, the consequences of this type of accident are not significantly increased from previously evaluated loss of cooling events.

The staff evaluated the consequences of a design basis seismic event in Section 2.4 of this evaluation. On the basis of our review, the staff concluded that the licensee's structural analysis and design of the racks are acceptable and that the SFP structures are adequate to withstand the effects of the applicable loads, including that of the SSE. The analysis and design are in compliance with the current licensing basis set forth in the FSAR and applicable provisions of the SRP and are, therefore, acceptable. Thus, the consequences of a seismic event are not significantly increased from previously evaluated events.

The staff evaluated the consequences of fuel mis-loading and mis-location events in Section 2.6 of this evaluation. In their evaluation, the staff states that, while most abnormal storage conditions will not result in an increase in the k_{eff}, it is possible to postulate events that could lead to an increase in reactivity. However, for such events, credit may be taken for the presence of soluble boron in the pool water (which is assured by chemistry procedures), based upon the double contingency principle that requires at least two unlikely, independent, concurrent events to occur before a nuclear criticality accident is possible. Therefore, since soluble boron is normally present in the SFP water, credit for soluble boron may be assumed in evaluating other accident conditions such as the mis-loading of fresh fuel. The licensee is required to maintain the soluble boron concentration in the pool at or above 800 ppm and is required to confirm this by periodic surveillance measurements. The negative reactivity credited to the boron more than offsets the reactivity addition caused by credible accidents. Thus, the consequences of fuel mis-loading and mis-location events are not significantly increased from previously evaluated events.

3.3 Second Standard

"Create the possibility of a new or different kind of accident from any previously analyzed."

As noted in various sections of this safety evaluation, the staff evaluated the proposed changes in accordance with appropriate NRC Regulatory Guides, SRP sections, and industry codes and standards. In addition, the staff has previously prepared several safety evaluations for increased SFP storage capacity applications that are similar to this proposal. No unproven techniques or methodologies were used in the analysis and design of the additional racks to be used in SFP. No unproven technology will be used in the installation of the additional racks. The proposed change involves additional consideration of existing analyses for control of heavy loads, and therefore, does not create a new or different kind of accident from any previously analyzed.

In its analysis, the licensee conservatively considered an accidental drop of a rack during rack installation in the SFP. The staff evaluated the handling of heavy loads and SFAs in Section 2.8 of this evaluation. On the basis of their review, the staff determined that the licensee's use of the 10-ton new fuel receiving crane, 10-ton new fuel handling crane, the 10-ton hoist, the spent fuel rack lifting rig, and administrative controls and procedures that are in accordance with NUREG-0612 and ANSI N14.6, are consistent with existing methods for control of heavy loads.

The reliability of the cranes, coupled with the design, testing and inspection of the cranes, the lifting rig and other lifting devices, will enable the licensee to handle the racks and other heavy loads during the rack installation process in a manner consistent with existing controls for movement of heavy loads. Thus, the proposed amendment does not create the possibility of a new or different kind of accident from any previously analyzed.

3.4 Third Standard

"Involve a significant reduction in a margin of safety."

The function of the SFP is to store the fuel assemblies in a subcritical and coolable configuration through all environmental and abnormal loadings, such as an earthquake or fuel assembly drop. The new rack design must meet all applicable requirements for safe storage.

In evaluating a potential reduction in the margin of safety, the licensee addressed the safety issues related to the expanded pool storage capacity in the following areas. The staff likewise evaluated the same areas.

1. material, mechanical, and structural considerations
2. nuclear criticality considerations
3. thermal-hydraulic and pool cooling considerations

The staff evaluated the material, mechanical, and structural considerations of the proposed amendment in Sections 2.4 and 2.5 of this safety evaluation. Based on its evaluation, the staff determined that the materials used in the fabrication of the spent fuel racks manufactured by Holtec are compatible with the SFP environment at MNPS3. The staff concluded, therefore, that the materials used in the new spent fuel storage racks meet the regulatory standards that form the basis for acceptance of the existing SFP design. With respect to mechanical and structural considerations, the staff concluded that the licensee's structural analysis and design of the spent fuel storage racks and the SFP structures are within existing margins for withstanding the effects of the applicable loads, including that of the SSE. The analysis and design are in compliance with the current licensing margins set forth in the FSAR. Thus, there is no significant reduction in a margin of safety related to the material, mechanical, and structural considerations.

The staff evaluated the nuclear criticality aspects of the proposed amendment in Section 2.6 of this safety evaluation. The NRC acceptance criterion for sub-criticality (SRP Section 9.1.2) is that the effective multiplication factor (k_{eff}) in the SFP storage racks when fully flooded by unborated water shall be no greater than 0.95, including uncertainties at a 95 percent probability, 95 percent confidence level (95/95) under all conditions. On the basis of their review, the staff determined that the analysis methods used meet this criterion. Therefore, the staff concluded that the criticality aspects of the proposed storage capacity expansion for the MNP3 SFP are within the current margins of safety. Thus, there is no significant reduction in a margin of safety related to nuclear criticality considerations.

The staff evaluated the thermal-hydraulic and pool cooling aspects of the proposed amendment in Section 3.2 of the safety evaluation (in NRC letter dated September 12, 2000, Accession No. ML003744109) for the full-core offload amendment application dated January 18, 1999. On the basis of their review, the staff determined that thermal margin is maintained by assuring the bulk fuel pool coolant temperature in the SFP is maintained within its design limit assuming a single active failure. In addition the staff concluded that sufficient time is available for plant operators to take mitigating actions prior to pool boiling in the event of a loss of SFP cooling. Thus, there is no significant reduction in a margin of safety related to thermal-hydraulic and pool cooling considerations.

On the basis of the above evaluation, the NRC staff concludes that the proposed amendment meets the three criteria of 10 CFR 50.92. Therefore, the staff has made a final determination that the proposed amendment does not involve a significant hazards consideration.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact was published in the Federal Register on September 7, 1999 (64 FR 48675) with a correction on December 15, 1999 (64 FR 70076) for this amendment. Accordingly, based upon the environmental assessment, the Commission has determined that 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

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