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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.40

TO FACILITY OPERATING LICENSE NO. DPR-21

NORTHEAST NUCLEAR ENERGY COMPANY

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 1

DOCKET NO. 50-245

1.0 INTRODUCTION

By letter dated June 24, 1988, and as supplemented by additional information dated July 29, August 12 and December 2, 1988 and February 14, March 1, March 22 and April 10, 1989, Northeast Nuclear Energy Company (NNECO or licensee) submitted a proposed change to the Millstone Nuclear Power Station, Unit No. 1, Technical Specification (TS). This change adds a new requirement to the TS, a limitation on the maximum number of spent fuel assemblies that can be stored in the spent fuel pool. The limitation corresponds to the increased capacity of the spent fuel pool.

This report presents the NRC staff's Safety Evaluation for the increase in capacity of the Millstone Unit No. 1 spent fuel pool. A notice of environmental assessment and finding of no significant impact was previously published in the Federal Register on June 15, 1989 (54 FR 25511).

1.1 Description of Spent Fuel Pool Capacity Expansion

Millstone Unit No. 1 is a boiling water reactor (BWR), licensed to operate at 2011 megawatts-thermal, which went into commercial operation in 1970. The plant has recently completed its twelfth refueling outage. Prior to this refueling outage, 1732 fuel assemblies had been discharged to the spent fuel pool, which has a capacity of 2184 fuel assembles. Thus, only 452 available spent fuel storage locations were available during the refueling , well below the 580 locations necessary for a full core offload. NNECO has proposed to expand the spent fuel pool (SFP) capacity such that the loss-of-full core discharge capacity will be extended to about 1999.

Specifically, NNECO proposes to install new spent fuel racks to increase the number of storage cells in the Millstone Unit No. 1 spent fuel pool by 1045 locations to a total of 3229 cells. In addition, the reracked pool will contain 20 locations for storing failed fuel containers.

The proposed racks are free standing and self-supporting. The existing racks, which will all remain in the pool, also have been seismically requalified as free standing structures. The principal construction materials for the new racks are ASTM 240, Type 340 stainless steel for the structural members and shapes, and Boraflex, a patented product of BISCO (a division of Brand, Inc.) for neutron absorption. The support legs are of remotely adjustable type, consisting of an internally threaded, stationary member and a rotatable spindle. The latter is made of SA564-630 precipitation hardened stainless steel to reduce the probability of thread galling, and to build in a relatively high bending and compressive strength capability in the foot pedestals.

NNECO engaged CBI Services of Oak Brook, Illinois, to manufacture the new racks, and Holtec International of Mount Laurel, New Jersey, to design, analyze and qualify them for all loading. Holtec International performed three-D time history, nuclear criticality safety, mechanical integrity, and thermal/hydraulic analyses on these racks.

1.2 Previous Spent Fuel Pool Capacity Expansion

The capacity of the Millstone Unit No. 1 SFP was previously expanded from 1100 to 2184 fuel assemblies in 1977. The staff's safety evaluation and environmental impact appraisal for this expansion was issued in Amendment No. 39 to the Provisional Operating License, dated June 30, 1977.

2.0 CRITICALITY CONSIDERATIONS

2.1 Licensee Analyses

The current Millstone Unit No. 1 TS for the spent fuel storage pool and existing racks (TS 5.5.B) states that the K-effective of the pool shall be less than 0.90. The specification further indicates that this k-effective value is satisfied if the maximum exposure dependent k-infinity of the stored fuel assemblies is less than 1.35. The licensee does not propose to change this specification and the pool criterion, thus, remains at 0.90 for the new racks. The only proposed change to TS 5.5 is to add a Section 5.5.C indicating that 3229 fuel assemblies may be stored in the pool.

NNECO proposes to store fuel assemblies with average U-235 enrichments of up to 3.8 percent in the new racks. Calculations for the new racks indicate that, with appropriate uncertainties added and with no credit for the gadolinium loading or burnup of the fuel, the k-effective of the racks approaches 0.90 (with a small margin remaining) with a fuel average enrichment of 3.05 percent U-235. This enrichment (and no gadolinium or burnup) corresponds to a k-infinity of 1.35. Thus, to store fuel with enrichment up to 3.8 percent, NNECO proposes to take credit for gadolinium or burnup, and has provided calculations and resulting curves which indicate requirements for gadolinium loading or burnup as a function of fuel average enrichment. This proposal is not directly in the form of a k-infinity criterion, but it is similar in concept.

The new rack design and safety analyses, including nuclear criticality analyses, were provided for NNECO by Holtec International. The rack design is a rectangular array of stainless steel "box" storage cells of 6.30 inch pitch and 6.06 inch inside dimension. There is a Boraflex sheet between each cell and on outside surfaces of the racks. The Boraflex is retained between adjacent cell surfaces (or outside cell cover sheet) and lateral top and bottom strips but is otherwise not fastened to the structure, so that it is not constrained from movement resulting from possible radiation induced dimension change.

The criticality and associated sensitivity calculations were done with both the Monte Carlo code AMPX-KENO (using the 27 group SCALE cross sections, with NITAWL), and, as the primary method, with the two-dimensional transport code CASMO-2E. These are commonly used methodologies and their use has been approved in a number of previous NRC staff reviews of fuel pool criticality.

The methodologies have been benchmarked against a number of relevant critical experiments simulating storage racks by Holtec (and by many other groups). These experiments have covered a range of geometries, material compositions, fuel enrichments and poison sheets. These benchmark calculations have been used to develop methodology bias and uncertainty factors to be added to the nominal k-effective calculations for the Millstone Unit No. 1 racks.

Holtec has also determined the potential variation of the rack and fuel parameters which are used in determining the k-effective of the rack-fuel system. These parameters include poison thickness, width and density, stainless steel thickness, channel effects including bulges, and fuel density, enrichment and eccentric fuel position. The variation of k-effective with these parameters (taken at a 95/95 probability/confidence level) was determined. These independent parameters were statistically combined with the methodology uncertainty to provide a delta-k uncertainty which was added to the base k-effective calculation.

Most rack calculations were done using three-dimensional infinite arrays of cells and infinite fuel lengths. Finite axial length effects for the fuel and Boraflex (including a reduced Boraflex length of three percent from irradiation) were examined using one-dimensional calculations. It was determined that, with such axial geometry effects, results fell within the reference design infinite array results. A reactivity effect for an assumed radiation induced reduction in the width of the Boraflex sheets was also calculated and added to the rack reactivity as indicated by the licensee in its April 10, 1989 submittal.

Holtec has investigated abnormal conditions which might be associated with the spent fuel pool. These included pool water temperature effects (reference temperature was 20°C), eccentric fuel locations, fuel rack lateral movement, dropped fuel assembly and placement outside of racks, and the effect of a missing Boraflex plate. Only the missing Boraflex resulted in a reactivity increase of 0.0027 delta-k.

The rack base calculation result for the 8x8 (R or EB) fuel with an average enrichment of 3.05 percent (k-infinity 1.35) was a k-effective of 0.883. Adding the bias and uncertainty corrections raises that to 0.895, and including the abnormal condition of the missing Boraflex plate (only positive reactivity effect) results in a total of 0.897, which is less than the 0.90 criterion.

For fuel enrichments greater than 3.05 percent (up to 3.8 percent), Holtec has calculated the reactivity effect of adding gadolinium to fuel pins (minimum of three per assembly) or of fuel burnup or both. From these calculations a curve of either minimum required gadolinium percent concentration or fuel burnup was developed as a function of fuel initial enrichment to meet the rack k-effective of 0.883 for the 3.05 percent fuel of the base design calculation. For example, for 3.8 percent fuel either a gadolinium concentration of at least 2.4 percent (in at least 3 fuel pins) or a burnup of at least 7.4 MWD/KgU would be required. The combined gadolinium and burnup calculations indicated (for all fuel enrichments) that when using the individual requirements in combination the peak rack k-effective as a function of burnup would be well below the base value of 0.883.

2.2 Criticality Evaluation

The basic criticality design of the new racks, using boron lined cells to provide the appropriate neutron multiplication level for the closer packed array of high density racks, is a commonly used concept and has been accepted for many spent fuel storage pools. It is an acceptable design concept for maintaining criticality levels for the Millstone Unit No. 1 pool.

The analytical methodologies used by Holtec to analyze the criticality and reactivity change characteristics of the racks are standard methodologies, commonly used and approved for other utilities for such analyses. The CASMO-2E code provides an acceptable methodology for base calculations and for sensitivity calculations, and the AMPX-KENO-SCALE code package provides suitable backup and confirmation calculations. These methods have been benchmarked against an appropriate selection of critical experiments, with results falling within expected ranges of deviations from the experiments. The derivation of the uncertainty of the methodology from this benchmarking follows normal procedures and also falls within an expected range. It is acceptable.

A burnup dependent uncertainty factor accepted in previous fuel pool reviews was included in the analysis of the burnup limit curves. Previous analyses have shown that normal fuel assembly design enrichment patterns result in decreased rack reactivity compared to the uniform average enrichment assumed in this analysis. Furthermore, the burnup calculations with gadolinium result in a margin of over 1 percent delta-k between the peak reactivity of the racks and that required to meet, with uncertainties and anomalies added, the rack k-effective limit of 0.90. This margin is sufficient to provide a fully acceptable margin for uncertainties in the enrichment pattern and burnup analysis. The examination of uncertainties to be attributed to variances in dimensions and materials in the fuel and racks has covered an acceptable range of parameters and has used a suitable, standard methodology for determining the reactivity effects and their statistical combination. The examination of the effects of abnormal conditions has covered the standard events relating to changes in temperature, movements, misplacement and dropping of assemblies and other equipment, and the results are reasonable and acceptable.

It is noted that the proposed required gadolinium limit is well below the range normally used for the enrichment range and the minimum burnup is below that expected in only one operating cycle, and thus operational errors in limits would not be expected.

Based on the review of the NNECO submittal and the Holtec calculations and results, it is concluded that, for the range of fuel enrichments considered and with the proposed limits on minimum gadolinium loading or fuel burnup, the 0.90 spent fuel pool maximum reactivity requirement is met for the new racks. Furthermore, it should be noted that the NRC approval of spent fuel racks has, for an extensive period of time, been based not on a pool k-effective of 0.90, but rather on 0.95. The Holtec calculations for an average fuel enrichment of 3.8 percent, and taking no credit for gadolinium or burnup (and thus requiring no uncertainty analyses in this area), meet the 0.95 criterion, including required uncertainty additions, abnormal conditions (missing Boraflex plate) addition and Boraflex width shrinkage. This could provide an alternate basis for the acceptability of storing up to 3.8 percent average enrichment in the new racks. However, such an alternative and the required corresponding change in the TS limit has not been requested by the licensee.

The first part of the existing TS 5.5.B, indicating a k-effective criterion of 0.90 for the spent fuel pool, is applicable for both the old and the new racks. However, the second part of the specification, stating that the criterion is satisfied if k-infinity of the fuel assembly is less than 1.35 is applicable to only the old racks and should not be considered as applying to the new racks. While this second statement is probably true for the new racks, this particular aspect has not been directly addressed in the submittal and is therefore not considered to be relevant to them. The proposed gadolinium and burnup limit curves are acceptable for the new racks and should become a part of procedures developed to assure that these limits are observed. By letter dated April 10, 1989, the licensee committed to modify TS 5.5.B to address this issue. That change will be addressed in separate license amendment. The currently proposed addition to the TS, 5.5.C on the increase in storage locations, is acceptable based on our criticality evaluation.

2.3 Conclusion on Criticality Considerations

The staff has reviewed the report submitted by NNECO proposing a Technical Specification change for the number of fuel assemblies which can be stored in the Millstone Unit No. 1 SFP and describing the calculations performed for the criticality analyses of the new fuel racks. Based on this review we conclude that appropriate documentation was submitted and that the proposed changes satisfy staff positions and requirements in these areas. The criticality aspects of the new spent fuel racks are acceptable.

3.0 MATERIAL COMPATIBILITY

Nuclear power plants provide storage facilities or pools for the wet storage of spent fuel assemblies. The safety function of the spent fuel storage pools is to maintain the spent fuel assemblies in a sub-critical array during all credible storage conditions. The staff has reviewed the compatability and chemical stability of the materials (except the fuel assemblies) wetted by the pool water, in accordance with Section 9.1.2 of the Standard Review Plan.

3.1 Storage Rack Design

The SFP at the Millstone Unit No. 1 contains oxygen-saturated demineralized water which is unborated. The pool is lined with stainless steel. The principal construction materials for the proposed new racks in the spent fuel storage pool are Type 304L stainless steel for the structure and Boraflex for neutron absorption. The racks are interconnected honeycomb arrays of square stainless steel boxes forming individual cells for fuel storage. All storage cells have Boraflex sheets on four sides.

Each Boraflex sheet was placed within a frame defined by the top, edge, and bottom spacer strips on each side of the storage cell. The Boraflex sheet is held in place by stainless steel plate sheathing and is not fastened to or glued onto any surface or structure. A single sheet of Boraflex panel was used on each side of the storage cell.

The licensee proposed a long-term surveillance program to ensure continued acceptable performance of the Boraflex. Surveillance specimens in capsules will be placed within selected racks. Initial examination of the surveillance specimens will be made about five years after exposure in the SFP. The examination will include visual inspection for damage, dimensional and weight measurements, hardness testing, and neutron absorption. If degraded Boraflex is found, corrective actions could include abandonment of the affected storage location, lattice storage strategies, or insertion of solid neutron absorbers.

3.2 Materials Evaluation

The stainless steel in the storage pool liners and rack assemblies is compatible with the oxygen-saturated water and radiation environment of the spent fuel pool. In this environment, corrosion of Type 304L stainless steel is not expected to exceed a rate of 6×10^{-7} inch per year (E. G. Brush and W. L. Pearl, "Corrosion and Corrosion Product Release in Neutral Feedwater," Corrosion, Vol. 28, p. 129, April 1972). This corrosion rate is negligible for even the thinnest stainless steel walls in the rack assemblies. Contact corrosion or galvanic attack between the stainless steel in the pool liners or rack assemblies and the Inconel/Zircaloy in the fuel assemblies to be stored will not be significant, because all these materials are protected by passivating oxide films. Boraflex is composed of non-metallic materials and, therefore, will not develop a galvanic potential with the metal components.

Space is available to allow escape of any gas which may be generated from the polymer binders in the Boraflex during heating and irradiation, thus preventing possible bulging or swelling of the Boraflex assemblies. Boraflex, an elastomer of methylated polysiloxane filled with boron carbide powder, is used as a neutron absorber (poison) in the spent fuel storage facilities of many nuclear power plants. It has undergone extensive testing to determine the effects of gamma radiation in various environments and to verify its structural integrity and suitability as a neutron absorbing material (Bisco Products, Inc., Technical Report No. NS-1-001, "Irradiation Study of Boraflex Neutron Shielding Materials," August 12, 1981). The evaluation tests have shown that Boraflex is unaffected by the pool water environment and will not be degraded by corrosion. Tests were performed at the University of Michigan, exposing Boraflex in deionized water to 1.03×10^{11} rads of gamma radiation with a concurrent neutron flux of 8.3×10^{13} neutrons/cm^2/sec. These tests indicate that Boraflex maintains its neutron attenuation capabilities after being subjected to an environment of water and gamma and neutron irradiation. However, irradiation caused some loss of flexibility and shrinkage of the Boraflex.

Long-term water soak tests at high temperatures were also conducted (Bisco Products, Inc., Technical Report No. NS-1-002, "Boraflex Neutron Shielding Material Product Performance Data," August 25, 1981). The tests show that Boraflex withstands a temperature of 240°F for 251 days without visible distortion or softening. The Boraflex showed no evidence of swelling or loss of ability to maintain a uniform distribution of boron carbide. The maximum bulk temperature of the spent fuel pool water at Millstone Unit No. 1 under normal operating conditions will be approximately 137°F which is well below the 240°F test temperature. In general, the rate of a chemical reaction decreases exponentially with decreasing temperatures. Therefore, the staff does not anticipate any significant deterioration of the Boraflex at the normal operating conditions of the pool over the design life of the spent fuel racks.

The tests (ibid) have shown that neither irradiation, environment, nor Boraflex composition have a discernible effect on the neutron transmission of the Boraflex material. The tests also have shown that Boraflex does not possess leachable halogens that might be released into the pool environment in the presence of radiation. Similar conclusions are reached regarding the leaching of elemental boron from the Boraflex. Boron carbide of the grade normally present in the Boraflex will typically contain 0.1 weight percent of soluble boron. The test results have confirmed the encapsulation capability of the silicone polymer matrix to prevent the leaching of soluble species from the boron carbide.

Recently, anomalies ranging from minor physical changes in color, size, hardness, and brittleness to formation of gaps (separation) of up to four inches in width were observed in Boraflex panels that have been used in the spent fuel pools of four nuclear power plants. The exact mechanisms that caused the observed physical degradations of Boraflex have not been confirmed. The staff postulates that gamma radiation from the spent fuel initially induced crosslinking of the polymer in Boraflex, producing shrinkage of the Boraflex material. When crosslinking became saturated, scissioning (a process in which bonds between atoms are broken) of the polymer predominated as the accumulated radiation dose increased. Scissioning produced porosity, which allowed the spent fuel pool water to permeate the Boraflex material. Scissioning and water permeation could embrittle the Boraflex material. In short, gamma radiation from spent fuel is the most probable cause of the observed physical degradations, such as changes in color, size, hardness, and brittleness. The staff does not have sufficient information to determine conclusively what caused the gap formation in some Boraflex panels. However, it is conceivable that if the two ends of a full-length Boraflex panel are physically restrained, then shrinkage caused by gamma radiation can break up the panel and lead to gap formation.

The staff determined that reasonable assurance exists that physical restraints are absent in the Boraflex panels of Millstone Unit No. 1 because the Boraflex sheets are not physically fastened to or permanently glued onto any structure. It is not likely that gaps will form to any significant extent in the Boraflex panels during the projected life of the Boraflex assemblies. However, minor physical degradations can take place in the Boraflex from gamma radiation in the spent fuel pool.

In the unlikely event of gap formation in the Boraflex panels that would lead to loss of neutron absorbing capability, the monitoring program will detect such degraded Boraflex panels, and the licensee will have sufficient time to perform a criticality evaluation and to take the appropriate corrective action.

3.3 Conclusions on Material Compatibility

Based on the above discussion, the staff concludes that the corrosion of the SFP components due to the pool environment should be of little significance during the design life of the facility. Components in the SFP are constructed of alloys which have a low differential galvanic potential between them and have a high resistance to general corrosion, localized corrosion, and galvanic corrosion.

The staff further concludes that the environmental compatibility of the materials used in the spent fuel storage pool is adequate based on the test data cited above and actual service experience at operating reactor facilities.

The staff has reviewed the proposed surveillance program for monitoring the Boraflex in the spent fuel storage pool and concludes that the program can reveal deterioration that might lead to loss of neutron absorbing capability during the design life of the spent fuel storage racks. However, if a significant loss of neutron absorbing capability is found in any Boraflex panel, the licensee could take corrective actions such as restriction of use of the affected cell for fuel storage or insertion of additional neutron absorbers.

The staff finds that the proposed monitoring program and the selection of appropriate materials of construction by the licensee meet the requirements of 10 CFR 50, Appendix A, General Design Criterion 61 regarding the capability to permit appropriate periodic inspection and testing of components, and General Design Criterion 62 regarding prevention of criticality by the use of neutron absorbers and by maintaining structural integrity of components, and are, therefore, acceptable.

4.0 STRUCTURAL CONSIDERATIONS

The acceptability of the removal of the seismic restraints and of the expanded fuel pool storage capacity required a detailed evaluation of the building and of the subject spent fuel rack modules for the postulated FSAR loads, including the applicable seismic loads. Also, the licensee was required to re-evaluate the racks to address the concern regarding potential impact between adjacent racks based on the new gap dimensions resulting from the fabrication and orientation of the new spent fuel racks.

4.1 Evaluation

The licensee had qualified the existing spent fuel racks as part of six supermodules braced against the walls of the pool and against each other, so that deformation of the racks need not be considered. Due to the proposed removal of the seismic restraints, each of the supermodules and new racks have been requalified as single rigid racks. The rack modules are assumed to be free standing and subject to a three-dimensional time history analysis. This evaluation utilizes the DYNARACK computer code, which has been utilized in previous free standing spent fuel rack analyses accepted by the staff. The analyses incorporate the hydrodynamic effects and account for the appropriate rack-to-rack and rack-to-wall spacings. Simulation for rack-to-rack and rack-to-wall impacts have been performed by the licensee utilizing two bounding coefficients of friction of 0.8 and 0.2.

The supermodule rack, an existing rack configuration, has a maximum cross section of 143" x 146" and can accommodate 440 fuel assemblies. The larger of the new racks (Rack D) has a cross section of 102" x 50" and contains 128 fuel assemblies. Each fuel assembly has a weight of 680 pounds. The 440 cell supermodule is the largest among the supermodules. It produces the maximum rattling mass inertia, and the maximum stress levels. The licensee has also documented the resulting stresses and displacements for new racks modules A and D. These evaluations have considered several partial and full fuel loads for the racks. The licensee has identified and evaluated the maximum load, stress and displacement values, even though they may not occur at the same instant of time.

During the site visit of August 1, 1988, the staff requested additional information related to several concerns. The requested information included: the spent fuel pool floor's response time history, the justification of minor moment and shear overstress for the spent fuel pool floor, the evaluation of the liner for effect of the motion of the feet of the rack modules, the stress level resulting from the 1600 pound force developed between the fuel assemblies and the cell walls, and the load distribution of the fuel racks on the spent fuel pool floor for the actual loading conditions.

The licensee provided the applicable site specific design spectra utilized in previous evaluation at Millstone Unit No. 1 and the resulting acceleration time histories at the ground level and at elevation 65.75 ft., which is the elevation of the spent fuel pool floor. The staff has determined that the peak accelerations identified in these time histories agree with those utilized in the evaluation of the spent fuel rack modules. This is acceptable.

The analytical results are presented in terms of stress ratios (R) of actual stress to allowable stress. There are six stress ratio categories, as follow:

R_1 = Ratio of direct tensile or compressive stress on a net section to its allowable value.

R_2 = Ratio of gross shear on a net section to its allowable value

R_3 = Ratio of maximum bending stress due to bending about the x-axis to its allowable value for the section

R_4 = Ratio of maximum bending stress due to bending about the y-axis to its allowable value

R_5 = Combined flexure and compressive factor

R_6 = Combined flexure and tension (or compression) factor

The results were reviewed to determine that they meet the acceptance requirements of less than one for normal condition and less than two for severe accident conditions which include the seismic condition (safe shutdown earthquake). The licensee's response of December 2, 1988, as amended by their submittal of February 14, 1989, provided data addressing the controlling movement of the new racks and the available gaps.

Rack D is considered to be the controlling rack. The smallest rack-to-rack gap, assuming out of phase rack displacement, occurs for an original gap of 0.8125" and a total out of phase displacement of 0.70" (twice 0.35"). This condition indicates that after the worst postulated seismic motion, no rack-to-rack impact will take place. The results indicate that no impacts occur with either adjacent racks or with adjacent walls.

The structural evaluation of the spent fuel pool included the reinforced concrete, the reinforcing steel and the liner and its anchoring components. NNECO's analyses indicated a minor overstress due to the moment at the center of the spent fuel pool slab and due to shear at the edge adjacent to the drywell. However, these overstresses are 5% and 1% for the moment and shear, respectively. Also, these minor overstresses apply to the fully loaded condition of 3249 cells and a 150-ton cask. The licensee has addressed these minor overstressed conditions.

The minor overstresses for moment at the center of the pool slab and shear at the edge of the pool have been responded to by the licensee in their submittal of August 12, 1988. The 5% moment exceedance would indicate yielding of the tensile steel reinforcement at center of the slab when using a minimum rebar yield value of 40 ksi. However, the licensee has demonstrated by reference to "Design of Multistory Reinforced Concrete Building for Earthquake Motion," by Blume, Newmark, and Corning, that 45 ksi minimum yield stress can be expected for an average grade reinforcing bar. NNECO also argued that redistribution of loads would result in a lower percentage of overstress and the load combination that controls the slab design is primarily dominated by the temperature gradient across the slab of 212°F (inside) and 60°F (outside). This gradient is considered conservative.

With respect to the 1% exceedence in shear at the edge of the slab for the seismic load combination, the licensee stated that the shear load was based on maximum postulated values of dead and live loads, although they may never be superimposed during the life of the plant. Also, the allowable values were calculated utilizing a rather conservative slender section criterion of the ACI-349 design code (A member with a 10:1 span to thickness ratio). Therefore, the above indicated 1% exceedance in shear is judged as not significant.

The licensee has addressed the effects of the horizontal forces exerted by the supports on the liner for the critical location and loading. NNECO has determined that this condition occurs for the new racks only, at intersection of the new racks D-1, D-2, D-3 and E. This evaluation considers the thermal effects of 212°F and the reduced liner material stress allowable of 90% of the yield value. The results indicate that liner integrity is maintained with adequate margins. The licensee has verified at staff's request that the resulting 1600 pound load between the fuel assembly and its cell produce stresses below the allowable for the material. NNECO has indicated the the fuel assembly stresses have been evaluated for the 1600 pounds load and that the allowable material stresses are not exceeded.

Finally, the licensee has addressed our concern regarding the measures taken to assure that all of the leg loads of the various rack modules are transmitted to the spent fuel pool floor and that they exert an evenly distributed load. NNECO has stated that leg location and height adjustment will be part of the QA for Category I Plant Operation Review Committee procedures. Also, plumbing of the rack modules and assurance of contact during installation will assure proper load distribution to the spent fuel pool floor.

The staff has performed an audit of the calculations for the rack modules performed by Holtec International, on October 20 and 21, 1988. The audit included the verification of the assumptions used in the calculations, the computational methods, and the results, to determine if they were in agreement with the information presented in the safety analysis report, and with the acceptable industry and regulatory practices and acceptance criteria. Also, the staff ascertained that the seismic floor response spectra values and other loads used in the evaluation of the racks and other physical properties were properly accounted for in the dynamic and static models of the rack modules. We also confirmed that the limiting case rack module and loading conditions, including the proper range in sliding coefficients of friction, were utilized in the analyses.

We found that Holtec International's analysis met the requirements identified in the initial safety analysis report. The results of the analysis confirmed that the stress values reported by the licensee are in agreement with the results of the analyses audited by the staff. All of the applicable stress ratios (R_1 through R_6) are below the stress allowables identified in regulatory acceptance criteria of less than 2 for ASME Code, Section III, Subsection NRF, Level D Service Limits conditions and less than 1 for normal conditions. Actually a large portion of the stress ratios were found to be less than 50% or 75% of the allowable values.

Also, the licensee has addressed and dismissed the need to evaluate a multi-rack condition based on low fuel assembly weight of 680 lbs/cell, and the fact that rack movement diminishes with reduced fluid gap between the racks. A fluid gap of six inches or greater approximate a free standing rack in fluid medium. The smallest gap is identified as 0.8125" for the Millstone spent fuel pool.

A multi-rack analysis is usually performed to confirm that rack-to-rack or rack-to-wall impact loads as predicted by a single rack 3-D analysis are of the correct order of magnitude. Previous industry studies have found that if rack-to-wall impact does not occur, then all racks in a line move relatively in phase and will not impact, even though the individual rack displacement may be greater than that predicted in the single rack analysis. The single rack analysis model described in the Millstone Unit No. 1 evaluation of a single free standing rack constitutes a more severe scenario than would a multi-rack analysis, since Millstone Unit No. 1's model assumes that racks move 180 degrees out-of-phase. Consequently, the single rack analysis would predict impact if the rack being analyzed advances towards an adjacent rack by more than 50 percent of the initial gap between them. In multi-rack analysis, no assumption regarding the relative module motions need be made and an individual module may move greater than 50 percent of the gap without colliding with the neighboring rack.

The rack-to-wall impact simulated by a single rack model is not regarded as the most conservative representation. During the meeting of January 13, 1989, the staff requested NNECO to provide a comparison of the Millstone Unit No. 1 single rack analysis with Vogtle's multi-rack analysis that accounted for the final as-built dimensions for the new spent fuel racks.

The Vogtle multi-rack analysis was performed for a row of four modules, assuming rack-to-liner coefficient of friction equal to 0.5. The results of this analysis, when compared with the Millstone Unit No. 1 single rack analysis, demonstrated that the Millstone scenario provides additional conservatism to the Vogtle's results. Also, we need to acknowledge that the seismic design requirements for Millstone Unit 1 are less severe than for Vogtle site.

The staff has visited the site and has evaluated the assumptions, models and available results developed by NNECO and their consultants. These evaluations have included discussions with the licensee's technical staff, the technical staff of URS/Blume for the structures and components, and the technical staff of Holtec International for the evaluation of the racks. The results were generally found by the staff to meet the acceptance requirements for the proposed spent fuel pool expansion. However, based on the smaller gap between the new racks the staff is requesting that the licensee inspect the rack spacing following a seismic event greater or equal to an operating basis earthquake (OBE).

4.2 Structural Conclusions

The licensee has addressed all of the staff concerns relating to the adequacy of the SFP floor's time history, the reported minor moment and shear overstresses at the center and at the edges of the pool slab, the proper distribution of the rack's load on the fuel pool, the effects of rack's motion on the integrity of the liner, and the clarification of the adequacy of the individual cell straps to resist the maximum postulated 1600 pounds load. Also, the site visit and the audit of the analysis have been completed leaving only one safety issue related to the inspection of the spacing of the racks following a seismic event equivalent to plant OBE. The staff requests that the licensee modify the applicable plant procedures which govern activities after an OBE to include a requirement to perform an evaluation of the SFP to check the rack spacing configuration. The licensee has indicated agreement to implement a walkdown following the above defined seismic event.

Based on the evaluation of the licensee submittal, the supplementary information provided by the licensee, discussions with the licensee at meetings during the site visit, and the licensee's agreement to the disposition of the final safety issue related to the post OBE fuel rack space inspection, the staff concludes that the licensee's structural analyses of the existing spent fuel racks are in compliance with the acceptance criteria and consistent with current licensing practice and, therefore, are acceptable.

5.0 SPENT FUEL POOL COOLING AND LOAD HANDLING

5.1 SFP Cooling System

The SFP cooling system circulates water using two pumps, which take suction from the skimmer surge tanks and direct it to the heat exchangers, filter, demineralizer and back through diffusers to the bottom of the pool. No modification to the spent fuel pool cooling system has been proposed by the licensee.

The licensee calculated the decay heat loads of the spent fuel assemblies discharged to the pool in accordance with Branch Technical Position ASB9-2, "Residual Decay Heat Energy for Light-Water Reactors for Long-Term Cooling," and Standard Review Plan (SRP, NUREG-0800) Section 9.1.3, "Spent Fuel Cooling and Cleaning System." The licensee also analyzed the resulting temperature for normal discharge heat loads given a single active failure and the resulting temperature for abnormal discharge heat loads.

The thermal output of Millstone Unit No. 1 is 2011 Mwt from its 580 fuel assemblies. The total proposed storage capacity of the spent fuel pool is 3229 assemblies. In order to bound the results, all calculations were carried out for a time in the future when the pool has just enough open storage locations to accept a full core discharge. The licensee assumed that the total exposure period was 1825 days and that a normal discharge is one third of the core (196 fuel assemblies).

The licensee analyzed two scenarios of fuel offload to the spent fuel pool, as follows:

(a) Normal discharge - to obtain an upper bound on the normal discharge heat generation rate, it was assumed that 17 batches of 145 assemblies each are discharged at 372 day intervals with 1488 days exposure and that the latest batch of 204 assemblies (an upper bound) with 180 days exposure is discharged after 150 hours of decay.

(b) Abnormal discharge - a full core offload that takes place 3 months after the latest normal discharge and the fuel transfer starts after 250 hours of decay.

The licensee stated that it used a methodology consistent with the criteria of the SRP to compute the heat dissipation requirements in the spent fuel pool. In addition, the licensee assumed that the heat exchangers were fouled, no credit was taken for improvements in the film coefficient of the heat exchangers as temperature increased, no credit was taken for heat loss due to evaporation of pool water and no credit was taken for heat loss through the structure. These assumptions helped to simplify the calculations and to ensure a bounding analysis.

The licensee's results indicate that the maximum spent fuel pool water bulk temperature for a normal discharge with no single failure is 137°F and with a single failure (pool pump) is 140.4°F, which are both below the acceptance criteria limit of 150°F. In the case of an abnormal discharge with no single failure, the maximum pool water bulk temperature is 121.2°F, which is below the acceptance criteria of 212°F. In this case, as is normal in this situation, the licensee had one train of the shutdown cooling system acting in parallel with the spent fuel pool cooling system.

The staff notes that the licensee's calculation of the normal refueling discharge heat load case does not assume that the spent fuel pool is full, but rather reserves 560 empty cells. Therefore, the staff has made an independent assessment which confirms that adequate heat removal capability is available in the event these cells are filled with spent fuel since their contribution to the total heat load is negligible.

The staff has reviewed the licensee's analysis regarding the adequacy of the spent fuel pool cooling system in light of the proposed increase in storage capacity and finds that this cooling system meets the criteria in the SRP and is, thus, acceptable.

5.2 Control of Heavy Loads

The spent fuel racks are considered to be heavy loads since they weigh more than a spent fuel assembly and its handling tool. The reracking of the SFP involves the movement of the existing racks to the southwest corner of the pool and the installation of 10 new high density racks using 100 ton overhead crane. The licensee stated that the movement of the spent fuel racks will be performed in accordance with NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants, Resolution of Generic Technical Activity A-36," which provides guidelines for reracking of spent fuel pools. Crane interlocks and limit switches have been provided to prevent heavy load travel over irradiated fuel assemblies.

As a result of its review, the staff finds that heavy load handling will be performed in accordance with the guidelines of NUREG-0612 and, therefore, the requirements of GDC 61 and 62 are met as they relate to proper load handling to ensure against an unacceptable release of radioactivity or a criticality accident as a result of a heavy load drop.

5.3 Conclusion on Cooling and Load Handling

Based on the above, the staff concludes that the proposed SFP storage capacity expansion is acceptable with respect to the SFP cooling system and handling of heavy loads.

6.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact has been prepared and published in the Federal Register on June 15, 1989 (54 FR 25511). Accordingly, based upon the environmental assessment, we have determined that the issuance of the amendment will not have a significant effect on the quality of the human environment. A "Notice of Consideration of Issuance of Amendment to Facility Operating License and Opportunity for Hearing" was published in the Federal Register on August 23, 1988 (53 FR 32124). No hearing requests were received.

7.0 CONCLUSION

We have 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, and (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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