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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 144 TO FACILITY OPERATING LICENSE NO. DPR-36

MAINE YANKEE ATOMIC POWER COMPANY

MAINE YANKEE ATOMIC POWER STATION

DOCKET NO. 50-309

1.0 INTRODUCTION

The present Maine Yankee (MY) spent fuel storage racks provide adequate capacity for storage of 1476 spent fuel assemblies in a single region rack configuration. However, beginning with the 1996 refueling, MY would lose full core discharge capability with the existing racks. Therefore, to preclude this situation and ensure that sufficient spent fuel storage capacity continues to exist at its plant, the licensee has applied to install high density spent fuel storage racks. The design of these racks incorporates Boral as a neutron absorber in the cell walls, thereby allowing for closer, more dense storage of spent fuel. The new racks would provide 2019 storage locations in a two-region arrangement. Region I would consist of 228 locations, and Region II would consist of 1791 locations. The present fuel design and arrangement is capable of storing 1476 spent fuel assemblies.

By letter dated January 25, 1993, as supplemented by letters dated November 3, 23, and December 9, 1993, and January 5 and 24, 1994, the Maine Yankee Atomic Power Company submitted a request for changes to the Maine Yankee Atomic Power Station Technical Specifications (TS). The November 3 and 23, and December 9, 1993, and January 5 and 9, 1994, letters were in response to Request for Additional Information (RAI) letters sent by the NRC staff on October 7, 25, and 28, 1993. In addition, the licensee responded to a December 21, 1993, 10 CFR Part 21 (Reporting of Defects and Noncompliance) letter regarding the ANSYS computer code, on January 5, 1994. The licensee's letters of response provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The following evaluation presents the NRC staff's consideration of each technical aspect of the licensee's amendment request to rerack its spent fuel pool (SFP).

2.1 Heavy Loads

2.1.1 Rack Weights

The licensee notes that the heaviest rack to be removed from the SFP weighs 22,400 lbs., while the heaviest new rack to be installed weighs 13,000 lbs.

When positioned on the floor of the SFP, there will be approximately 22 feet of water over the top of the racks and their weight will be partially offset by the weight of water displaced.

2.1.2 Yard Crane

The yard crane is load rated at 125 tons, and is fitted with an auxiliary hoist whose load rating is 20 tons. (The auxiliary hoist will not be used to lift racks or fuel, nor will it carry any loads over fuel or safety-related equipment during the reracking project.) The yard crane will be used to move racks and the temporary gantry crane between the cask loading pit (a 10-foot square, 2-1/2 foot depression in the floor of the west side of the SFP) and the cask decontamination area. The yard crane will not be used to move loads over the SFP, except within the area of the cask loading pit.

The yard crane meets the single-failure-proof guideline, because the ratio between its ultimate load capability (500 tons) and the 30 ton lift restriction during reracking is approximately 17:1. (The guideline for considering such a crane single-failure-proof is a ratio of 10:1.)

2.1.3 Temporary Gantry Crane

The temporary gantry crane alone will be used to move racks between the cask loading pit and their installation location on the floor of the SFP. Old racks will first be emptied of spent fuel assemblies; new racks will be fully installed before being loaded with assemblies. Table 1 presents the load bearing parts of the temporary gantry crane, along with the ultimate load each part can sustain before failure (ultimate stress). It must be noted that for the end trucks and gantry legs, the load shown must be multiplied by 2 (for the total of two end trucks and two gantry legs per beam). These totals are shown in parentheses.

TABLE 1
TEMPORARY GANTRY CRANE

CRANE PART	MAXIMUM LOAD LIFTED BEFORE ULTIMATE STRESS IS REACHED
Bridge (double beam)	80 tons
End Trucks (each 2 per beam)	80 tons each (160 tons, total)
Trolley	80 tons
Gantry Legs* (each 2 per beam)	80 tons each (160 tons, total)
Hoist Unit**	200 tons

* Short gantry legs are used to maintain bridge, trolley and hoist unit clearance above the SFP curb and other interferences.

** The capability of the single failure proof hoist unit exceeds that of other crane parts to meet the criteria of NUREG-0554. Overload protection is provided to prevent lifting any load that would approach the yield strength of a crane part.

The temporary gantry crane hoist unit is considered single-failure-proof because the ratio between the ultimate load capacity of its two hoist units (200 tons) and the maximum load for this reracking project (11.2 tons for an old fuel rack) is approximately 17.9 to 1.

The licensee notes that temporary gantry crane parts are designed to the following standards:

Crane Part	Design Standard
Bridge & end trucks	NUREG-0554, ASME NOG-1, CMAA-70, ASME B30.2
Trolley	NUREG-0554, ASME NOG-1, CMAA-70, ASME B30.2
Gantry legs	NUREG-0554, ASME NOG-1, CMAA-70, ASME B30.2
Hoist unit	NUREG-0554, ASME N14.6, CMAA-70, ASME B30.2

2.1.4 Special Lifting Device (Rack Lifting Rig)

The licensee will use a lifting rig having two sets of four legs; a short set of legs 4 feet long, and a long set of legs 16 feet long. The short legs will be attached to old racks (at the top) for removal. The long legs will be attached to new racks (at the bottom) for installation in the SFP. The rack lifting rig is capable of lifting 150 tons with either set of legs before exceeding its ultimate strength. It also can lift a load of 90 tons before exceeding its yield stress.

The rack lifting rig will meet the provisions of guideline 4 (Special Lifting Devices) of Section 5.1.1 (General) of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" and will comply with the requirements of ANSI 14.6-1978, "Special Lifting Devices for Shipping Containers Weighing 10,000 pounds (4500 Kg) or More." The lifting rig will also comply with the requirements of Section 5.3, "Testing to Verify Continuing Compliance," of ANSI 14.6.

2.1.5 Lifting Devices Not Specially Designed

After the old racks have been cleaned and suitably decontaminated, they will be placed in bags in the decontamination pad area. Slings will be used to raise the racks and move them to a special container and a flatbed truck for shipment. The path from the decontamination pad to the truck is a safe load path. That is, neither spent fuel nor equipment required for safe shutdown lies along or below the path.

Slings will be used to bring new racks into the plant. The slings will be designed in accordance with the guidelines of NUREG-0612.

2.1.6 Conclusion--Heavy Loads

Based on the foregoing evaluation and the licensee's compliance with applicable guidelines for cranes, movement of heavy loads over and near spent nuclear fuel, and specially and nonspecially designed lifting devices, the NRC staff finds the licensee's amendment acceptable as it applies to the area of heavy loads.

2.2 Thermal-Hydraulic Aspects

2.2.1 Spent Fuel Pool Cooling System

The SFP cooling system consists of two SFP cooling pumps and one heat exchanger. Water is taken from the SFP, pumped through the heat exchanger, and returned to the SFP. Component cooling water flowing through the shell side of the heat exchanger is used to cool the SFP water. Each SFP cooling system pump is rated at 750 gpm.

The SFP cooling system also contains a purification loop, which consists of a pump, two filters and a demineralizer. The purification pump can take its suction from either the SFP cooling system suction or discharge line.

2.2.2 Decay Heat Calculation

The licensee's full core offload procedure does not allow a total SFP heat load in excess of the design heat removal capability of the SFP cooling system (22,000,000 BTU/HR), or a bulk SFP water temperature above 154 °F. Using the most restrictive assumptions, the licensee has calculated that at the conclusion of the last operating cycle, offloading the entire core into the SFP will exceed the SFP design heat removal capability for the first 23 days.

The licensee also has considered a less restrictive decay heat removal circumstance, in which a normal reload complement of 73 spent fuel assemblies is placed in the SFP to reach the total maximum SFP storage capacity of 2019 assemblies. In this case, the calculated heat generation rate is 15,300,000 BTU/HR.

2.2.3 SFP Water Temperature

As stated above, the maximum SFP water temperature is 154 °F when the most restrictive SFP loading assumptions are made and two SFP cooling system pumps are running. With one SFP cooling system pump running, the maximum SFP water temperature is calculated to be 177 °F. Similarily, in the less restrictive circumstance considered above, the maximum SFP water temperature with one SFP cooling system pump running is 134 °F.

The licensee, in accordance with previously approved procedures, will move spent fuel assemblies into the SFP at a rate that will maintain SFP water temperature at 154 °F or less, or will return spent fuel assemblies to the reactor vessel as necessary to ensure SFP water temperature does not exceed 154 °F.

2.2.4 Fuel Pin Cladding Temperature

The licensee has calculated fuel pin cladding temperatures using the RETRAN code. Use of the RETRAN code is consistent with methods reviewed and approved by the NRC staff in previous licensing submittals. The licensee's model includes downward flow only between the racks and SFP wall, and assumes the minimum acceptable spacing allowed between racks and the SFP wall. Downward flow within or between racks is neglected. The racks are assumed to be filled with fuel assemblies containing a spacer grid having bounding flow resistance. Heat transfer between cells is ignored. For additional conservatism, the licensee assumes that the fuel assemblies each operated at 150% of the average power level of all fuel assembies in the core.

The licensee calculated the results for two cases; one spent fuel assembly having normal flow, and one with its exit flow area reduced by 79%. The maximum cladding surface temperatures calculated for these cases were 239.1 and 239.2 °F, respectively.

2.2.5 Spent Fuel Pool Boiling

The cladding surface temperatures and location along the fuel assembly calculated for each of the the cases above indicated that minimum water level in the SFP prevented the occurrence of nucleate boiling. Thus, reactivity calculations assuming no void effects are conservative.

2.2.6 Loss of Spent Fuel Pool Cooling Cooling

a. Time to Reach Boiling

In the event that both SFP cooling pumps fail, the SFP bulk water temperature would rise until the water reached its boiling point of

212 °F. The licensee calculates that the minimum time for boiling to begin would be 6.5 hours, assuming the most restrictive case of a full core (217 assemblies) of spent fuel added to reach 2019 assemblies in the SFP. The resulting "boiloff" rate is calculated at 50 gpm.

b. Makeup Water

Makeup water is normally added to the pool by means of the chemical and volume control system (CVCS). In addition, there are three hose connections in the vicinity of the SFP that are supplied with primary grade water. In case neither of these two methods is available, the licensee can resort to makeup from the fire system. The licensee states that none of these methods requires longer than 20 minutes to initiate makeup flow, and both the CVCS and fire system remain available when offsite power is lost. Both the CVCS and fire system are capable of providing SFP makeup flow at greater than 150 gpm; each hose connection can supply 20 gpm.

2.2.7 Conclusion--Thermal/Hydraulic Aspects

The licensee's decay heat calculations for both normal refueling offload (73 spent fuel assemblies) and full core offload (217 spent fuel assemblies) to fill the SFP are found to be in accordance with applicable guidance, and thus are found acceptable.

The SFP bulk water temperatures, as determined by the licensee, are found to be in accordance with previously acceptable guidance for maximum bulk water temperatures during abnormal SFP cooling and purification system operation, and are, therefore, acceptable.

The fuel pin cladding calculational methodology and results are found to be acceptable. Because of the location of maximum temperature along the fuel assemblies, and the minimum allowed static head of water above the spent fuel assemblies, the maximum cladding temperature does not allow local (nucleate) boiling. Thus, there is no possibility of voids in the SFP water that could affect reactivity calculations.

The licensee calculated that a total loss of SFP cooling allows 6.5 hours prior to achieve SFP boiling. This appears sufficient either to reinitiate cooling, or provide makeup water to the SFP if cooling is not available. In either case, the availablity of methods to provide makeup water to the SFP at a rate sufficient to prevent lowering the water level in the SFP is determined to be adequate.

In view of the foregoing, the licensee's proposed amendment to rerack the SFP is found to be acceptable with regard to the resolution of any thermal hydraulic concerns.

2.3 Seismic Aspects

2.3.1 Spent Fuel Pool Loadings

The SFP is a Seismic Category I structure, the primary function of which is to provide safe storage for spent fuel assemblies. (New fuel is stored in a separate location, adjacent to the SFP.) The SFP is box shaped, but open at the top. The steel reinforced concrete walls and floor are 6-feet thick, and the floor and bottom 12 feet 6 inches of all four walls are supported by site bedrock.

The inside surface of the pool is lined with stainless steel for corrosion resistance, cleanliness and to ensure watertight integrity. A leakage detection system is provided to ensure all stainless steel seams maintain their watertight integrity.

SFP loadings consist of static, dynamic and thermal loads. Static loads include the weight of the pool structure, the water in the pool, and the weight of the fuel assemblies and their spent fuel storage rack modules. The reracking will result in an approximate 18% increase in the weight supported by the floor slab and bedrock foundation. Dynamic loadings consider both SSE and OBE (safe shutdown and operating basis) earthquakes. In addition, stresses generated by rack-to-wall hydrodynamic pressure effects caused by earthquake-induced rack movement are considered, and the effects of such rack motion on the steel pool liner due to an earthquake were incorporated in the licensee's analysis.

The SFP was analyzed using the ultimate strength concept. The licensee determined that the increased loads produced by additional spent fuel assemblies and racks, and the above described hydrodynamic pressure effects, are within the load carrying capacity of the SFP. The NRC staff has reviewed the licensee's calculations and confirmed that the loads applied are within the load carrying capacity of the SFP.

2.3.2 Fuel Handling Accidents

Section 4.2.2 of the Licensing Report accompanying the licensee's request for amendment provides a description of the fuel handling accident analysis. The description states that the weight of the fuel and operating constraints (restricting movement of fuel to a maximum of 18 inches above the rack) have not changed from the previous fuel handling accident analysis. (The previous fuel handling accident analysis was performed as part of the most recent license amendment associated with reracking the licensee's SFP (1983).) The licensee also states that the new spent fuel racks will not require any changes to refueling procedures, tooling or refueling equipment. In addition, the previously analyzed case of a 100 ton spent fuel shipping cask being dropped on the pool floor will continue to bound any applicable fuel assembly or fuel rack drop accident analysis, with regard to any damage to the SFP liner and structure. Therefore, the licensee concludes that the current rerack amendment request does not raise any new refueling related safety issue.

As to the possibility of an accidental drop of a fuel assembly directly onto another assembly stored in a fuel rack, review of the new rack design indicates that the top of the rack is sufficiently above the top of the fuel assemblies in the rack such that a vertical drop of an assembly directly impacting a stored assembly is very unlikely. If a direct impact were to occur, the chance of causing fuel cladding damage from such impact is judged insignificant. Additionally, it is the NRC staff's judgment that the upper structure of the rack and top end fitting of the impacted fuel assembly would provide sufficient energy absorption such that no damage to the impacted assembly would occur. Any damage to the falling assembly and upper structure of the rack could be repaired.

The licensee also states its fuel handling experience has been excellent. (During the initial fuel load and through 13 refueling outages, MY has never dropped a fuel assembly.)

2.3.3 High Density Spent Fuel Storage Racks

The new high density spent fuel storage racks are Seismic Category I and are, therefore, required to remain functional during and after an earthquake. A function of the rack is to maintain its structural integrity, as well as the structural integrity of spent fuel assemblies, while maintaining at least the minimum separation distance between adjacent stored fuel assemblies.

Individual storage cells are joined into a rack module. Like the old spent fuel storage racks, the new high density spent fuel storage racks are free standing; they are not anchored to the pool floor or the pool walls, nor are they structurally interconnected. Each rack module is provided with adjustable (leveling) support feet. Each fuel rack is a folded metal plate assembly of thin (14 gauge) metal, approximately 180 inches high, 117 inches wide and and 128 inches deep. The folded metal plate assembly is welded to a baseplate, which is supported by adjustable support feet.

The rack was modeled by a system of springs and lumped masses. There are many aspects of this modeling that make rack dynamic response analyses highly nonlinear. For example, there are gap spring elements that simulate gaps between a fuel assembly and a rack cell that constitute a nonlinear spring. Sliding elements are provided to model sliding action of the rack module along the SFP floor. The associated friction is nonlinear because friction depends on the relative velocity of the sliding surfaces. The licensee elected to model the sliding friction by a set of constant friction coefficients of 0.20 and 0.80.

SFP water is modeled by equivalent hydrodynamic masses. These elements are integrated into a computer model suitable for analysis by a computer code named "ANSYS." The ANSYS computer code executes time-history integration of the governing nonlinear differential equations of motion of a more than one hundred degrees of freedom system consisting of a rack and fuel assemblies.

The licensee also provided the results of the multirack analysis, as well as evaluations for different combinations of loading patterns, such as half empty and fully loaded racks. The licensee indicated that the two types of analysis make no significant difference in values for rack uplifting and sliding, and rack foot reaction forces.

Based on the above described analyses, the licensee concluded that the racks would neither impact each other, nor SFP walls. Stresses in components of the rack (fuel cells, baseplate, support feet and connecting welds) were found to be less than the allowables specified in ASME Boiler and Pressure Vessel Code Section III, Subsection NF, and therefore acceptable.

However, the NRC staff concluded that the verification process employed by the licensee for the analyses, including the ANSYS code, was limited. Specifically, there have been no realistic physical tests to verify the overall analytical results obtained by ANSYS. In addition, the licensee has not demonstrated that the errors from the analytical method associated with numerical integration of the governing nonlinear differential equation of motion are within an acceptable range, or that the stability of the analytical-numerical solution of the response is controlled or accounted for. For these reasons, the NRC staff performed additional independent assessments. The results of these staff assessments are discussed below.

The staff reviewed the safety margin for overturning a rack. It was found that the estimated safety factor for overturning a rack was greater than 5.0, thus exceeding the allowable factor of 1.1 provided in Standard Review Plan (SRP) Section 3.8.5. Another aspect of rack response that the staff considered was the structural integrity of the rack with regard to lateral impact. Maximum impact occurs in the extreme bounding condition when there is no friction between rack support feet and the pool floor. A simulation of this condition has been performed using boxes suspended inside a water-filled container. The configuration of the test was similar to MY's SFP and new high density fuel storage racks. With dynamic forces applied to the container to produce motion, the test results yielded no impact among the boxes and no impact between the boxes and the container wall. The test demonstrated that water between the boxes provided a significant cushion (hydrodynamic coupling effect) to prevent impact between closely spaced objects. Even allowing for a certain number of isolated impacts among the racks or between racks and pool wall in an actual SFP, the impact forces are not expected to be significant because of the demonstrated hydrodynamic coupling effect of water.

Finally, the licensee performed a calculation using a static lateral force, with the bottom of the rack assumed to be fixed to the floor. This calculation was requested by the staff because it is believed to eliminate the major portion of the uncertainties associated with a nonlinear evaluation and provides an upper bounding case of stresses in the rack. It is an upper bounding case because the sliding associated with a rack in the SFP would relieve stresses in the rack applied by the static lateral force. When the largest rack stress is compared to the allowable stress, a margin of approximately 1.6 exists to ensure that a conservative bounding analysis demonstrated structural integrity of the rack.

As stated above, a function of the fuel storage rack is to maintain the structural integrity of spent fuel assemblies. Once the structural integrity of the rack is assured, fuel assemblies are protected when worst-case conditions are assumed. The worst-case condition for fuel damage is when the impact force between a rack and a spent fuel bundle stored in the rack is large enough to produce permanent deformation of the grids that maintain physical separation of the fuel rods. Permanent deformation of the grids, if any, is judged not to lead to any significant fuel rod cladding damage, because the rods are supported by a number of grids that should be able to successfully accommodate such permanent deformation without damaging any fuel rods.

Fuel assembly criticality due to limited permanent deformation of fuel assembly grids is not a concern. Criticality is not sensitive to the slight geometry changes expected from grid deformation. Further, the boron plates installed as a part of each rack are expected to overcome any criticality effects of the associated geometry changes.

In response to the NRC staff's request for additional information on post earthquake rack configuration control, the licensee has committed to modify inspection procedure MYPTP-12, Engineering Evaluation Following an Earthquake at Maine Yankee, to include examination of rack-to-rack and rack-to-wall gaps. The proposed procedure modification will be completed prior to installation of the racks in the licensee's SFP.

2.3.4 Conclusion--Seismic Aspects

The NRC staff concludes that the SFP would continue to support the postulated loads under normal, severe environmental, and accident conditions while maintaining its physical integrity.

Based on its review of the licensee's past and current accident analyses of a dropped fuel assembly, as well as the satisfactory refueling experience at MY, the NRC staff concludes that the evaluation provided by the licensee for the consideration of fuel handling accidents is satisfactory and acceptable.

The staff found that the estimated safety factor for overturning a new spent fuel storage rack was greater than 5.0, thus exceeding the allowable factor of 1.1 provided in Standard Review Plan (SRP) Section 3.8.5 and is, therefore, acceptable.

Based on the review and evaluation of the licensee's submittal, additional test information and analysis provided by the licensee as well as the staff's independent assessment, it is concluded that the licensee's design of the spent fuel rerack modules and the SFP are adequate to withstand the effects of the required design basis environmental, abnormal and accident loads and able to maintain the integrity of the fuel assemblies. Therefore, the licensee's request for the use of the high-density racks, as proposed, is acceptable with regard to the seismic adequacy of the structural design.

2.4 Criticality

2.4.1 Fuel Storage Locations

Two separate storage regions are proposed in the MY SFP, with independent criteria defining the highest potential reactivity in each of the two regions. Region I has a nominal center-to-center spacing between storage cells of 10.5 in. and is designed to accommodate new fuel with a maximum enrichment of 4.5 weight percent (w/o) U-235, or spent fuel regardless of its discharge burnup. Region II has a nominal center-to-center spacing of 9.085 in. and is designed to accommodate fuel assemblies of various initial enrichments that have accumulated specified minimum burnups.

Analysis of the reactivity effects of fuel storage in Regions I and II was performed with the two-dimensional, multi-group transport theory computer code, CASMO-3. Independent verification calculations were performed with a Monte Carlo technique using the KENO-V.a computer package and the 123-group nuclide cross section library prepared by the NITAWL-S code. To minimize the statistical uncertainty of the KENO-V.a calculations, 1.2 million neutron histories in 2000 generations of 600 neutrons each were accumulated in each calculation. Experience has shown that this number of histories is sufficient to assure convergence of KENO-V.a reactivity calculations. The PDQ-7 fine mesh diffusion theory code was used to analyze abnormal configurations associated with accident conditions. SIMULATE-3, a nodal diffusion theory code, was used to determine axial burnup effects on reactivity.

The criticality analyses were performed with several assumptions that tend to maximize reactivity. These include:

(1) Unborated pool water at the temperature yielding the highest reactivity (68 °F).

(2) Minimum Boral loading of 0.020 g/cm^2.

(3) Neutron absorption effect of structural material is neglected.

(4) Infinite fuel array with no radial or axial leakage.

For the nominal storage cell design in Region I, uncertainties due to center-to-center spacing, thickness of the stainless steel, storage cell inner dimension, Boral sheath thickness, Boral width, Boral thickness, fuel enrichment, and fuel density were accounted for. These uncertainties were appropriately determined to be at least at a 95 percent probability, with a 95 percent confidence (95/95 probability/confidence) level. In addition, a calculational bias and uncertainty was determined from benchmark calculations. When fully loaded with fuel enriched to 4.5 w/o U-235, the final Region I rack design resulted in a k-eff of 0.942 when combined with all known uncertainties. This meets the NRC staff's criterion of k-eff no greater than 0.95, including all uncertainties at the 95/95 probability/confidence level and is, therefore, acceptable.

For the Region II storage cell design, the same uncertainties were considered. In addition, an allowance for uncertainty in the burnup analyses and the axial burnup distribution, as well as an adjustment to the CASMO-3 result based on KENO-V.a calculations, were included. A series of reactivity calculations was made to generate a set of enrichment-fuel assembly discharge burnup ordered pairs that all yielded an equivalent k-eff. This method for obtaining the constant reactivity curve for required burnup as a function of enrichment is the standard method used for rack reactivity evaluations and is acceptable. The licensee's proposed TS Figure 1.1-1 shows the constant k-eff contour generated for the Region II racks. From this figure, it can be seen that the reactivity of a Region II rack containing fuel at 30 GWd/MTU burnup and an initial enrichment of 4.5 w/o U-235 is equivalent to the reactivity of a Region II rack with fresh fuel (zero burnup) having an enrichment of 1.55 w/o U-235. This configuration resulted in an acceptable maximum k-eff of 0.945, including all appropriate uncertainties.

2.4.2 Fuel Storage Conditions

Most abnormal storage conditions will not result in an increase in the k-eff of the racks. However, it is possible to postulate events, such as the misloading of an assembly with a burnup and enrichment combination outside of the acceptable area of Figure 1.1-1, or dropping an assembly between the pool wall and the fuel racks, which could lead to an increase in reactivity. However, for such events, credit may be taken for the presence of approximately 1500 parts per million of soluble boron in the SFP water (refueling boron concentration), required by TS 1.1C whenever there is fuel in the SFP, because the NRC staff does not require the assumption of two unlikely, independent, concurrent events to ensure protection against a criticality accident (double contingency principle). The reduction in k-eff caused by the boron more than offsets the reactivity addition caused by credible accidents. The licensee has shown that a boron concentration of 663 ppm is the maximum concentration required to maintain k-eff less than the NRC acceptance criterion of 0.95 for the most reactive accident condition.

2.4.3 Conclusion--Reactivity Effects

The codes used to calculate the reactivity effects of fuel storage in Regions I and II, as described in Section 2.6 above, have been benchmarked against results from numerous critical experiments and have been approved and used on previous MY spent fuel rack analyses. Therefore, the NRC staff concludes that the analysis methods used are acceptable.

The staff further concludes that appropriately conservative criticality assumptions were made in the rerack amendment request submitted by the licensee.

Based on the review described Section 2.4 above, the staff finds that the criticality aspects of the proposed rerack amendment are acceptable and meet the requirements of Appendix A to 10 CFR Part 50, for the prevention of criticality during fuel storage and handling at nuclear power plants (General Design Criterion 62). The staff concludes that the licensee may safely store its fuel in Region I of the SFP, provided the fuel enrichment does not exceed 4.5 w/o U-235. The licensee may store any fuel assembly in Region II, provided the fuel meets the burnup and enrichment limits specified in proposed TS Figure 1.1-1.

2.5 Material Selection

2.5.1 Structural Material

The following structural material has been selected for fabrication of the proposed new spent fuel storage racks:

- American Society of Mechanical Engineers (ASME) Section II, SA240, Type 304 stainless steel

ASME Section II, SA240, Type 304 stainless steel is a common austenitic alloy frequently used in nuclear applications. The choice of type 304 stainless steel for fabricating the new spent fuel storage racks is reascnable. The high chromium content of this steel imparts reasonable corrosion resistance to the oxidizing effects of most electrolytes when these electrolytes are at low concentration. The steel is, however, susceptible to corrosion in acidic solutions (pH less than 7.0) containing chloride or fluoride anions. These anions can lead to pitting of the material. The corrosion effects by chloride or fluoride anions is not as pronounced in a neutral or basic medium (pH equal to or greater than 7.0).

Control of water impurities in nuclear plant SFP water is typically provided by the SFP cooling system demineralizers. The demineralizers function to keep the chemistry of the SFP water approximately the same as the reactor coolant system, which minimizes the probability of abnormal chemistry changes during refueling operations when the two systems are connected. Control of SFP water chemistry, however, also serves to reduce corrosion effects by keeping the concentration of water impurities at low levels. Therefore, pitting, induced by residual chloride or fluoride ions in the SFP, will not present a significant problem with the SA240, Type 304 stainless steel selected for use in fabricating the new spent fuel storage racks.

2.5.2 Poison Material

The following poison material has been selected as a neutron atsorber to control reactivity in the stored spent fuel assembly array:

- Boral (a patented material produced by AAR Brooks and Perkins, Inc.)

Boral is a cermet composite material made of boron carbide particles in a Type 1100 aluminum matrix. Boral panels consist of two outer sheets of type 1100 aluminum that clad a sintered plate of boron carbide in a Type 1100 aluminum matrix. The Type 1100 aluminum material imparts sufficient pitting and general corrosion resistance by forming an aluminum oxide layer on its surface when exposed to oxidizing environments. This oxide layer is stable in environments having a pH range between 4.5 and 8.5. The boron carbide particles have been shown to have good structural compatibility with the Type 1100 aluminum matrix material.

The Boral panels to be used in the licensee's proposed rack modification are manufactured in accordance with AAR Brooks and Perkins certified procedures. Production of Boral falls within the scope of the manufacturer's quality assurance program (10 CFR Part 50 Appendix B) for nuclear grade materials. The licensee intends to install the Boral sheets by inserting them between the outside wall of the storage cells and the 304 stainless steel sheath that is welded to the wall.

Inserting the Boral panels into the sheathed areas will create a snug fit. Independent studies by industry organizations and by NRC contractors have shown that Boral may react with water or moisture to generate hydrogen gas. If unable to escape, hydrogen gas may result in deformation of the rack cells by imparting additional stresses on the walls. Information Notice 83-29, "Fuel Binding Caused by Fuel Bundle Deformation," was issued to alert the industry to this concern. The licensee's submittal indicates that holes at the corners of the stainless steel sheath will create a sufficient vent path for any potential hydrogen that may be produced by a water-aluminum reaction.

The licensee has created a Boral surveillance program to characterize the performance of the Boral panels during the remaining lifetime of the plant. The licensee's Boral surveillance program calls for placing six full length Boral test panels in the SFP. The surveillance program is designed so that the exposure received by the test panels in 1 year in the SFP will correspond to 4 or 5 years of normal exposure. This is accomplished by placing freshly discharged spent fuel assemblies adjacent to the test panels during each refueling.

For reliability purposes, the Boral test panels will be identical to the Boral panels used in the new spent fuel storage racks. Six additional full length Boral test panels will be set aside in the SFP as control samples. The accelerated Boral surveillance program calls for removing and testing one full length Boral test panel at 1, 3, 5 and 10-year intervals of accelerated exposure in the SFP. Upon removal, each test panel will be analyzed according to the following tests:

. Visual examination

. Dimensional measurements

- Neutron attenuation
- Neutron radiography
- Chemical analyses
- Hardness testing

The neutron attenuation and dimensional measurements are the more important tests, because they are used to determine whether or not the coupons are exhibiting any signs of boron loss or structural deformation. Identical tests will be performed on the test panels and the results compared. This will create a reliable method of assessing the extent of deformation or degradation in Boral panels exposed to the SFP environment.

2.5.3 Conclusion--Material Selection

Type 304 (SA240) stainless steel is an acceptable material from which to fabricate the licensee's new spent fuel storage racks. Boral is an acceptable poison material. However, because Boral may generate hydrogen when in contact with water, care must be taken to provide a sufficient path to allow hydrogen to escape from its stainless steel sheath. The design of the new spent fuel storage racks proposed by the licensee provides such an escape path for hydrogen.

Upon review of the licensee's submittal, the NRC staff concludes that the structural and poison materials selected by the licensee are acceptable for construction of the new spent fuel storage racks.

Based on the considerations discussed above, the staff concludes 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 issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

2.6 Occupational Dose Control

2.6.1 Estimated Dose

The licensee estimates that the occupational dose for the proposed reracking effort will fall between 4 and 6 person-rem.

This overall occupational dose estimate is based on individual dose estimates for each of the series of anticipated activities to be performed during the reracking effort. These activities include fuel shuffling, removing and decontaminating (hydrolasing) the old, empty racks; installing new racks; and preparing the old racks for shipment.

2.6.2 Dose Control Guidelines

The licensee has indicated that removal of the old racks and installation of the new racks will be performed using remote handling tools. Diving operations are not anticipated and will only be used as a last resort. However, if divers are used, the licensee has committed to the guidance of Appendix A to Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants."

The licensee will prepare and follow detailed procedures that are consistent with principles and practices that will keep personnel exposures as low as reasonably achievable (ALARA). Work in and around the SFP will be controlled and monitored by the licensee's radiation control staff at all times, and the installation and decontamination efforts will be planned to minimize the potential for exposure to fuel fragments and highly radioactive ("hot") particles.

The licensee has indicated that radiation near the surface of the SFP, and fuel building airborne radioactivity are dependent on the type and concentration of radionuclides in the pool water. The licensee stated that it does not anticipate an increase in type or concentration of radionuclides in the pool water, and therefore expects little change in the radiation levels or airborne concentration in or around the SFP. In addition, a temporary area radiation monitoring system will be installed to monitor gamma radiation levels in the SFP area.

The licensee notes that continuous air samplers will be used whenever a potential exists for significant airborne radioactivity. In addition to the routine use of both self-reading and thermoluminescence dosimeters, extremity and multiple whole body badges will be used, as appropriate. Further, work activities are to be governed by job-specific radiation work permits (RWP) specifying appropriate radiation protection control measures. The licensee further stated that work activities and equipment movement will be monitored and controlled to minimize contamination and maintain personnel exposure ALARA.

2.6.3 Conclusion--Occupational Dose Control

Based on the NRC staff's review of the licensee's application, the proposed occupational dose control aspects of the SFP rerack effort are acceptable.

2.7 Solid Radioactive Waste

2.7.1 Old Spent Fuel Storage Racks

In its application, the licensee states that the existing fuel racks will be shipped in accordance with 10 CFR Part 71 and applicable Department of Transportation requirements to an approved processing facility for decontamination and volume reduction. Shipping will be accomplished via the use of special shipping containers and transportation. The licensee estimates the volume of radioactive waste resulting from this project will be 415 cubic ft., and will represent an approximate 97% reduction in the volume of the material in the old racks removed from the SFP.

2.7.2 SFP Cooling and Cleanup System Spent Fuel

The licensee states that about 22 cubic feet of solid radioactive waste per year is generated by the SFP cooling and purifications system. As a result of the frequent fuel shuffling and underwater cleaning (hydrolasing) of the old racks during removal, the SFP cooling and purification system may register a small increase in radioactive material trapped by its filters and demineralizers. These resins are periodically replaced (annually) and disposed of as solid radioactive waste. However, no change in this volume of solid radioactive waste is expected as a result of the reracking effort.

2.7.3 Conclusion--Solid Radioactive Waste

Based on its review, the NRC staff finds that the licensee's plans for handling and disposing of solid radioactive waste generated in connection with the planned reracking operation meet regulatory requirements and are, therefore, acceptable.

2.8 Design Basis Accidents

2.8.1 Current Licensing Basis

In its application for a license to operate Maine Yankee Atomic Power Station, the licensee evaluated the possible consequences of postulated fuel handling accidents--including means for avoidance--and provided means for mitigating the consequences should they occur. In addition, the NRC staff conservatively estimated offsite doses from the design basis fuel handling accident and concluded that plant mitigative features would reduce the offsite doses to below the doses specified in the Standard Review Plan (NUREG-0800).

2.8.2 Proposed Amendment Basis

The licensee has evaluated the effects of the proposed amendment on the calculated consequences of a spectrum of postulated design basis accidents and concludes that the effect of the proposed amendment is small, and that the calculated consequences are within regulatory requirements and staff guideline dose values. Because the licensee is now permitted to use fuel having a higher enrichment than when originally licensed, the NRC staff has reanalyzed the fuel handling design-basis accident (DBA) for this case. According to NUREG/CR-5009 (February 1989) increasing fuel enrichment to 5.0 weight percent U-235, with a maximum burnup of 60 GWd/MTU, increases the doses from a fuel handling accident by a factor of 1.2. The licensee currently is authorized to use fuel enriched to 4.5 weight percent U-235. Therefore, the 1.2 factor increase in dose displayed in Table 1 below, bounds the dose consequences of the licensee's proposal. In Table 1, the new and old DBA doses are presented and compared to the guideline doses of NUREG-0800.

TABLE 1
Radiological Consequences of Fuel Handling Design Basis Accident

	Exclusion Area Thyroid (Rem)	Low Population Zone Thyroid (Rem)
Staff Evaluation, Operating License	25	1.0
Bounding Estimates for Higher Enrichment and Fuel Burnup*	30	1.2
Guideline (NUREG-0800, Chapter 15.7.4)	75	75

* Factor of 1.2 greater than original estimate for iodine.

2.8.3 Conclusion--Fuel Handling Design Basis Accidents

The NRC staff concludes that only potential thyroid doses increase as a result of postulated fuel handling accidents with the increased enrichment and burnup. These doses remain well within the dose limits set forth in NUREG-0800 and are therefore acceptable.

3.0 STATE CONSULTATION

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

4.0 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 February 10, 1994 (59 FR 6310). Accordingly, based on this environmental assessment and finding of no significant impact, the Commission has determined that the issuance of the subject amendment will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

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

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Date: March 15, 1994