June 22, 1994

Docket No. 50-293

Mr. E. Thomas Boulette, Ph.D
Senior Vice President - Nuclear
Boston Edison Company
Pilgrim Nuclear Power Station
RFD #1 Rocky Hill Road
Plymouth, Massachusetts 02360

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Dear Mr. Boulette:

SUBJECT: ISSUANCE OF AMENDMENT NO. 155 TO FACILITY OPERATING LICENSE NO. DPR-35, PILGRIM NUCLEAR POWER STATION (TAC NO. M85898)

The Commission has issued the enclosed Amendment No. 155 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station. This amendment is in response to your application dated February 11, 1993, as supplemented December 2, 1993, January 5, February 22, March 1, April 15, and May 16, 1994.

This amendment increases the allowed fuel assembly storage cells from 2320 to 3859, and changes the maximum loads allowed to travel over the spent fuel assemblies from 1000 to 2000 lbs., and changes the limiting characteristics of assemblies to be stored in the spent fuel from a maximum $K_{INIFITY} \leq 1.35$ to a maximum $K_{INIFITY} \leq 1.32$ and a maximum lattice average uranium enrichment of $\leq 4.6\%$ by weight.

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,
Original signed by:
Ronald B. Eaton, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II

Enclosures:
1. Amendment No. 155 to License No. DPR-35
2. Safety Evaluation

Please make changes indicated.

cc w/enclosures:
See next page

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

June 22, 1994

Docket No. 50-293

Mr. E. Thomas Boulette, Ph.D
Senior Vice President - Nuclear
Boston Edison Company
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Division of Reactor Projects - I/II

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2. Safety Evaluation

cc w/enclosures:
See next page

cc:

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

BOSTON EDISON COMPANY

DOCKET NO. 50-293

PILGRIM NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 155
License No. DPR-35

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

   A. The application for amendment filed by the Boston Edison Company (the licensee) dated February 11, 1993, as supplemented December 2, 1993, January 5, February 22, March 1, April 15, and May 16, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations;

   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.

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3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 22, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 155

FACILITY OPERATING LICENSE NO. DPR-35

DOCKET NO. 50-293

Replace the following page of the Appendix A Technical Specifications with the attached page. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
207	207

5.5 FUEL STORAGE

A. The new fuel storage facility shall be such that the K_{eff} dry is less than 0.90 and flooded is less than 0.95.

B. The K_{eff} of the spent fuel storage pool shall be less than or equal to 0.95.

C. Each fuel assembly in the spent fuel pool shall have a maximum K-infinity less than or equal to 1.32 and an enrichment of 4.6% U-235 or less averaged over the axial planar zone of highest average enrichment.

D. The number of spent fuel assemblies stored in the spent fuel pool shall not exceed 3859.

E. Loads in excess of 2000 lbs. shall be prohibited from travel over fuel assemblies in the spent fuel storage pool.

F. No fuel which has decayed for less than 200 days shall be stored in racks within an arc described by the height of the cask around the periphery of the energy absorbing pad.

5.6 SEISMIC DESIGN

The station Class I structures and systems have been designed for ground accelerations of 0.08g (design earthquake) and 0.15g (maximum credible earthquake).

BASES:

5.5 FUEL STORAGE

The fuel storage assembly K-infinity in Section 5.5.C refers to the maximum K-infinity for the standard reactor core geometry. Storage of fuel assemblies meeting specification 5.5.C will result in K_{eff} less than 0.95 for both normal and abnormal storage conditions.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 155 TO FACILITY OPERATING LICENSE NO. DPR-35

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

1.0 INTRODUCTION

By letter dated February 11, 1993, as supplemented on December 2, 1993, January 5, February 22, March 1, April 15, and May 16, 1994, Boston Edison Company (BECo or licensee) requested to amend Operating License (OL) DPR-35 by changing the Technical Specifications (TSs) for Pilgrim Nuclear Power Station (PNPS). The Commission issued a proposed finding of no significant hazards consideration in the Federal Register on April 30, 1993 (58 FR 26171). On May 27, 1993, the Attorney General of the Commonwealth of Massachusetts filed comments in Opposition to Proposed Finding of No Significant Hazards Consideration, Request for Hearing and Petition to Intervene. On August 25, 1993, the Massachusetts Attorney General withdrew the Petition to Intervene and the Request for Hearing, noting that the parties involved had "resolved the matters at issue." There were no further public comments on the Commission's proposed finding. The proposed changes to the TS do the following:

- Increase the fuel-assembly storage cells from 2320 to 3859.
- Change the maximum loads allowed to travel over the fuel assemblies from 1000 lbs. to 2000 lbs.
- Change the limiting characteristics of assemblies to be stored in the spent fuel pool (SFP) from a maximum $K_{infinity} \leq 1.35$ to a maximum $K_{infinity} \leq 1.32$ and a maximum lattice average Uranium enrichment of $\leq$4.6% by weight.

The increase in the storage capacity will be accomplished by installing six additional stainless steel storage racks with Boral as the neutron absorbing material. The racks will be added in at least two steps. The first step will include the addition of two racks, N_1 and N_2, with additional storage capacities of 288 and 270, respectively, for a total of 558 locations. This first addition will permit operation of PNPS until the year 2003. Additional racks will be installed as needed to maintain the capability to offload the full core.

The licensee has designed overhead platforms to be placed only on new racks. The platforms are intended to store material which may be radioactive but not fissionable.

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2.0 Evaluation

2.1 Heavy Load Concerns

2.1.1 Weight of Heaviest Rack

The licensee notes that the heaviest rack to be moved into the SFP, rack N_1, weighs 29,400 lbs. When fully immersed in water, rack weights will be somewhat reduced, depending upon the weight of water displaced by each rack.

2.1.2 Reactor Building Crane

The reactor building crane (RBC) has both a main and auxiliary hoist. With a factor of safety of five, the main hoist, which has been designed to lift 100 tons, should be able to lift 500 tons (1,000,000 lbs.) before any part fails upon reaching its ultimate stress value. Similarly, the auxiliary hoist, with a design capacity of 5 tons, should be able to lift 25 tons (50,000 lbs.) before any part fails upon reaching its ultimate stress value. Only the main hoist will be used to transport racks, producing a minimum factor of safety of 34 (1,000,000/29,400) for this portion of the racking process. The auxiliary hoist or main hoist may be used to transport overhead platforms producing a minimum factor of safety of 27 for the auxiliary hoist; the factor of safety for the main hoist (in the event the main hoist is used) would be much larger.

2.1.3 Special Lifting Device (Lifting Rig)

The licensee has designed a lifting rig with four legs. The licensee reported that the lifting rig is capable of lifting in excess of six times the maximum rack load of 29,400 lbs. before the stress in any part reaches the yield stress and in excess of ten times the maximum load before any part stress reaches or exceeds the ultimate stress.

Prior to use each set of legs constituting a single load path will be tested by the manufacturer at 150% of the maximum rack weight. The licensee will store the lifting rig after initial use in compliance with the plant heavy load procedure, until needed. Thereafter, the licensee will conduct a nondestructive examination of the rig prior to any further use as an approach consistent with NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." This is consistent with the guidelines of ANSI 14.6-1978, "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 Kg) or More."

2.1.4 Overhead Platforms

The licensee intends to place structures (overhead platforms) on top of the new racks to store material which has been irradiated but is not fissionable. Each platform is estimated to weigh about 1850 lbs. (including its lifting device) and is intended to store a maximum load of 10,000 lbs., including the weight of the platform. A calibrated load cell will be used to assure the

10,000 lb. limitation. After reaching the 10,000 lb. limit, any platform so loaded will be covered and labelled so as to prevent further storage of material. The licensee stated that platforms would be moved only when empty.

2.1.5 Lifting Devices Not Specially Designed

The Licensee plans to use four turnbuckles, each rated to hold 10 tons and to fail at a proof load of 50 tons (100,000) lbs., as rigging for each overhead platform. Overall the turnbuckles would fail under a total load in excess of 400,000 lbs.

2.1.6 Other Heavy Loads

The licensee committed to use the RBC for any other loads on the refueling floor dealing with new racks and the overhead platforms which are in excess of 1000 lbs., with the exception of those normally carried by the refueling bridge. Further, the licensee committed to using lifting gear having minimum safety margins of 10, for such use.

2.1.7 Other Protective Measures

2.1.7.1 Procedures

The licensee states that the activity connected with the addition of new racks will be conducted in accordance with written procedures which comply with the licensee's heavy load handling procedures. The licensee will also review and approve any procedures developed by the contractor prior to use.

2.1.7.2 Overhead Platforms

The licensee intends to maintain a height of 36 inches or less over racks when moving platforms to minimize damage to the racks in the event of a platform drop. The 36-inch height will be specified in installation instructions and will be administratively controlled.

2.1.7.3 Miscellany

The licensee proposes to follow additional measures, items (i) through (v) below, to increase the safety of the overall process:

(i) The cranes used in the project receive preventative maintenance checkup and inspection per the PNPS maintenance procedures.

(ii) The cranes used will lift no more than 50% of their rated capacity at any time during the reracking operation.

(iii) Safe load paths will be developed for moving the new racks. The racks will not be carried over any region of the pool containing active fuel.

(iv) The rack upending or laying down will be carried out in an area which is not overlapping to any safety-related component.

(v) All crew members involved in the reracking operation will be given training in the use of the lifting and upending equipment.

2.1.8 Conclusion on Heavy Loads Concerns

The licensee plans to use the RBC to move racks and overhead platforms, in accordance with applicable guidelines. These guidelines specify that, in the lack of a single-failure-proof (dual) hoist mechanism, for cranes already in place, the use of a total handling capability at least 10 times the load being handled suffices to assure that the potential for a load drop is extremely small. Thus, the use of the RBC crane for the movement of racks and platforms is found to be acceptable. In addition, the licensee has committed to use the RBC to carry any other loads over 1000 lbs. which involve movement of racks, platforms, or platform loads over the SFP (except for those loads normally handled by the refueling bridge) and to use lifting gear having minimum safety margins of 10. Thus, the handling of any heavy loads between 1000 and 2000 lbs. (the maximum load the licensee proposes to allow) over fuel assemblies in the SFP is also found acceptable.

The special lifting device complies with the guidelines of ANSI N14.6-1978, for single-failure-proof special lifting devices and, thus, is found to be acceptable for use in the process of adding racks.

The licensee proposes to use other measures (see section 2.1.7, above) to improve the safety of the handling of heavy loads. With the other details discussed above, the NRC staff considers that all concerns related to the handling of heavy loads have been resolved.

2.2 Thermal/Hydraulic Concerns

2.2.1 Fuel Pool Cooling and Cleanup System (FPCC)

The FPCC consists of two trains, each of which contains a pump and heat exchanger (Hx). There is also a filter, filter backwash system and a demineralizer in the FPCC. The Hx is cooled by water from the reactor building closed cooling system. The FPCC is rated for a combined heat load of 6.3E06 BTU/HR with a SFP coolant temperature of 125 degrees F. When the plant is in the refueling mode, the FPCC may be augmented by use of the RHR system in the event additional cooling is required.

2.2.2 Decay Heat Calculation

The licensee calculated the decay heat for two cases: (1) Case 1-a normal refueling offload of 164 spent fuel assemblies, and (2) Case 2-a full core offload of 580 spent fuel assemblies. Both cases were assumed to occur after operating cycle 20, in the year 2015, with the SFP completely filled with 3589 fuel assemblies. The decay heat load for the 19 previous offloads was calculated to be 3.31E06 BTU/HR. For Case 1, the total decay heat load was calculated to be 8.61E06 BTU/HR; for Case 2 the heat load was 25.9E06 BTU/HR. The licensee calculated these heat loads at a time coincident with the maximum coolant temperature, in each case.

2.2.3 SFP Coolant Temperatures

The licensee calculated the SFP coolant temperatures for the two cases discussed previously. For Case 1, the licensee assumed that both FPCC trains were in operation; the maximum bulk coolant temperature calculated for that case was 142 degrees F. In Case 2, i.e., full core offload, the licensee specified the use of the RHR system without the FPCC system in operation; the licensee stated that this case would provide the minimum cooling with RHR augmentation. The SFP bulk coolant temperature calculated for that case was 129 degrees F.

2.2 Fuel Pin Cladding Temperature

In the model designed by the licensee to calculate the maximum fuel pin cladding temperature, a circumscribing circle was drawn around the spent fuel racks and the empty areas were assumed to be filled with additional spent fuel assemblies. This was done in order to eliminate the irregularity obtained in using the actual shape of the SFP and racks. Downward flow was assumed to occur between the fuel assemblies and the SFP wall with the minimum gap between pool and wall assumed in the calculation. The decay heat generation for the spent fuel used in the calculation was assumed to be from the latest batch, discharged simultaneously so as to provide equal heat generation for all the assemblies. The calculation also assumed that a misplaced fuel assembly blocked 50% of the top opening for the thermally limiting cell. The presence of the overhead platform as a resistance to flow was examined; it was determined that the 50% blockage exceeded the effect caused by the overhead platform. The cladding temperatures resulting from these calculations were found to be 229.2 degrees F for the unblocked case with a maximum local pool water temperature of 184.4 degrees F; 236.3 degrees for the blocked case with a local pool water temperature of 193.3 degrees F. The licensee concluded that nucleate boiling did not occur and, thus, reactivity calculations were not affected because of the absence of voids which could have occurred as a resulting of nucleate boiling.

2.2.5 Loss of Cooling

2.2.5.1 Makeup Water for the SFP

The licensee reported that there were four methods of providing water to maintain the SFP water level in the event the FPCC system failed. These are shown in Table 1, below.

Table 1
Methods to Provide Water for SFP

1. Condensate Transfer System
   a. 3-inch piping - maximum of 200 gpm with one pump
   b. 10-inch piping - approximately 2000 gpm with one pump

2. Demineralized Water Transfer System - 110 gpm with one pump

3. Fire Protection System - 150 gpm from one hose station

None of these systems are designed to seismic Category I; thus they may not be available after a seismic event. Only the RHR system, which may be used to transfer water through the RHR/FPCC intertie is seismic. The intertie system, however, may only be used when the plant is at the cold shutdown condition.

For the condensate and demineralized water transfer systems there are three alternate storage tanks, four pumps and three separate flow paths to the FPCC. The diesel fire pump would be available in the event of loss of offsite power (LOOP). The RHR system would also be available to supply water to the SFP via the FPCC in the event of LOOP, but only after the plant is established in the shutdown cooling mode.

2.2.5.2 Time to Boil

The licensee calculated the time to reach boiling (212°F) for both Case 1 (normal refueling offload) and Case 2 (full core offload), assuming that the FPCC cooling system failed at the point at which the maximum bulk coolant temperature was reached. The time to boil and the maximum boil-off rate is shown below (Table 2):

Table 2

Time to Boil

Case Number	Time to Boil (Hours)	Maximum Boil-Off Rate(gpm)
1	16.17	18.33
2	6.14	54.69

2.2.6 Conclusions on Thermal/Hydraulic Concerns

The staff checked the decay heat values determined by the licensee and found them to be conservative. The calculated SFP coolant temperature during refueling and thereafter were consistent with previously approved values, including those resulting from the use of the RHR system. The calculated fuel pin cladding temperatures are much lower than those encountered during normal operations and, thus, are acceptable. The lack of voids is consistent with reactivity calculations.

The loss of cooling and minimum time to boil for a full core offload, 6.4 hours, provides enough time to enable plant operations to introduce alternative cooling methods or to supply water to replace that which would be lost as a result of boiling. There are enough methods to replace such loss to assure maintenance of the SFP water level. The staff considers the foregoing to be acceptable and, thus, the thermal/hydraulic concerns are found to be resolved.

2.3 EVALUATION OF THE CRITICALITY ASPECTS

2.3.1 Installation of Additional Storage

The TS change proposed by the licensee is required to support the installation of six additional spent fuel storage racks in the space available in the PNPS SFP. This will increase the cumulative spent fuel storage capacity of the pool to 3859 cells and will extend the full core reserve capacity to the end of licensed life (year 2012).

Although the proposed TS change pertains to an additional six storage racks containing a total of 1526 cells, only two racks will be installed in the pool at the present time, limiting the immediate increase in the storage capacity to 558 cells. This interim increase will enable BECo to operate PNPS with a full-core off-load reserve until Cycle 13 (now calculated to occur in 2003). Additional spent fuel storage racks will be installed in the SFP as necessary.

The safety assessment of the proposed rack modules demonstrates their thermal-hydraulic, criticality, and structural compliance with established requirements. Thermal-hydraulic compliance requires that fuel cladding not fail because of excessive thermal stress and that the steady-state pool bulk temperature remain within the limits prescribed for the SFP.

The HOLTEC report submitted by the licensee provides the basis and safety analysis performed to demonstrate that the new spent fuel storage racks and the existing racks meet applicable industry codes and standards. The report also contains information requested by the NRC in "Office of Technologies (OT) Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," and a 1979 Addendum thereto. The staff has reviewed the changes requested by the licensee and finds them acceptable.

2.2 Reactivity Analysis

The analysis of the reactivity effects of the fuel storage racks was performed with both the CASMO-3 computer code (a two-dimensional multigroup transport theory code) and the KENO-5a code (a Monte Carlo code), using the 27 energy group SCALE neutron cross section library. CASMO-3 was also used to evaluate small reactivity increments associated with manufacturing tolerances. These codes are widely used for the analysis of fuel storage rack reactivity and have been benchmarked against results from numerous critical experiments. The staff concludes that the analysis methods used are acceptable.

To ensure that true reactivity will always be less than the calculated reactivity, the following assumptions which tend to maximize the rack reactivity were made:

(1) The racks contain the most reactive fuel authorized to be stored in the facility without any controls or any uncontained burnable poison, and with the fuel at the burnup corresponding to the highest reactivity during its burnup history.

(2) Moderator is pure, unborated water at a temperature within the design-basis range corresponding to the highest reactivity.

(3) Criticality safety analyses are based on the infinite multiplication factor (K-infinity); that is, lattice of storage racks is infinite in all directions.

(4) Neutron absorption effects of structural material is neglected.

The staff concludes that appropriately conservative assumptions were made.

For the design-basis reactivity calculations, uncertainties due to tolerances in the following were accounted for: boron loading, Boral, cell lattice spacing, stainless steel thickness, and fuel enrichment and density. These uncertainties were appropriately determined at least at the 95 percent probability, 95 percent confidence (95/95 probability/confidence) level. In addition, a calculational bias and uncertainty were determined from benchmark calculations. The final design when fully loaded with fuel enriched to 4.6% wt U-235 resulted in a calculated K-infinity of 0.922 when combined with all known uncertainties. This value meets the NRC criterion of 0.95 including all uncertainties at the 95/95 probability/confidence level and is, therefore acceptable.

The new spent fuel storage racks will store fuel in discrete modules in the SFP; one module will be available for installation in the cask pit. The maximum K-infinity limit of each of the boiling water reactor (BWR) fuel assemblies stored in the SFP will be changed from 1.35 to 1.32. Although the HOLTEC report discusses K-infinity for U-235 enrichment up to 4.9% wt., the proposed TS limits the U-235 enrichment to 4.6% wt. (for standard 8x8 and 9x9 geometry) to maintain additional conservatism. The standard core geometry is defined as an infinite array of fuel assemblies on a 6-inch lattice spacing at 20° C without any control absorber or voids.

The reactivity effects during abnormal and accident conditions due to the effects of temperature and water density, abnormal location of a fuel assembly, eccentric fuel assembly positioning, fuel rack lateral movement, or the dropping of a fuel assembly on top of the storage rack were considered. None of the credible conditions resulted in exceeding the limiting reactivity criterion of K-effective no greater than 0.95.

The following TS changes have been proposed as a result of the requested SFP reracking. The staff finds these changes acceptable.

(1) TS 5.5.C will be revised to change the K-infinity factor from 1.35 to 1.32 for U-235 enrichment up to 4.6% wt averaged over the axial planar zone of highest average enrichment.

(2) TS 5.5.D will be revised to increase the storage capacity of the SFP from 2333 to 3859 storage cells.

2.3.3 CONCLUSION

The NRC staff finds that the criticality aspects of the proposed modifications to the Pilgrim Unit 1 TS and the requirement for additional storage racks are acceptable and meet General Design Criterion 62 of Appendix A to Part 50 of Title 10 of the Code of Federal Regulations regarding the prevention of criticality in fuel storage and handling.

2.4 STRUCTURAL ASPECTS

2.4.1 Background

At present, ten free-standing racks are located in the SFP as shown in Figure 1.1 of the application (attached). The existing rack configuration was approved by Amendment 91 to Facility Operating License. This proposal consists of installation of six additional racks; two racks will be installed immediately after the issuance of the amendment approving the proposed modifications (Campaign I), and the remaining four racks will be installed after year 2000 (Campaign II) as shown in figures 2.1 and 2.2 (attached).

All applicable loads and load combinations, as provided for in Appendix D of Standard Review Plan (SRP) 3.8.4 have been considered by the licensee for demonstrating the acceptability of the racks. As the (existing and proposed) racks are free standing, the seismic load is important. The following discussion addresses the acceptability of the racks under seismic loads.

2.4.2 Seismic Input

The design response spectra or the safe shutdown earthquake (SSE) for the site are specified by Housner spectra anchored at 0.15g. The licensee has developed the floor response spectra at the pool floor elevation (El.- 74 ft. 3 in.) for 1% damping (to be used with the Operating Basis Earthquake - (OBE)) and for 2% damping (to be used with the SSE). The staff finds the use of these spectra in the seismic analyses of racks to be acceptable. The rack vendor, Holtec International, Inc. (HI), utilized these spectra for generating four sets of time-histories and demonstrated that the average spectra generated from the four sets of time-histories envelop the respective floor spectra. HI also confirmed that any two time-histories within each set are statistically independent, i.e., their normalized correlation coefficients are less than 0.15. Furthermore, HI determined a set of time-histories that produced the maximum stresses and displacements at the critical locations in a typical rack. Recognizing that a single set cannot produce the maximum responses in all cases, and to ensure that the analysis results from this set would always bound the results from any other set, HI applied a multiplication factor of 1.15 to this set of time histories, and called that the controlling set. HI used the controlling set for various analyses of racks. The staff finds the procedure used by the HI in developing the seismic input for the rack analyses to be in accordance with the provisions of SRP 3.7.1 and therefore, acceptable.

2.4.2 Rack Modelling and Analyses

Rack Structure: The proposed racks are designed and constructed as free-standing and self-supporting. The principal construction materials are ASME SA240-Type 304L stainless steel sheet and plate stock for cell-box structure, and SA564 for the adjustable support spindles. The neutron-absorbing material "Boral" in sheet form is located between the adjacent fuel cells. From the structural analysis standpoint, the existing racks have similar characteristics. A continuous common baseplate with chamfered holes defines the lower portion of cellular region in both sets of racks.

Rack Modelling: The rack structure (including the fuel assemblies) is modelled as a 3-D, 22 degrees-of-freedom (DOF) system. The movement of the rack at any height is described by six DOF at the rack base and six DOF at the rack top. The fuel assemblies are modelled as five lumped masses located at four equally spaced intervals along the height. Each mass has two DOF allowing the masses to displace in two horizontal directions. The vertical motion of the fuel assembly mass is considered to be equal to the rack vertical motion at the base plate, as no relative separation of the fuel assembly would be expected at the base plate level. The centroid of each fuel assembly mass can be located off center, relative to the rack structure centroid at that level, to simulate a partially loaded rack. All fuel assemblies are assumed to move in-phase within the rack to yield conservative estimates of forces and displacements. Fluid coupling, between rack cells and fuel assemblies, and between racks and walls, is simulated by appropriate inertial coupling in the system.

Rack Analysis Parameters: Because of the multiple non-linearities involved, and not knowing unique parameters that could give the most adverse (i.e., maximum) forces and displacements, HI utilized extreme parametric values in the analyses of all single racks. For example, the racks adjacent to a rack being analyzed were considered 'in-Phase' to simulate lowest fluid resistance, and 'out-of-Phase' to simulate minimum gap between the racks. The friction coefficients at the floor-pedestal interface were considered as 0.2 and 0.8 (the extreme values found in the experiments) for each analysis. Three fuel loading cases, i.e., fully loaded, empty and half loaded, (with maximum eccentricity with the centroid of the rack structure) were considered for each simulation.

Rack Analysis: A proprietary computer program "DYNARACK" is used for the analysis of racks subjected to the controlling set of time-histories. The program has been validated partly against exact solutions, limited experimental data, and solutions obtained using alternate numerical schemes. The time step for dynamic simulations is established by ensuring convergence and stability of results. Results of the analyses are time-history responses of all DOF of the particular model, and of all forces and moments at important sections of the structure.

The above procedures and DYNARACK computer program have been used in a number of prior rerack applications for the analysis of rack structures. The staff has accepted these procedures in the safety evaluations of the reracking

applications. For PNPS proposal, the staff finds the use of above procedures and DYNARACK computer program to be acceptable for estimating the stresses and displacement of the racks.

The largest uncertainty in predicting the behavior of racks by a 3-D single rack model is in the simulation of fluid coupling effects. To simulate the effects of far field (compared to near field in the single rack simulation) fluid coupling, in recent years, HI has extended the use of the DYNARACK program for the analysis of all racks in a pool in a single simulation, and termed it as whole pool multi-rack (WPMR) 3-D analysis. In order to reduce the number of DOF to be incorporated in the WPMR analyses, HI utilized DOF reduction procedure on the selected single rack 3-D models such that the kinematic responses calculated by the 8 DOF model are in agreement with the results of 22 DOF 3-D rack models. Also, the friction coefficient between the rack pedestals and the pool is ascribed to each rack in the WPMR model by a random number generator with Gaussian normal distribution characteristics, and varying between the bounding values of 0.2 and 0.8 and a mean of 0.5.

2.4.4 Evaluation of Rack Analyses Results

The results of the 3-D single rack analyses indicate that the maximum displacement could be as high as 0.39 in. at a top corner of rack E9 (see fig. 2.1). The minimum gap in the rack configuration is 1.88 in. Considering the unlikely case, when the adjacent rack could go through a similar displacement, the minimum gap between the racks or between a rack and any wall will not be compromised, and no rack impact with the wall or another rack would be expected. The results of the WPMR analysis indicate that the maximum displacement at the top of E9 rack could be as high as 0.84 in. However, the results of the impact loads at the top of the racks show no impact between the racks, or between any rack and the walls. Maintenance of specified gaps between the racks, and the racks and the walls, as indicated in Figures 2.1 and 2.2, is desirable from a structural point of view. Therefore, as per staff recommendation, the licensee committed to ensure these gaps during the installation of the racks and inspect them subsequent to an earthquake exceeding the plant's OBE.

Overturning of any of the racks is not possible even when the outer racks (racks having no physical barrier against overturning) in Campaign I are partially loaded to simulate maximum eccentricity from the centroid of the racks.

The stress factors (i.e., ratios of actual stresses to allowable stresses) for pedestal-components under worst-loading conditions are found to be less than 0.3 (allowable for SSE load would be 2.0). The pressure and impact load capacities of rack cell-plates are at least twice the corresponding imposed loads.

2.4.5 Evaluation of Spent Fuel Pool

The SFP is a Seismic Category I reinforced concrete structure with steel framing to support the floor slab. HI analyzed the structure using the finite element program-ANSYS. All applicable loads are considered in the analysis.

The results of individual loads are combined using ACI 349-85 Code. In the original submittal, HI did not consider the exceptions to the Code taken by the staff in Regulatory Guide 1.142. In response to the staff request for additional information, the licensee provided information which showed that the original calculations will be only slightly (< 3%) affected by the applicable exception.

For the SFP analysis, HI utilized the pedestal loads on the floor and the hydrodynamic pressures on the walls as obtained from the WPMR analyses. HI also incorporated lower stiffnesses of the slab and the walls to simulate existing cracking, where indicated. The results indicated the minimum margin of safety in moment against reaching the code established limit state to be 1.4 under load combination involving the SSE load, and 1.25 under load combination involving accidental temperature (i.e., boiling). The margin of safety in shear was conservatively calculated as 1.33 under load combination involving the SSE load. The staff finds the approach used and the results thereof acceptable.

2.4.6 Evaluation of Accidental and Platform Loads

Drop of heavy fuel assembly and rack platform: Though the proposal does not stipulate the use of consolidated fuel assemblies, for calculating the effects of such a drop, HI has used the weight of a fuel assembly (FA) as 1500 lbs. (1360 lbs. - weight of the consolidated fuel bundle and 140 lbs for miscellaneous handling devices). The calculations indicated that the maximum height of the plastically deformed rack will not exceed 1.6 in. when a FA accidentally falls on the top of the rack from a height of 36 in. The top of the active fuel and the poison frame will be more than 10 inches beneath the top of the rack. HI also postulated such a drop on the rack base plate, and concluded that the conservatively estimated deformation of the base plate could be as high as 2.6 in., demonstrating that the pool liner will not be affected by such accidental drops.

During Campaign I, when a portion of the pool will be without any racks, if such an accidental drop occurs, the handled FA could get damaged. However, the radiological consequences of the drop are enveloped by the analysis in Section 9 of the proposal. The pool liner would also undergo localized deformation. Such an event will not be of immediate safety significance. However, the licensee will be required to report it under Paragraph 50.72 of 10 CFR Part 50.

The proposed new racks will be equipped with the provisions to install overhead storage platforms. The maximum weight of such a platform has been stated as 2000 lbs. The platform will cover major portions of the rack. As the weight of a dropped travelling platform will be distributed to the end plates of the four platform legs, the consequences of an accidental drop of such a platform, from a height of less than 36 in. from the top of the rack, are bounded by the heavy fuel drop analyses discussed above.

The platforms will be installed after the new proposed racks are completely filled with spent fuel assemblies. The platforms may carry as much as 10,000 lbs. of miscellaneous items. The licensee will have a plant procedure

to ensure that the loads are symmetrically distributed on the platforms. The staff evaluation of the licensee's analysis related to the effects of the platform loading on the rack analysis indicate that the platform loading will have very little effect on the behavior of the racks.

The staff finds that the licensee's analysis related to the accidental drop of a heavy fuel assembly and a storage platform acceptable.

2.4.7 Conclusion on Structural Aspects

Based on the evaluation of the licensee's submittal, the supplementary information provided by the licensee, and the information audited by the staff at the site, the staff concludes that the structural analysis and design of the proposed reracking (including the existing and new proposed racks, SFP and accidental drop of a fuel assembly) are in compliance with the staff position in Appendix D of Standard Review Plan 3.8.4 and the current licensing practice and are, therefore, acceptable. The staff also evaluated the consequences of an accidental drop of the proposed utility platforms (max. wt. 2000 lbs) on the structural integrity of racks and fuel assemblies and concluded that occasional travel of a platform over spent fuel assemblies is acceptable.

Recognizing the importance of maintaining the specified gaps in the rack configurations, as per staff recommendation, the licensee has committed to ensure these gaps during installation of the racks, and inspect them after an earthquake event exceeding the plant's OBE.

2.5 STRUCTURAL AND POISON MATERIALS

2.5.1 Structural Materials

The licensee has hired an outside contractor to perform a safety analysis for the proposed license amendment. The contractor has selected the following structural materials for use in the proposed storage rack modification:

- American Society of Mechanical Engineers (ASME) Section II SA240-304 stainless steel for fabrication of the racks
- ASME SA240-304 for the internally threaded support legs
- ASME SA564-630 for the externally threaded support spindle - this is a precipitation hardened stainless steel, heat treated to 1100 °F
- Weld material - type R308L stainless steel conforming to ASME specification SFA 5.9

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 fabrication of the rack assembly legs is reasonable. The high chromium content imparts reasonable corrosion resistance to oxidizing effects

of most electrolytes when at low concentration levels. The steel is, however, susceptible to corrosion in acidic solutions (pH < 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 basic media (pH > 7.0).

The licensee has opted to use a Type 630 martensitic, precipitation hardened, stainless steel for the externally threaded support spindle. Type 630 stainless steels have increased strength, without suffering considerable loss of ductility. The corrosion resistance, however, is not quite as good as that of austenitic stainless steels. The Type 630 stainless steel has been heat treated at 1100 °F to increase its resistance to stress corrosion cracking. It should be noted that control of water impurities in nuclear plant SFP water is typically provided by the SFP demineralizers in the spent fuel cooling system. The demineralizers function to keep the chemistry of the SFP water approximately the same as that of the reactor coolant system, in order to minimize the probability of abnormal chemistry incursions during refueling operations when the two systems link together. Control of SFP chemistry, however, also serves to reduce corrosion effects by keeping the concentrations of water impurities at low levels. Therefore, stress corrosion cracking or pitting, induced by residual chloride or fluoride ions in the fuel pool, should not be a problem with the SA240-304 stainless steel.

2.5.2 Poison Material

- Boral - patented material produced by AAR Brooks and Perkins

The Boral panels used in the proposed rack modifications 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 50 Appendix B) for nuclear grade materials. The licensee intends to install the Boral sheets by freely inserting them between the 304 stainless steel walls of the rack assemblies and the 304 stainless steel sheaths which are to be welded to the wall.

It is evident that the insertion of the Boral panels into the sheathed areas will create a tight fit. Independent studies by industry organizations and by NRC contractors have shown that Boral may react with water or moisture to generate hydrogen gas. Production of hydrogen 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 sheath areas will create a sufficient vent path for any potential hydrogen which may be produced by a water-aluminum reaction.

The licensee has also created an accelerated Boral Surveillance Program to characterize the performance of the Boral panels during the remaining lifetime of the plant. This program is in accordance with the NRC letter of April 14, 1978 to all nuclear power licensees, which stated that "Methods for verification of long-term material stability and mechanical integrity of special poison materials utilized for neutron absorption should include actual tests."

The licensee's accelerated Boral Surveillance Program calls for placing eight Boral test coupons (mounted on a "tree") in the SFP rack area. At the end of the first five operating cycles following the modification, the coupon tree will be surrounded with freshly discharged fuel assemblies. This is done to assure that the coupons experience a higher radiation dose than the Boral panels in the storage racks. For reliability purposes, the Boral test coupons will be mounted in a steel jacket to simulate the actual inservice geometry, physical mounting, materials, and flow conditions experienced by Boral panels used in rack design. Two Boral test coupons will be set aside as control samples.

The accelerated Boral Surveillance Program calls for removing and testing one full length Boral test coupon at the following refueling outages after the rack modification is complete: 1st, 2nd, 3rd, 5th, and 8th. The remaining three panels will be tested at periods of 5 yr, 10 yr, and 16 yr following the 8th refueling outage. Each test panel, upon its removal, will be analyzed according to the following tests:

- Visual Observation and Photography
- Neutron Attenuation
- Dimensional Measurements (length, width, and thickness)
- Weight and Specific Gravity Analyses
- Wet Chemical Analysis (Optional)

The neutron attenuation and the dimensional measurements are the more important tests of the group since they are used to determine whether or not the coupons are exhibiting any signs of boron loss or structural deformation, respectively. The licensee's contractor has established an acceptable set of screening criteria for evaluating the Boral test coupons. The results of testing on the Boral test coupons will be compared to identical tests run on the Boral control coupons.

2.5.3 Conclusion

The BECo license amendment request submittal indicates that material selection for the PNPS spent fuel rack modification has been satisfactorily thought out. The rack is to be constructed from a Type 304 stainless steel fabricated according to an approved ASME Section II specification. Boral is an acceptable poison material; however, since the Boral may generate hydrogen when in contact with water or moisture, care must be taken to provide a sufficient path to allow potential hydrogen generation to vent from the sheath area. The Boral Surveillance Program will provide a reliable method of assessing the potential deformation or degradation of Boral panels which are exposed to radiation in the spent fuel area over time. Following the review of the licensee's submittal, the staff concludes that the licensee's selection of structural, welding and poison materials meets current industry and regulatory standards and that these materials are acceptable for construction of the new rack modules.

2.6 RADIATION PROTECTION ASPECTS

2.6.1 OCCUPATIONAL DOSE CONTROL

2.6.1.a SPENT FUEL STORAGE RACKS AND STORAGE PLATFORMS

The licensee estimates in its February 11, 1993, application that the total occupational dose for the planned SFP fuel rack augmentation activities will be between 2 and 4 person-rem, including any necessary diving operations.

This overall estimate is based on individual dose estimates for each of the series of anticipated activities to be performed during the rack augmentation operations. These activities include cleaning and vacuuming the pool, removing underwater appurtenances, fuel movements, and the installation of new racks topped with storage platforms.

The licensee has indicated that the removal of underwater appurtenances may require diving operations. If so, careful monitoring and adherence to procedures should assure that the radiation dose to the divers is ALARA. Further, if divers are used, the licensee has committed to the guidance provided in Appendix A, "Procedures for Diving Operations in High and Very High Radiation Areas" to Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants."

The licensee notes that detailed procedures prepared with consideration of the ALARA principle will be utilized. In addition, the BECo states in its submittal that continuous air samplers will be utilized where a potential for significant airborne activity exists and that personnel will wear protective clothing and, as appropriate, respiratory protective equipment. Further, work activities are to be governed by radiation work permits specifying appropriate radiation protection measures. In addition to the routine use of self-reading dosimeters and thermoluminescent dosimeters, extremity badges and alarm dosimeters will be utilized as appropriate. The licensee further states that work activities, personnel traffic, and equipment movement will be monitored and controlled such that contamination will be minimized and personnel exposures maintained ALARA. Based on our review of the licensee's application, the staff finds the proposed radiation protection aspects of the SFP fuel rack augmentation acceptable.

The licensee estimates the accumulation of no more than 4,500 cobalt-60 equivalent curies of materials in the storage boxes to be placed above the fuel racks and has confirmed that the stored materials will not extend above the level of the top of the storage box sidewalls. This amount of radioactivity will increase the pool area dose rates less than 1 mr/h and should not significantly increase plant occupational dose. The licensee has described its procedures, including inventory controls, for controlling the types and amounts of radioactive materials to be stored in the storage boxes.

2.6.1.b ABNORMAL OCCURRENCES

The radioactive material to be stored in the storage boxes on the storage platforms above the new fuel racks constitutes an additional, potential source of radiation exposure in the event of abnormal operating conditions. The most serious, credible occurrence is the drain-down of the SFP to the lowest level at which conduits penetrate the pool. Such penetrations could, under abnormal circumstances, constitute a drain path. The lowest such penetration is that of the reactor cavity sparger line. A break in the non-safety-related reactor cavity sparger piping could result in water from the SFP draining through a safety-related check valve at a maximum rate of 3 gpm. At this rate, it would require several days to lower the water level 1 foot. A 5" drop in water level would result in both local and control room alarms followed by mitigating actions, including the closing of a manually operated, safety-related block valve to stop the water flow from the SFP. Since normal pool water level is 16 feet above the sparger line penetration and a 1 foot drop in water level would require 5.8 days at a drain rate of 3 gpm, 3 months would be required for the water level to reach the level of the sparger line penetration. In the unlikely event of drain-down to the level of the fuel pool sparger line penetration, a minimum depth of 2 feet, 9 inches of water shielding would remain above the storage boxes. The licensee estimates the resulting dose rate on the refueling bridge above the pool due to the presence of 4,500 cobalt-60 equivalent curies in the storage boxes as 6 rem/h. Confirmatory calculations indicate that this is a conservative estimate. The dose rate on the refueling bridge from a maximum of 55 activated control rod blades hanging on the fuel pool curbs is estimated to be 75 rem/h and the stored spent fuel would contribute 200 mrem/h. A radiation level of 81 rem/h would make access to the refueling bridge difficult from a radiation safety standpoint; however, radiation levels in areas surrounding the pool would be significantly less, and access could be accomplished if found to be necessary for mitigation purposes. The licensee has confirmed that its written procedures include actions necessary to the mitigation of an event of this nature and that radiation levels associated with a pool drain-down of this extent would not compromise plant system functions.

2.6.2 SOLID RADIOACTIVE WASTE

The licensee stated in its application that the existing spent fuel storage racks will remain in place thereby precluding the need for their cleaning, packaging, and disposal as may be associated with a fuel reracking. According to the licensee, a small amount of additional water cleanup resins may be generated on a one-time basis during the pool rack augmentation operations. Such resins would be handled in accordance with the plant's normal waste handling procedures.

Based on our review, the staff finds that the licensee's plan for handling and disposing of solid radioactive waste generated in connection with the planned rack augmentation operation meets regulatory requirements and is, therefore, acceptable.

2.6.3 DESIGN BASIS ACCIDENTS

In its application, the licensee evaluated the possible consequences of postulated accidents, including means for avoiding them in the design and operation of the facility, and recommended means for mitigating their consequences should they occur. The licensee has evaluated the effect of the changes on the calculated consequences of a spectrum of postulated design-basis accidents (DBA) (i.e, fuel handling accidents) and concludes that the effect of the proposed TS change is small, and that the calculated consequences are within regulatory requirements and staff guidelines on dose values. Since the licensee proposes to utilize higher enrichment fuel, the staff reevaluated the fuel handling accident for the Pilgrim Plant to consider the effects of higher fuel burnup.

In its evaluation for the Pilgrim Plant, issued on July 16, 1986, the staff conservatively estimated offsite doses due to radionuclides released to the atmosphere from a fuel handling accident. The staff concluded that the plant mitigative features would reduce the doses for this DBA to below the doses specified in SRP Section 15.7.4.

Although the licensee did not address a specific higher fuel burnup value in its February 11, 1993, application (relative to that currently authorized), the staff evaluated the consequences of operation at a bounding value (60,000 MWD/T) because of the licensee's reference to the use of higher enriched fuel (up to 4.9 weight percent U-235). In Table 1, fuel handling accident doses associated with extended fuel burnup, as well as that contained in the current licensing basis, are presented and compared to the guideline doses in SRP Section 15.7.4 (established on the basis of 10 CFR Part 100).

TABLE 1
Radiological Consequences of Fuel Handling Design Basis Accident (rem)

	Exclusion Area	Low Population Zone
	Thyroid	Thyroid
Staff Evaluation July 16 1986	20	2
Bounding Estimates for Extended Burnup Fuel[1] (60,000 MWD/T)	24	2.4
Staff Acceptance Criteria (NUREG-0800, Section 15.7.4)	75	75

The staff concludes that the only potential increased radiological consequences resulting from fuel handling accidents associated with extended burnup fuel are the thyroid doses; these doses remains well within the acceptance criteria given in NUREG-0800 and are, therefore, acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Massachusetts 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 have been prepared and published in the Federal Register on June 21, 1994 (59 FR 32027). Accordingly, based upon environmental assessment, the Commission has determined that the issuance of the amendment will not have a significant effect on the quality of the human environment.

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[1]According to NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors, " PNL, 1987, increasing fuel enrichment to 5.0 weight percent U-235 with a maximum burnup of 60,000 MWD/T increases the doses for fuel handling accident by a factor of 1.2.

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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