BEFORE THE UNITED STATES

NUCLEAR REGULATORY COMMISSION

In the Matter of POWER AUTHORITY OF THE STATE OF NEW YORK and CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.	)	DOCKET NO. 50-286

AMENDMENT NO. 1 to

APPLICATION FOR AMENDMENT TO

OPERATING LICENSE

On September 2, 1977 the Power Authority of the State of New York ("Power Authority"), as sole owner of the Indian Point 3 Nuclear Power Plant and co-holder of Facility Operating License No. DPR-64 (for which Consolidated Edison presently has operating responsibility as co-licensee) filed an application for amendment to Operating License No. DPR-64 for the purpose of amending portions of Technical Specification 3.8 set forth in Appendix A to that license and requesting the Commission to review a proposed modification to the Indian Point 3 Nuclear Power Plant.

The Power Authority hereby amends Attachment A of the above-mentioned application to include additional information as requested by the NRC Staff. This additional information is presented in Attachments I and II to this application, which are incorporated herein by reference.

POWER AUTHORITY OF THE
STATE OF NEW YORK

By George T. Berry
George T. Berry
General Manager and
Chief Engineer

Subscribed and sworn to before
me this 17 day of October, 1977

Helen J. McCormick
Notary Public

HELEN J. McCORMICK
Notary Public, State of New York
No. 01MC 2607500
Qualified in Kings County
Term Expires March 30, 1979

BEFORE THE UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the Matter of POWER AUTHORITY OF THE STATE OF NEW YORK and CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.	Docket No. 50-286

CERTIFICATE OF SERVICE

I certify that I have, this 21st day of November, 1977, served the foregoing document entitled "Amendment No. I to Application for Amendment to Operating License," together with Attachments, dated October 17, 1977 by mailing copies thereof, first class postage prepaid and properly addressed to the following persons:

Hon. George V. Begany
Mayor, Village of Buchanan
188 Westchester Avenue
Buchanan, New York 10511

Hendrick Hudson Free Library
31 Albany Post Road
Montrose, New York 10548

M. Reamy Ancarrow
M. Reamy Ancarrow

ATTACHMENT I

RESPONSES TO SEPTEMBER 1, 1977 QUESTIONS BY NRC

Power Authority of the State of New York

I.1. In your July 13, 1977, submittal in regard to increasing the capacity for spent fuel storage, you state that you will perform the criticality calculations assuming a uranium-235 enrichment of 3.5 percent. Please convert this enrichment number to the number of grams of uranium-235 per axial centimeter of fuel assembly that was used in the criticality calculations. We will require this number of grams of uranium-235 per axial centimeter of assembly to be specified as a Technical Specification limit on fuel assemblies that are to be placed in these high density storage racks.

Table 2.2.1 of the September 1, 1977 detailed submittal states that the maximum U-235 assembly loading will be 44.4557 gms U-235 per axial centimeter.

I.2. Provide the minimum areal density of boron-ten atoms that will be between the fuel assemblies (i.e., B^{10} atoms/Cm^2) for the two types of proposed racks.

The minimum areal boron density for the Indian Point Unit 3 design is as follows:

For 1 poison plate between adjacent fuel assemblies within a rack module, there are 0.00495 gms B-10/cm^2.

For 2 poison plates between adjacent fuel assemblies within a rack module there are 0.0099 gms B-10/cm^2.

I.3. Provide the change in k_∞ for this high density storage lattice with a small change in the areal density of B^{10} atoms between the fuel assemblies.

A calculation was performed of the total reactivity worth of contained boron within the borated stainless steel plates. ΔK_{eff} associated with removing all boron from the poison plate was 0.05 (five percent in reactivity).

A linear interpolation of the worth for small changes in areal density is reasonable and conservative for lightly poisoned plates as utilized for this design. Therefore, for a $\pm$5% change in areal density of B-10 atoms between fuel assemblies, a corresponding change of $\pm$0.0025 in K_{eff} will result.

I.4. Provide the change in k_∞ for this high density storage lattice with a small change in the uranium-235 enrichment.

A change in U-235 enrichment of ± 0.10w/o will result in a change in k_∞ of $\cong 0.0045$.

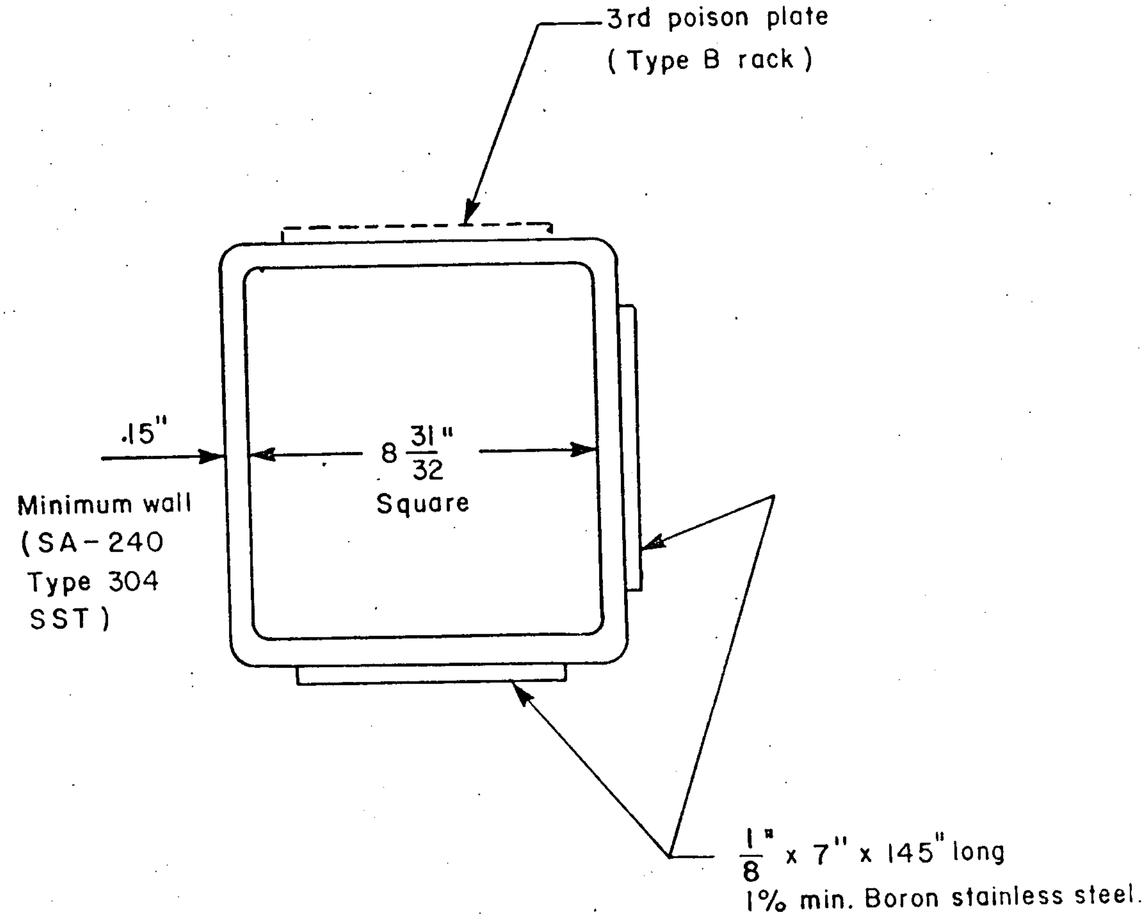
I.5. Since the borated plates are proposed to be only seven inches wide and the fuel assemblies are 8.43 inches wide, describe how the effect of the missing boron at the corners of the assemblies is accounted for in the calculations for k_{∞}.

Figue 2.2.1-2.2.3 of the September 1, 1977 detailed submittal show the geometry for which both the ancillary codes and the KENO calculation were performed. The effects of having 7" poison plates attached to 9" fuel cells were accounted for by utilizing discrete geometry in the analysis.

I.6. State the type and minimum thickness of stainless steel that is proposed to be used for the storage containers on to which the borated plates are to be welded. Provide a cross sectional sketch of the containers with dimensions.

A cross sectional sketch is provided as Attachment A.

ATTACHMENT "A"



TYPE "A" RACK
FUEL STORAGE CELL-CROSS SECTION

INDIAN POINT 3 NUCLEAR POWER PLANT

I.7. Provide a description of the onsite test you intend to perform to verify, within 95 percent confidence limits, that a sufficient number of the welded-on plates will contain the required boron content to maintain the k_{∞} ≤ 0.95.

All borated stainless steel plates were receipt inspected at the rack fabricator's plant. This inspection consisted of visual inspection of all plates for surface condition and 10% random dimensional inspection. At the same time each plate was stamped with a unique identification number traceable to its heat number. In addition, plates from each heat were randomly selected for neutron transmission testing and coupons were sheared for chemical overchecks. These tests were performed in addition to the dimensional and poison verification tests performed by the plate manufacturer to verify the chemical and physical properties of the borated steel plates.

The independent testing has verified that the borated stainless steel does contain boron in excess of the 1.0% by weight upon which the criticality analysis was based.

During fabrication, a map will be made locating each sheet of borated stainless steel by serial number within each rack module. Prior to installation a 100% visual inspection will be made onsite to verify that all B-SS sheets are installed. In addition a random check of the location of the B-SS sheets by serial number will be conducted onsite to independently verify the accuracy and adequacy of the shop generated map.

In summary, 4 independent tests have verified that the borated plates are of the proper dimension and contain equal to or greater than the areal density of B-10 assumed for criticality analysis. As all serialized borated steel plates are visible in the completed rack module, a 100% mapping of each rack module will verify the presence of the neutron absorbing plates at a greater than 95% confidence level.

I.8 Give the details and results of a calculation of the maximum k_{∞} for the case where a fuel assembly is accidentally brought as close as possible to the outside of any module in the pool, which is assumed to be filled with fuel assemblies.

The postulated accident in which a fuel assembly is accidentally brought as close as possible to the outside of any module in the pool has been analyzed. The only area where a fuel assembly could be brought close to the outside of any module is in the cask handling area. The results of the calculation have shown that assuming a conservative, soluble boron concentration of 1000 ppm in the pool water the k_{eff} will not exceed 0.90. This result is in agreement with SRP guidelines and the assumptions are consistent with NRC's Proposed Branch Position for Review and Acceptance of Spent Fuel Storage and Handling Application.

I.9. It appears that the neutron multiplication factor for the 11.25x12 inch storage lattice is going to be very close to the NRC limit of 0.95. Provide conclusive data to show that the maximum possible k_∞ for all possible tolerances and conditions will not exceed 0.95.

The September 1, 1977 detailed version of the submittal contains the requested data.

ATTACHMENT II

RESPONSES TO SEPTEMBER 16, 1977 QUESTIONS BY NRC

Power Authority of the State of New York

II.1 Discuss the dispositions of the material to be removed from the spent fuel pool (e.g., spent fuel racks, seismic restraints) during the proposed modification. If the material to be removed will be disposed of as solid radwaste, provide the volume of the packaged waste.

The present schedule for implementing the spent fuel pool modifications, as proposed in our submittals of July 13 and September 1, 1977, calls for dry installation of the proposed spent fuel storage racks prior to the first refueling. Under these conditions the material removed from the spent fuel pool will not be disposed of as solid radwaste. The removed material will be verified non-radioactive and disposed of as scrap metal.

II.2 Provide the volume of solid waste generated by the replacement of the cartridge filter in the Spent Fuel Pit Cooling Loop. Provide the basis for replacing the cartridge filter and demineralizer. Discuss and quantify any expected changes in the above due to the proposed modification.

The cartridge filter and demineralizer resin will be replaced as necessary when either the pressure drop across the system reaches the design values or when the resin Decontamination Factor decreases below an acceptable level or when the radiation levels exceed a predetermined value.

No changes are planned for the existing demineralizer and filter and no significant additional quantities of solid radioactive waste are expected as a result of this modification.

II.3 In consideration of the increase of storage capacity in the spent fuel pool to 840 assemblies with the concomitent additional radioactivity burden from crud and fission product concentration build-up in the spent fuel pool water, provide an estimate of the additional man-rem expected annually from all operations in the spent fuel pool area as a result of the proposed modification.

The results of the radiological calculations performed for the spent fuel rack expansion program are presented in Sections 3.1 of our September 1, 1977 submittal.

The additional radioactive burden from crud/fission product concentration buildup in the Spent Fuel Pit water due to 840 fuel assemblies was found to be minimal. The decay time between each core off-loading is sufficient to reduce the radioactivity of the Spent Fuel in the pool to only a small negligible fraction of the radioactivity of the full core being off-loaded. Therefore, results of the radiological calculation performed for the spent fuel expansion and presented in Sect. 3.1 are the most conservative anticipated.

II.4 Provide an estimate of the man-rem exposure that will be received during the removal and disposal of the old racks and the installation of the new ones. The estimate should include the number of workers involved in each phase of the operation including divers, if any, the duration of the operation, the exposure rate for each phase of the operations, and the total man-rem received by all workers involved. Relevant experience may be cited.

The proposed dry installation in the presently unused Spent Fuel Storage Pit at IP3 will result in no additional exposure rates.