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Docket No. 50-247

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Mr. John D. O'Toole, Vice President
Nuclear Engineering and Quality Assurance
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Dear Mr. O'Toole:

The Commission has issued the enclosed Amendment No. 75 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated September 7, 1979, as supplemented May 6, 1980.

The amendment revises the Appendix A Technical Specifications to authorize an increase in the capacity of the spent fuel storage pools. Some portions of your Technical Specifications have been modified to meet our requirements. These modifications have been discussed with and agreed to by your staff. Specifically they involve control of heavy loads in the spent fuel building.

Copies of the Safety Evaluation, Environmental Impact Appraisal and the Notice of Issuance and Negative Declaration are also enclosed.

Sincerely,

ORIGINAL SIGNED

John Hannon, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 75 to DPR-26
2. Safety Evaluation
3. Notice of Issuance and Negative Declaration
4. Environmental Impact Appraisal

cc w/enclosures:
See next page

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

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 75
License No. DPR-26

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

   A. The application for amendment by Consolidated Edison Company of New York, Inc. (the licensee) dated September 7, 1979, as supplemented May 6, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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2. Revise existing paragraph 2.B.(2) of Facility Operating License No. DPR-26 as follows:

   2.B.(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Facility Description and Safety Analysis Report, as supplemented and amended and as described in the Commission's authorization through Amendment No. 75 to this license.

3. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-26 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 75 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

4. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas M. Novak, Assistant Director
for Operating Reactors
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 11, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 75

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.8-2	3.8-2
5.3-1	5.3-1
5.4-1	5.4-1

4. At least one residual heat removal pump and heat exchanger shall be operable.

5. During reactor vessel head removal and while loading and unloading fuel from the reactor, T_{avg} shall be $\leq$ 140°F and the minimum boron concentration sufficient to maintain the reactor subcritical by at least 10% $\Delta k/k$. The required boron concentration shall be verified by chemical analysis daily.

6. Direct communication between the control room and the refueling cavity manipulator crane shall be available whenever changes in core geometry are taking place.

7. If the spent fuel pit contains spent fuel, the spent fuel cask shall not be moved over <u>any</u> region of the spent fuel pit until the cask handling system has been reviewed by the Nuclear Regulatory Commission and found to be acceptable. Furthermore, any load in excess of the nominal weight of a spent fuel storage rack and associated handling tool shall not be moved on or above El.-95′ in the Fuel Storage Building. Additionally, loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool shall not be moved over spent fuel in the spent fuel pit. The weight of installed crane systems shall not be considered part of these loads.

8. The containment vent and purge system, including the radiation monitors which initiate isolation, shall be tested and verified to be operable immediately prior to refueling operations.

9. No movement of fuel in the reactor shall be made until the reactor has been subcritical for at least ninety hours. In the event that more than one region of

5.3 REACTOR

Applicability

Applies to the reactor core, reactor coolant system, and emergency core cooling systems.

Objective

To define those design features which are essential in providing for safe system operations.

A. Reactor Core

1. The reactor core contains approximately 87 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. The reactor core is made up of 193 fuel assemblies. Each fuel assembly contains 204 fuel rods.(1)

2. The average enrichment of the initial core is a nomical 2.8 weight per cent of U-235. Three fuel enrichments are used in the initial core. The highest enrichment is a nominal 3.3 weight per cent of U-235.(2)

3. The enrichment of reload fuel will be no more than 3.5 weight per cent of U-235.

5.4 FUEL STORAGE

Applicability

Applies to the capacity and storage arrays of new and spent fuel.

Objective

To define those aspects of fuel storage relating to prevention of criticality in fuel storage areas.

Specification

1. The spent fuel pit structure is designed to withstand the anticipated earthquake loadings as a Class I structure. The spent fuel pit has a stainless steel liner to insure against loss of water.

2. The new and spent fuel storage racks are designed so that it is impossible to insert assemblies in other than an array of vertical fuel assemblies with the sufficient center-to-center distance between assemblies to assure $K_{eff} \leq 0.95$ even if unborated water were used to fill the pit and with the fuel loading in the assemblies limited to 43.9 grams of U-235 per axial centimeter of fuel assembly.

3. Whenever there is fuel in the pit, the spent fuel storage pit is filled and borated to the concentration to match that used in the reactor cavity and refueling canal during refueling operations.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 75 TO FACILITY OPERATING LICENSE NO. DPR-26

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

1.0 Introduction

By application dated September 7, 1979, as supplemented by letter dated May 6, 1980, the Consolidated Edison Company proposed to install high density fuel storage racks at the Indian Point Nuclear Generating Unit No. 2 Plant.

The proposed expansion is to increase storage capacity from 482 to 980 fuel assemblies. Assuming no offsite shipments of fuel assemblies, the existing storage facility will not have the capacity for a full core discharge after the next refueling. The expansion program will extend the capacity for a full-core discharge into 1991.

The modification consists of replacing existing fuel assembly racks with high density, free standing storage racks without changing the basic structural geometry of the spent fuel pool (SPF).

2.0 Discussion

The Indian Point Unit 2 high density spent fuel storage space has been designed to provide a maximum storage capacity of 980 locations arranged in 10 high density stainless steel racks. Each rack consists of an assembly of either 2 x 2 or 2 x 3 modular cell units arranged in arrays of 8 x 8, 8 x 9, 9 x 9, or 10 x 10. Each modular storage cell is a type 304 stainless steel box. The box is supported by and welded to the rack base grid structure, which is also constructed from type 304 stainless steel. The top opening of the storage cell is flared to facilitate insertion of the fuel assembly. The bottom member of the storage cell provides the level support surface required for the fuel assembly and contains the cooling flow orifice.

The wall of each storage cell is made from Type 304 stainless steel, 0.0825 inches thick. Attached to the outside cell walls is a borated stainless steel sheet which is 0.1 inches thick and 7 inches wide containing 1.1 percent by weight (w/o) boron for criticality control. The cells

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are maintained at a center-to-center spacing of 10-15/16 inches by welded spacers. The spacers are in the form of continuous bars provided at the top of the storage cells to ensure that the required pitch is maintained between storage cells in both directions under lateral load conditions. The spacer bars are supported on the storage cells and serve to maintain the vertical alignment of the cells. Support pads attached to the bottom of the rack base raise the rack above the pool floor to the height required to clear existing anchor pins on the pool floor area and to provide an adequately sized cooling water supply plenum for natural circulation. Each support pad contains a remotely adjustable jack screw to permit the rack to be leveled following wet installation. The rack support pads will rest directly on the pool floor wherever possible. In case of interference with the existing floor shims, the pads will rest on local plates designed to bridge over the shims.

The storage racks are free-standing structures that are free to slide horizontally on the pool floor. The storage racks are positioned on the pool floor so that adequate clearances are provided between racks and between the racks and pool structures to avoid impacting of the sliding racks during seismic events. The horizontal seismic loads transmitted from the rack structure to the pool floor are only those associated with friction between the rack structure and the pool liner. The vertical deadweight and seismic loads are transmitted directly to the pool floor by the support pads .

3.0 Evaluation

3.1 Structural and Mechanical

The spent fuel storage racks have been designed to meet seismic Category I requirements. Therefore, the high density racks are designed to withstand the most severe combinations of environmental, accidental, and seismic loads in accordance with Standard Review Plan Section 3.8.4. The structural acceptance criteria is in conformance with: Part I AISC "Specification for the Design Fabrication and Erection of Structural Steel for Buildings," February 12, 1969; Standard Review Plan Section 3.8.4.II.5; and NRC Position Paper on Fuel Storage Racks contained in a letter from Brian K. Grimes to all power reactor licensees dated April 14, 1978 and amended January 18, 1977. In addition, the racks and their support are designed to withstand both the uplift force associated with the fuel-handling equipment jamming or malfunctioning, and the impact emergy associated with the postulated falling objects. In all of these accidents, the damage, if any, is limited to local areas, thus the functional capability of the fuel racks is not impaired.

The seismic analysis, performed by the licensee, of the storage racks employed both linear and non-linear methods. Linear seismic analysis was performed to verify the structural adequacy of the design to withstand the loadings encountered during the severe and extreme conditions of Operating Basis Earthquake (OBE) and Design Basis Earthquake (DBE). Non-linear time history seismic analyses were performed to evaluate the maximum sliding of the storage racks and to determine the maximum frictional resistance load transmitted by the storage racks to the pool floor liner during the DBE. The time history analysis was performed for two assumed values of surface friction, which bound the actual coefficient of friction. In addition to friction forces during seismic movement, nonlinearities of fuel-to-rack impact are accounted for in the analysis. Both linear and non-linear analyses used lumped mass stick models.

Results of the seismic and structural analysis indicate that all the racks are capable of withstanding the loads associated with all the design loading conditions based on the original seismic design criteria.

The criteria used in the analysis and design of the spent fuel storage racks to account for anticipated loads that may be imposed on the racks during their service lifetime, are in conformance with the NRC staff Position Paper on Fuel Storage Racks, and with codes, standards and specifications acceptable to the staff. We conclude therefore that the spent fuel rack design meets the applicable portions of General Design Criteria 2, 61 and 62, and is acceptable.

3.2 Materials

The proposed spent fuel storage racks are fabricated of Type 304 stainless steel with the exception of the adjusting bolts of the rack feet. These bolts are made from Type 17-4 PH stainless steel. The 17-4 PH stainless steel will be heat-treated at 1150°F and the heat treat scale will be removed by grit blasting.

The Indian Point Unit 2 spent fuel storage pool is filled with high purity water containing approximately 2,000 ppm boron. The boron is added as boric acid. Tight controls are placed on impurities in this water such as chlorides and fluorides that could lead to stress corrosion cracking (SCC). The new spent fuel storage cells are constructed of a Type 304 stainless steel box with a boron stainless steel plate inserted in the racks and held to the stainless steel boxes by welded brackets. Care has been taken to minimize sensitization of the material during fabrication. The Type 304 stainless steel sheet and the boron stainless steel poison plate were purchased in the annealed condition. The welding procedures are low heat input methods, to minimize sensitization of the stainless steel.

The pool liner, rack lattice structure and fuel storage tubes are stainless steel which is compatible with the storage pool environment. In this environment of oxygen-saturated borated water, the corrosive deterioration of the Type 304 stainless steel should not exceed a depth of 6.00×10^{-5} inches in 100 years, which is negligible relative to the initial thickness. Dissimilar metal contact corrosion (galvanic attack) between the stainless steel of the pool liner, rack lattice structure, fuel storage tubes and the Inconel and the Zircaloy in the spent fuel assemblies will not be significant because all of these materials are protected by highly passivating oxide films and are therefore at similar potentials.

The history of the corrosion resistance of stainless steel in spent fuel storage pools at PWR sites has been excellent. In these cases, the spent fuel storage pool water chemistry has been equivalent to that at Indian Point. There have been no known incidences of any SCC developing on spent fuel storage racks. Should it occur, it would tend to be highly localized and have little or no significant effect on the structural capabilities of the racks themselves.

We conclude that the corrosion that will occur in the Indian Point Station Unit 2 spent fuel storage pool environment should be of little significance during the remaining life of the plant. Components in the spent fuel storage pool are constructed of alloys which have a low differential galvanic potential between them and have a high resistance to general corrosion and localized corrosion. No materials other than stainless steels or boron stainless steels are present in these structures. Therefore, with the selection of Type 304 and boron stainless steel, and in water chemistry with careful monitoring of the coolant for possible corrosive impurities, no significant corrosion should occur in the spent fuel storage racks at Indian Point Unit 2 for a period well in excess of the 40 years designed life of the unit.

We, therefore, conclude that the environmental compatibility and stability of the materials used in the spent fuel storage pool is adequate based on test data and actual service experience in operating reactors. We find that the selection of appropriate materials of construction by the licensee meets the requirements of 10 CFR Part 50, Appendix A, Criterion 61, by having a capability to permit appropriate periodic inspection and testing of components, and Criterion 62, by preventing criticality by maintaining structural integrity of components and is therefore acceptable.

3.3 Criticality Considerations

A detailed nuclear analysis was performed by the licensee for the fuel storage racks for all anticipated normal and abnormal configurations of fuel assemblies within the fuel storage racks. The purpose of the analysis was to demonstrate that the K_{eff} of the system is less than

the criticality criterion under all anticipated conditions. The criticality criterion used was that K_{eff} must be less than 0.95 for 15 x 15 Westinghouse fuel assemblies containing fuel enriched to 3.5 percent by weight of U-235. Conservative assumptions about the fuel assemblies and racks were used in the calculations. These assumptions included only fresh fuel with a 3.5 w/o U^{235} enrichment is stored in the fuel pit, absorption of fuel assembly spacers was not considered, burnable poison in the fuel assemblies was not considered, and soluble poison in the pool water was not considered.

The analysis was performed with the KENO-IV Monte Carlo code. A three dimensional 123-group calculation was used to determine K_{eff}. The reference configuration contained an infinite square array of storage locations spaced on 10-15/16 inch centers. This is a conservative assumption since the arrays are finite structures. In addition the KENO results were compared to results obtained with HAMMER, a multigroup transport code and EXTERMINATOR, a two dimensional multigroup diffusion program. The code has also been benchmarked versus several critical experiments. The conclusion of the benchmarking was that KENO-IV with the 123-group cross sections gives conservative results.

Parametric sensitivity calculations were performed in order to assess the effect of variation of the following items: fuel cell eccentric configuration, fuel enrichment, fuel rack pitch, cell wall thickness, poison content, borated steel sheet thickness, and cell inside dimension. The statistical combination of the variations of the effective multiplication leads to $K_{eff} = .933 \pm .010$ where .933 is the value for the normal reference configuration. Abnormal pool water temperature was investigated. However, the pool has a negative temperature coefficient with the highest K_{eff} occurring at 68°F.

An audit calculation was performed by our Brookhaven National Laboratory consultant of a configuration calculated by the licensee. The results obtained by our consultants confirmed those obtained by the licensee.

The criticality calculations were done with specified enrichment of 3.5 w/o of U^{235}. Given that the mass of uranium per assembly is 459kg and the active fuel length is 144 inches, a 3.5 w/o of U^{235} limit corresponds to a maximum of 43.9 grams of U^{235} per axial centimeter of fuel assembly length. Since the criticality calculations were based on these assumptions, we shall include as a technical specification limit both the 3.5 w/o of U^{235} maximum enrichment and the corresponding 43.9 grams of U^{235} per axial centimeter of fuel assembly length.

The quantity of boron in the steel sheets is assured with a quality assurance program, by the fabrication technique, and a system of coupon surveillance throughout the lifetime of the racks.

Based on our review we conclude that the reactivity aspects of the proposed rack design are acceptable. This conclusion is based on the following:

(a) The calculations were performed with codes which have been benchmarked by comparison to critical experiments.

(b) Conservative assumptions were made.

(c) Biases and uncertainties have been evaluated.

(d) The results of the calculations meets our acceptance criterion of $K_{eff} \leq .95$ and

(e) The licensee proposed a satisfactory program to retain the boron in the pool and assure its presence throughout the life of the storage facility.

3.4 Spent Fuel Cooling

The spent fuel pool cooling system consists of two pumps and one heat exchanger. One pump is used for normal operation and the second pump serves as a backup. The heat exchanger is cooled by the component cooling water system. The spent fuel pool heat removal capability is approximately 11 x 10^6 BTU/HR at 120°F pool temperature and 24.5 x 10^6 BTU/HR at 155°F pool temperature. The cooling system connections to the pool are located 6 feet above the fuel assemblies to protect against inadvertent drainage below the fuel level. In event of a loss of the cooling system, makeup water is available from three sources: demineralized water (unborated) from the primary water tank, borated water from the refueling water storage tank, and non-demineralized water from the fire protection system.

The future refueling cycle for Indian Point 2 will be at eighteen month intervals. A total of 76 fuel assemblies of the 193 assembly core will be removed and stored in the spent fuel pool after each cycle. To limit the decay heat load, the 76 fuel assemblies will be removed from the reactor core and stored in the spent fuel pool 90 hours after reactor shutdown. The future refueling discharges planned together with fuel presently stored in the pool (a total of eleven discharges) will use 787 of the 980 available storage spaces. Thus, adequate storage will be maintained for a full core discharge. In the event of a full-core discharge, the decay heat load will be limited by requiring a 400 hour decay time after shutdown before core discharge.

To calculate the heat loads for the discharges of spent fuel to the pool, the licensee used Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling" including recommended uncertainty factors. For times greater than 10^7 seconds an uncertainty factor of 0.1 was used. The reactor thermal power used in the calculation was 2813 Mwt, which is 102% of the rated thermal power of 2758 Mwt. The maximum normal heat load which occurs after the eleventh refueling discharge, was calculated to be 18.6 x 10^6 BTU/HR. The normal heat load resulted in a maximum bulk pool temperature of approximately 140°F which is in compliance with Standard Review Plan Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System." The maximum abnormal heat load results from a full-core discharge after the last normal refueling discharge and was calculated to be 24.5 X 10^6 BTU/HR. The abnormal heat load resulted in a maximum bulk pool

temperature of approximately 155°F. The American National Standard 57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations" indicates that the maximum pool temperature should not exceed 150°F under normal operating conditions with all storage full. The conservative case of a full-core discharge approaches the maximum temperature recommended by ANS 57.2 for normal conditions.

To verify that natural circulation of the pool water for the proposed expanded rack configuration provides adequate cooling of all fuel assemblies, the licensee performed a thermal-hydraulic analysis. The adequacy of natural circulation was verified for selected fuel assemblies in the most critical area in terms of cooling for the fuel assemblies. The selected assemblies were assumed to have sustained a 4 year equivalent full power operation and have decayed 90 hours. The pool temperature was conservatively assumed to be 155°F and only the downcomer area between the selected row and the pool wall was used. Larger downcomer areas (i.e., the cask area, rack to rack spacing, the failed fuel rack area) were neglected. Based on these conservative assumptions, the maximum assembly outlet temperature was approximately 12°F less than the boiling temperature. Additionally, the licensee demonstrated that, in the conservative case where all the fuel storage racks were moved to one corner of the pool (no clearance between racks would exist), adequate cooling for the fuel assemblies would be maintained via the downcomer area at the opposite corner of the pool and across the ten-inch rack bottom plenum. Also, the licensee indicated that if total flow blockage at the bottom entry hole to a fuel assembly were to occur, adequate cooling would be maintained by pool water draining into the rack cell.

In the event of a complete failure of the spent fuel pool cooling system, the licensee calculated that the time available before makeup water to the pool is required, is 8.1 hours for the normal heat load and 4.9 hours for the abnormal heat load. The required makeup rate is approximately 40 gpm for the normal heat load and 50 gpm for the abnormal heat load. Each of the three makeup water sources can be initiated in the required time. The primary water tank supply can be provided to the pool by either of two 150 gpm primary water makeup pumps. The primary water storage tank and the makeup pumps are seismic Category I; however, portions of the piping from the tank to the spent fuel pool are classified seismic Category II. The licensee indicated that the non-Category I piping was analyzed for Category I condition and therefore is equivalent. Sufficient makeup rates are available from the refueling water storage and the fire protection system; however, neither source uses seismic Category I piping.

We have reviewed the calculated decay heat values and conclude that the heat loads are consistent with the Branch Technical Position ASB 9-2 and therefore are acceptable. The spent fuel pool cooling system performance and the natural circulation calculation have been reviewed and we conclude that the pool cooling is adequate. The available makeup systems, their respective makeup rates and the time required before makeup is needed has been reviewed and found acceptable. We have reviewed the loading combinations and stress limits for the non-Category I piping from the primary water tank and conclude the piping is equivalent to Category I piping. Based on the above, we conclude that the spent fuel pool cooling system is acceptable.

3.5 Spent Fuel Pool Cleanup System

The spent fuel pool cleanup system consists of a filter demineralizer (mixed bed resin), filters, and associated piping, valves, and fittings. The system is designed to remove corrosion products, fission products, and impurities from the pool water. Pool water purity is monitored by weekly chemical and radiochemical analysis. Demineralizer resin will be replaced when pool water samples show reduced decontamination effectiveness. The licensee indicated that no change or equipment addition to the spent fuel pool cleanup system is necessary to maintain pool water quality for high density fuel storage.

Past experience showed that the greatest increase in radioactivity and impurities in spent fuel pool water occurs during refueling and spent fuel handling. The refueling frequency, the amount of core to be replaced for each fuel cycle, and the frequency of operating the spent fuel pool cleanup system are not expected to increase as a result of high density fuel storage. The chemical and radionuclide composition of the spent fuel pool water is not expected to change as a result of the proposed high density fuel storage. Past experience also shows that there is no significant leakage of fission products from spent fuel stored in pools after the fuel has cooled for several months.

Therefore, the increased quantity of spent fuel to be stored will not contribute significantly to the amount of radioactivity from fission products in the spent fuel pool water. We conclude the proposed expansion of the spent fuel pool will not appreciably affect the capability and capacity of the spent fuel pool cleanup system. Accordingly, no change to the present cleanup system is needed. More frequent replacements of filters or demineralizer resin, required when the differential pressure exceeds 11 psid or decontamination effectiveness is reduced, can offset any potential increase in radioactivity and impurities in the pool water as a result of the increased quantity of spent fuel. Thus, we have determined that the existing fuel pool cleanup system with the proposed high density fuel storage (1) provides the capability and capacity of removing radioactive materials, corrosion

products, and impurities from the pool and thus meets the requirements of General Design Criterion 61 in Appendix A of 10 CFR Part 50 as it relates to appropriate systems to fuel storage; (2) is capable of reducing occupational exposures to radiation by removing radioactive products from the pool water, and thus meets the requirements of Section 20.1(c) of 10 CFR Part 20, as it relates to maintaining radiation exposures as low as is reasonably achievable; (3) collects radioactive materials in the pool water in filters and demineralizers, and thus meets Regulatory Position C.2.f(2) of Regulatory Guide 8.8, as it relates to reducing the spread of contaminants from the source; and (4) removes suspended impurities from the pool water by filters, and thus meets Regulatory Position C.2.f(3) of Regulatory Guide 8.8, as it relates to removing crud from fluids through physical action.

We conclude that the existing spent fuel pool cleanup system meets GDC61, Section 20.1(c) of 10 CFR Part 20 and the appropriate Sections of Regulatory Guide 8.8 and, therefore, is acceptable for the proposed high density fuel storage.

3.6 Installation of Racks and Fuel Handling

The spent fuel storage pool and the spent fuel pool cooling system are located in the fuel storage building. The spent fuel storage pool is a seismic Category I structure and the pool contains a stainless steel liner. The fuel storage building is served by an overhead crane of 40 ton main hoist capacity and 5 ton auxiliary hoist capacity. Mechanical stops located on the crane rails prevent travel of the crane over the storage pool. The spent fuel pool is served by the spent fuel bridge hoist with a 2000 lb. hoist capacity.

During the expansion program, the fuel building crane of 40 ton capacity will be used to remove all of the present racks and insert twelve new racks. The heaviest rack to be handled is approximately 20 tons. Additionally brackets, support plates, light fixtures and other loads will be removed and inserted during the modification. The heaviest loads other than a rack which will be handled are the 1050 lb. rack templates. None of the loads will be handled directly over spent fuel; however, some loads will be handled in the vicinity of spent fuel. Procedures have been developed to maximize the spacing between lifted loads and spent fuel. There is no equipment, essential in the safe shutdown of the reactor or employed to mitigate the consequence of an accident which is beneath, adjacent to or otherwise within the area of influence of any loads that will be handled during the expansion modification.

The licensee submitted step-by-step procedures for the rack removal and installation sequence. A minimum horizontal spacing of three empty storage locations will be maintained between lifted racks and the stored fuel. Only a limited number of handling operations approach the minimum spacing requirement. All other fuel rack movements would maintain at least one empty rack space. Thus, the potential for fuel damage is limited to a small number of handling operations, a limited number of fuel assemblies, and limited to a failure of the lifting device or crane which would tip the load in the direction of the spent fuel. The licensee stated that the restrictions on the motion of the spent fuel crane meets the intent of Section 5.1.6 of NUREG-0612 in that a failure of the load handling system will not exceed the guidelines of Section 5.1 of NUREG-0612. A load drop during the expansion modifications would not damage the pool or the pool liner beyond the capability of the makeup sources available.

The potential for a load drop during the expansion modification has been reduced by implementation of the interim action for control of heavy load as outlined by Enclosure 2 to the NRC generic letter of December 22, 1980. In addition, the handling of the racks will utilize special lifting devices with a stress design factor exceeding that required by ANSI-N 14.6-1978. A load test of the special lifting devices will be conducted with a test load of 150% of the rated load prior to usage for rack handling in the spent fuel pool.

The maximum uplift force available from the load lifting devices spanning the spent fuel pool which could be applied to the free standing unanchored fuel storage racks is limited to the spent fuel bridge hoist with a lifting capacity of 2000 lbs. The maximum load will be controlled with a monitor control system with an overload cutoff device.

In accordance with Section IV(4) of the enclosure to NRC letter dated April 14, 1978, the specific loads and load combinations conform with 3.8.4-II-3 of the Standard Review Plan.

The restriction on the motion, the design and test of the special lifting devices, and the implementation of the interim action outlined by NUREG-0612 will reduce the possibility of dropping a load onto stored fuel assemblies to an acceptable level. We conclude that the health and safety of the public will not be endangered by the expansion program of the Indian Point 2 spent fuel pool and is, therefore, acceptable.

3.7 Fuel Handling

The NRC staff has underway a generic review of load handling operations in the vicinity of spent fuel pools to determine the likelihood of a heavy load impacting fuel in the pool and, if necessary, the radiological consequences of such an event. Because Indian Point 2 will be required

(by technical specification) to (1) prohibit loads greater than the nominal weight of a fuel assembly and handling tool to be transported over spent fuel in the SFP, and, (2) prohibit loads of weight greater than the weight of a fuel rack to be moved in the SFP building above the operating deck level, we have concluded that the likelihood of any other load handling accident is sufficiently small, the proposed modification is acceptable, and no additional restrictions on load handling operations in the vicinity of the SFP are necessary while our generic review is underway.

The potential consequences of fuel handling accidents (i.e., rupture of fuel pins in one fuel assembly and the subsequent release of the radioactive inventory within the gap) in the spent fuel pool area presented in the SE dated November 16, 1970 are not changed by the use of high density racks, since the amount of fuel damage in this accident remains unchanged.

3.8 Occupational Radiation Exposure

We have reviewed the licensee's Final Design Report for Reracking the Indian Point Unit No. 2 Spent Fuel Pool, describing actions for the removal and disposal of the existing racks and the installation of the new racks, including actions to maintain occupational radiation dose ALARA, the licensee's person-rem estimate for the operation, and the impact of the additional fuel on occupational doses.

The modification will be performed by rearranging the spent fuel elements stored in the pool in such manner as to yield the lowest dose rates in the area to be occupied by divers. Before work is begun, measurements of dose rates will be made both in and around the pool in order to obtain a dose rate "map" to guide the selection of working times and places. The radiation work permit (RWP) will be based on these measurements and calculations and will specify precautions and protective equipment to be used to maintain occupational exposures ALARA.

The existing spent fuel racks will be removed, washed down and decontaminated. The results of the decontamination will determine whether the intact racks will be disposed as clean waste (if the radiation surveys show less than 1000 cpm/100 cm^2 loose contamination and less than 200 cpm fixed contamination) or whether the racks will have to be cut up and packed for disposal as radioactive waste. Radiation protection precautions will be specified on RWPs for these operations also.

The total occupational dose for this operation is estimated by the licensee to be approximately 25 person-rem. We consider this to be a realistic estimate because it is based on the licensee's detailed breakdown of occupational exposure for each phase of the modification

(such as fuel movement, removal and installation of racks, diver support, decontamination and disposal of the racks), and also because the licensee's person-rem estimate is within estimates provided by other licensees for similar reracking operations. In arriving at this estimate, the licensee considered the number of individuals performing a specific job, their occupancy time and the average dose rate in the area where the job will be performed. Although diver support will be required during the modification, their cumulative dose equivalent will be less than 5 man-rem for a total of estimated 210 hours of in-water activity.

In addition to the reracking operation, we have examined the increment in onsite occupational dose resulting from the proposed increase in stored fuel assemblies on the basis of information provided by the licensee for dose rates in the area of the spent fuel storage pool. The spent fuel assemblies will contribute a negligible amount to dose rates in the operating level area because of sufficient depth of water above the stored spent fuel. The occupational radiation dose resulting from the additional spent fuel in the pool represents a negligible impact. Based on present and projected operations in the spent fuel pool area, we estimate that the proposed modification should add less than one percent to the total annual occupational radiation dose. The small increase in dose will not affect the licensee's ability to maintain individual occupational doses as low as reasonably achievable and within the limits of 10 CFR Part 20.

Thus, we conclude that the storing of additional fuel in the spent fuel storage pool at Indian Point Station will not result in any significant increase in doses received by workers.

3.9 Radioactive Waste Treatment

The plant contains waste treatment systems designed to collect and process the gaseous, liquid, and solid wastes that might contain radioactive material. Our evaluation of the radiological considerations supports the conclusion that the proposed modification to the Indian Point, Unit No. 2 spent fuel pool is acceptable because:

(1) The conclusions of the evaluation of the waste treatment systems, as found in the Indian Point, Unit 2 Safety Evaluation Report dated November 16, 1970, are unchanged by the modification of the SFP.

(2) The existing spent fuel pool (SFP) cleanup system is adequate for the proposed modification. Our evaluation of the SFP cleanup system is found in Section 3.5 of this report.

4.0 Conclusion

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

Date: January 11, 1982

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

ENVIRONMENTAL IMPACT APPRAISAL BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 75 TO FACILITY OPERATING LICENSE NO. DPR-26

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

1.0 Introduction

By application dated September 7, 1979, as supplemented by letter dated May 6, 1980, the Consolidated Edison Company proposed to install high density fuel storage racks at the Indian Point Nuclear Generating Unit No. 2 Plant.

The proposed expansion is to increase storage capacity from 482 to 980 fuel assemblies. Assuming no offsite shipments of fuel assemblies, the existing storage facility will not have the capacity for a full core discharge after the next refueling. The expansion program will extend the capacity for a full-core discharge into 1991.

The modification consists of replacing existing fuel assembly racks with high density, free standing storage racks without changing the basic structural geometry of the SFP.

The environmental impact of the existing Indian Point 2 fuel storage pool was considered in Amendment No. 14 to Facility Operating License No. DPR-26, issued December 16, 1975. The purpose of this appraisal is to evaluate any additional environmental effects of this proposed increase in storage capacity. The Indian Point-2 spent fuel storage system is described in our concurrently issued Safety Evaluation.

2.0 Generic Environmental Impact Statement

A Final Generic Environmental Impact Statement (FGEIS) on Handling and Storage of Spent Light Water Power Reactor Fuel (NUREG-0575, Volumes 1-3) was issued by NRC in August 1979. The NRC staff evaluated and analyzed alternatives for handling and storage of spent light water power reactor fuel with emphasis on long-range policy. Consistent with the long-range policy, the storage of spent fuel addressed in the FGEIS is considered to be interim storage to be used until the issue of permanent disposal is resolved and implemented.

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One spent fuel storage alternative considered in detail in the FGEIS is the expansion by licensees of onsite fuel storage capacity by modification of existing spent fuel pools (SFPs). By the date of issuance of the FGEIS (August 1979), 40 applications for SFP capacity expansions had been approved with the finding in each case that the environmental impact of the proposed increased storage was negligible. However, since there are variations in storage pool designs and limitations caused by the spent fuel already stored in some of the pools, the FGEIS recommended that licensing reviews be done on a case-by-case basis to resolve plant-specific concerns. This appraisal accomplishes that recommendation.

3.0 Need for Increased Storage Capacity

On December 16, 1975 the NRC issued Amendment No. 14 to Facility Operating License No. DPR-26 for Indian Point Unit No. 2 authorizing a spent fuel pool expansion from 264 to 482 storage locations. Presently there are 200 spent fuel assemblies stored in the spent fuel pool. The existing storage capacity will be reached in 1981 with a full core reserve maintained.

The proposed increase would be accomplished by replacing the existing spent fuel storage racks with new, more compact, neutron absorbing racks. The proposed rack design is discussed in the concurrently issued SER. This modification would extend spent fuel storage capacity through 1991. A more immediate concern is that of maintaining sufficient room in the SFP to off-load a full core should this be necessary for inspection or repair of reactor internal equipment or piping. While this capability is not necessary to protect the health and safety of the public, it is desirable to reduce occupational exposures.

4.0 Radioactive Wastes

The plant contains waste treatment systems designed to collect and process the gaseous, liquid and solid waste that might contain radioactive material. The waste treatment systems are evaluated in the Final Environmental Statement (FES) dated September 1972. There will be no change in the waste treatment systems described in Section III.E.2 of the FES because of the proposed modification.

5.0 Spent Fuel Pool (SFP) Cleanup System

The SFP cleanup system is part of the pool cooling system. It consists of a demineralizer with inlet and outlet filters, and the required piping, valves, and instrumentation. There is also a separate skimmer system to remove surface dust and debris from the SFP. This cleanup system is similar to such systems at other nuclear plants which maintain concentrations of radioactivity in the pool water at acceptably low levels.

We expect only a small increase in radioactivity released to the pool water as a result of the proposed modification. We, therefore, conclude the spent fuel pool cleanup system is adequate for the proposed modification and will keep the concentrations of radioactivity in the pool water to acceptably low levels.

6.0 Environmental Impacts of the Proposed Action

6.1 Non-radiological

The environmental impact of Indian Point Unit 2, as designed, was evaluated in the Final Environmental Statement (FES) dated September 1972. Increasing the number of assemblies stored in the existing fuel pool will not cause any new non-radiological environmental impacts. The amount of waste heat emitted by Indian Point Unit 2 will only increase slightly, resulting in no measurable increase in impact upon the environment.

6.2 Radiological Introduction

The potential offsite radiological environmental impacts associated with the expansion of the spent fuel storage capacity were evaluated and determined to be environmentally insignificant as addressed below.

During the storage of the spent fuel under water, both volatile and non-volatile radioactive nuclides may be released to the water from the surface of the assemblies or from defects in the fuel cladding. Most of the material released from the surface of the assemblies consists of activated corrosion products such as Co-58, Co-60, Fe-59 and Mn-54 which are not volatile. The radionuclides that might be released to the water through defects in the cladding, such as Cs-134, Cs-137, Sr-89 and Sr-90, are also predominately nonvolatile. The primary impact of such nonvolatile radioactive nuclides is their contribution to radiation levels to which workers in and near the SFP would be exposed. The volatile fission product nuclides of most concern that might be released through defects in the fuel cladding are the noble gases (xenon and krypton), tritium and the iodine isotopes.

Experience indicates that there is little radionuclide leakage from spent fuel stored in pools after the fuel has cooled for several months. The predominance of radionuclides in the SFP water appear to be radionuclides that were present in the reactor coolant system prior to refueling (which becomes mixed with water in the SFP during refueling operations) or crud dislodged from the surface of the spent fuel during transfer from the reactor core to the SFP . During and after refueling, the SFP cleanup system reduces the radioactivity concentrations considerably. It is theorized that most failed fuel contains small, pinhole-like perforations in the fuel cladding at the reactor operating condition of approximately 800°F. A few weeks after refueling, the spent fuel cools in the SFP so that fuel clad temperature is relatively cool, approximately 180°F. This substantial temperature reduction should reduce the rate of release of fission products from the fuel pellets and decrease the gas pressure in the gap between pellets and clad, thereby tending to retain the fission products within the gap.

In addition, most of the gaseous fission products have short half-lives and decay to insignificant levels within a few months. Based on the operational reports submitted by the licensees or discussions with the operators, there has not been any significant leakage of fission products from spent light water reactor fuel stored in the Morris Operation (MO) in Morris, Illinois (formerly Midwest Recovery Plant), or at NFS storage pool at West Valley, New York. Spent fuel has been stored in these two pools which, while it was in a reactor, was determined to have significant leakage and was therefore removed from the core. After storage in the onsite SFP, this fuel was later shipped to either MO or NFS for extended storage. Although the fuel exhibited significant leakage at reactor operating conditions, there was not significant leakage from this fuel in the offsite storage facility.

6.2.1 Radioactive Material Released to the Atmosphere

With respect to releases of gaseous materials to the atmosphere, the only radioactive gas of significance which could be attributable to storing additional fuel assemblies for a longer period of time would be the noble gas radionuclide Krypton-85. As dicussed previously, experience has demonstrated that after spent fuel has decayed 4 to 6 months, there is no longer a significant release of fission products--including Krypton-85--from stored fuel cladding defects. Seventy-two (72) fuel assemblies are expected to be stored each 18 months for an average annual input of 48 fuel assemblies per year. Since space must be reserved to accommodate a complete reactor core unloading operation (nominally 200 fuel assemblies), the useful pool capacity is 980-200 = 780 fuel assemblies. At an input of 72 fuel assemblies every eighteen months, it is seen that the pool has a useful storage capacity of approximately 16 years.

For the simplest case, one can assume that all of the Kr-85 that is going to leak from depleted fuel is going to do so in the 18 month interval between refuelings. In other words, all of the Kr-85 available for release is assumed to come out of the fuel before the next batch of fuel enters the pool. Our calculations show that the expected release of Kr-85 from a 72 fuel assembly refueling is approximately 125 Ci each 18 months or an average of about 84 Ci/yr. As far as the potential dose to offsite populations is concerned, this is actually the worst case, since each refueling would generate a new batch of Kr-85 to be released. As more and more fuel is added to the pool, one might think that this would increase the releases--but according to the terms of our model, this is not the case, since all of the Kr-85 available for release has already left the depleted fuel previously stored in the pool before the next batch enters, with the result that the annual releases are not cumulative but remain approximately the same.

Using the most conservative approach, i.e., assuming all Kr-85 from the failed fuel from a single fuel reload is released prior to the next reload operation, calculations show that this type of release is not cumulative.

In other words, the enlarged capacity of the pool has no effect on the total amount of Kr-85 released to the atmosphere each year. Thus, we conclude that the proposed modifications will not have any significant impact on exposures offsite.

Assuming that the spent fuel will be stored onsite for several years, Iodine-131 releases from spent fuel assemblies to the SFP water will not be significantly increased because of the expansion of the fuel storage capacity since the Iodine-131 inventory in the fuel will decay to negligible levels between refuelings.

Storing additional spent fuel assemblies is not expected to increase the bulk water temperature above 140°F which is approximately the value used in the design analysis. Therefore, it is not expected that there will be any significant change in the annual release of tritium or iodine as a result of the proposed modifications from that previously evaluated in the FES. Most airborne releases of tritium and iodine result from evaporation of reactor coolant, which contains tritium and iodine in higher concentrations than the spent fuel pool. Therefore, even if there were a higher evaporation rate from the spent fuel pool, the increase in tritium and iodine released from the plant as a result of the increase in stored spent fuel would be small compared to the amount normally released from the plant and that which was previously evaluated in the FES. If it is desired to reduce levels of radioiodine, the air can be diverted to charcoal filters for the removal of radioiodine before release to the environment. In addition, the station radiological effluent Technical Specifications which are not being changed by this action, limit the total releases of gaseous activity.

6.2.2 Solid Radioactive Wastes

The concentration of radionuclides in the pool water is controlled by the filters and the demineralizer and by decay of short-lived isotopes. The activity is highest during refueling operations when reactor coolant water is introduced into the pool, and decreases as the pool water is processed through the filters and demineralizer. The increase of radioactivity, if any, due to the proposed modification, should be minor because of the capability of the cleanup system to continuously remove radioactivity to acceptable levels.

The licensee does not expect any significant increase in the amount of solid waste generated from the spent fuel pool cleanup systems due to the proposed modification. While we agree with the licensee's conclusion, as a conservative estimate we have assumed that the amount of solid radwaste may be increased by an additional two resin beds a year due to the increased operation of the spent fuel pool cleanup system. The annual average volume,

per unit, of solid wastes shipped from the Indian Point site during 1978 through 1980 was 13,900 cubic feet. If the storage of additional spent fuel does increase the amount of solid waste from the SFP cleanup systems by about 160 cubic feet per year, the increase in total waste volume shipped would be approximately 1% and would not have any significant additional environmental impact.

The present spent fuel racks to be removed from the SFP because of the proposed modification may be permanently contaminated and if so will be disposed of as low level solid waste. The licensee has estimated a maximum of 10,500 cubic feet of solid radwaste will be removed from the plant because of the proposed modification. Averaged over the lifetime of the plant, this would increase the total waste volume shipped from the facility by less than 3%. This will not have any significant additional environmental impact.

6.2.3 Radioactive Material Released to Receiving Waters

There should not be a significant increase in the liquid release of radionuclides from the plant as a result of the proposed modification. Since the SFP cooling and cleanup system operates as a closed system, only water originating from cleanup of SFP floors and resin sluice water need be considered as potential sources of radioactivity.

It is expected that neither the quantity nor activity of the floor cleanup water will change as a result of this modification. The SFP demineralizer resin removes soluble radioactive material from the SFP water. These resins are periodically sluiced with water to the spent resin storage tank. The amount of radioactivity on the SFP demineralizer resin may increase slightly due to the additional spent fuel in the pool, but the soluble radioactive material should be retained on the resins. If any radioactive material is transferred from the spent resin to the sluice water, it will be removed by the liquid radwaste system for processing. After processing in the liquid radwaste system, the amount of radioactivity released to the environment as a result of the proposed modification would be negligible.

6.2.4 Occupational Radiation Exposures

We have reviewed the licensee's plans for the removal and disposal of the low density racks and the installation of the high density racks with respect to occupational radiation dose. The occupational dose for the entire operation is estimated by the licensee to be about 25 person-rems. We consider this to be a realistic estimate because it is based on realistic dose rates and occupancy factors for individuals performing a specific job during the pool modification. This operation is expected to be a small fraction of the total annual person-rems from occupational exposure.

We have estimated the increment in onsite occupational dose resulting from the proposed modifications utilizing information supplied by the licensee. Specifically the licensee supplied dose rates in the spent fuel pool area resulting from radionuclide concentrations in the SFP water and from the spent fuel assemblies. The spent fuel assemblies themselves will contribute a negligible amount to dose rates in the pool area because of the depth of water shielding the fuel. Consequently, the occupational radiation dose resulting from the additional spent fuel in the pool represents a negligible impact. Based on the present and projected operations in the spent fuel pool area, we estimate that the proposed modification should add less than one percent to total annual occupational radiation dose at this facility. Thus, we conclude that storing additional fuel in the SFP will not result in any significant increase in doses received by workers.

7.0 Environmental Impact of Postulated Accidents

Although the new high-density racks will accommodate a large inventory of spent fuel, we have determined that the installation and use of the racks will not change the radiological consequences of a postulated fuel handling accident in the SFP area from those values reported in the Indian Point-2 FES dated September 1972. The amount of fuel damaged in the postulated accident remains unchanged.

Additionally, the NRC staff has underway a generic review of load handling operations in the vicinity of spent fuel pools to determine the likelihood of a heavy load impacting fuel in the pool and, if necessary, the radiological consequences of such an event. Because Indian Point-2 will be required (by technical specification) to (1) prohibit loads greater than the nominal weight of a fuel assembly and handling tool to be transported over spent fuel in the SFP, and, (2) prohibit loads of weight greater than the weight of a fuel rack to be moved in the SFP building above the operating deck level, we have concluded that the likelihood of any other load handling accident is sufficiently small, that the proposed modification is acceptable, and no additional restrictions on load handling operations in the vicinity of the SFP are necessary while our review is underway.

8.0 Radiological Impact on Environment

As discussed in Section 6, expansion of the storage capacity of the SFP will not create any significant additional radiological effects. The total additional body dose that might be received by an individual within a 50 mile radius is small compared to the fluctuations found in natural background radiation. The increase in occupational radiation exposure of workers to install the new racks is estimated to be about 25 person rem. This is a small fraction of the total person-rem burden from occupational exposure at the plant.

9.0 Basis and Conclusion for Not Preparing an Environmental Impact Statement

On the basis of the foregoing analysis, it is concluded that there will be no significant environmental impact attributable to the proposed action other than has already been predicted and described in the Commission's FES for Indian Point 2. Having made this conclusion, the Commission has further concluded that no environmental impact statement for the proposed action need be prepared and that a negative declaration to this effect is appropriate.

Dated: January 11, 1982

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-247

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

NOTICE OF ISSUANCE OF AMENDMENT AND NEGATIVE DECLARATION TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 75 to Facility Operating License No. DPR-26, issued to the Consolidated Edison Company of New York, Inc. (the licensee), which revised Technical Specifications for operation of the Indian Point Nuclear Generating Unit No. 2 (the facility) located in Buchanan, Westchester County, New York. The amendment is effective as of the date of issuance.

The amendment revises the Technical Specifications to authorize an increase in the capacity of the spent fuel storage pool at the facility. It also places restrictions on load handling in the spent fuel building.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the Federal Register on May 28, 1980 (45 FR 35948). No request for a hearing or petition for leave to intervene was filed within the prescribed limit following notice of the proposed action.

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