April 19, 1990

Docket No. 50-247

Distribution:
Docket File
NRC/Local PDRs
PDI-1 Rdg
SVarga
BBoger
CVogan
DBrinkman
OGC
DHagan
EJordan
GHill (4)
GBagchi
RJones
JLinville
Wanda Jones
JCalvo
ACRS (10)
GPA/PA
OC/LFMB
JMinns
RPederson
LKopp
FWitt
NWagner
WThompson
CMcCracken
LCunningham
CCheng

Mr. Stephen B. Bram
Vice President, Nuclear Power
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenue
Buchanan, New York 10511

Dear Mr. Bram:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. 72962)

The Commission has issued the enclosed Amendment No. 150 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated June 20, 1989, as supplemented by letters dated August 25, 1989; October 23, 1989; January 19, 1990; January 24, 1990; February 9, 1990; February 23, 1990; and March 5, 1990.

The amendment revises the Technical Specifications to permit reracking of the spent fuel storage pit with high density storage racks containing "Boraflex" as a neutron absorber. The high density storage racks will increase the spent fuel storage pit storage capacity from 980 to 1376 fuel assemblies. This change also increases the fuel enrichment from 4.3 w/o U-235 to 5.0 w/o U-235.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

Original signed by

Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 150 to DPR-26
2. Safety Evaluation

cc: w/enclosures
See next page

| OFC | :PDI-1 | :PDI-1 | :OGC | :PDI-1 | : | : |
|---|---|---|---|---|---|---|
| NAME | :CVogan CV | :DBrinkman DB | : | :RCapra RC | : | : |
| DATE | : 4/16/90 | : 4/16/90 | : 4/17/90 | : 4/19/90 | : | : |

OFFICIAL RECORD COPY
Document Name: ISSUANCE OF AMENDMENT 72962



CP-1
DF01
1/1

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

April 19, 1990

Docket No. 50-247

Mr. Stephen B. Bram
Vice President, Nuclear Power
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenue
Buchanan, New York 10511

Dear Mr. Bram:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. 72962)

The Commission has issued the enclosed Amendment No. 150 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated June 20, 1989, as supplemented by letters dated August 25, 1989; October 23, 1989; January 19, 1990; January 24, 1990; February 9, 1990; February 23, 1990; and March 5, 1990.

The amendment revises the Technical Specifications to permit reracking of the spent fuel storage pit with high density storage racks containing "Boraflex" as a neutron absorber. The high density storage racks will increase the spent fuel storage pit storage capacity from 980 to 1376 fuel assemblies. This change also increases the fuel enrichment from 4.3 w/o U-235 to 5.0 w/o U-235.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

Donald S. Brinkman

Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 150 to DPR-26
2. Safety Evaluation

cc: w/enclosures
See next page

Mr. Stephen B. Bram
Consolidated Edison Company
of New York, Inc.

Indian Point Nuclear Generating
Station 1/2

cc:

Mayor, Village of Buchanan
236 Tate Avenue
Buchanan, New York 10511

Ms. Donna Ross
New York State Energy Office
2 Empire State Plaza
16th Floor
Albany, New York 12223

Mr. Charles W. Jackson
Manager of Nuclear Safety and
Licensing
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenue
Buchanan, New York 10511

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Post Office Box 38
Buchanan, New York 10511

Mr. Brent L. Brandenburg
Assistant General Counsel
Consolidated Edison Company
of New York, Inc.
4 Irving Place - 1822
New York, New York 10003

Mr. Charlie Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, New York 10271

Mr. Peter Kokolakis, Director
Nuclear Licensing
Power Authority of the State
of New York
123 Main Street
White Plains, New York 10601

Mr. Walter Stein
Secretary - NFSC
Consolidated Edison Company
of New York, Inc.
4 Irving Place - 1822
New York, New York 10003

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

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. 150
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 June 20, 1989, as supplemented by letters dated August 25, 1989; October 23, 1989; January 19, 1990; January 24, 1990; February 9, 1990; February 23, 1990; and March 5, 1990; 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.

2. 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-64 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No.150 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 19, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 150

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

| Remove Pages | Insert Pages |
|---|---|
| vii | vii |
| 3.8-2 | 3.8-2* |
| 3.8-3 | 3.8-3* |
| 3.8-4 | 3.8-4 |
| 3.8-5 | 3.8-5 |
| 3.8-6 | 3.8-6 |
| Figure 3.8-1 (deleted) | -- |
| -- | Figure 3.8-2 (added page) |
| -- | Figure 3.8-3 (added page) |
| Table 4.1-2 | Table 4.1-2 |
| 5.4-1 | 5.4-1 |
| 5.4-2 | 5.4-2 |

------------------

* No changes made on these pages. Pages retyped for clarity and consistency with Amendments No. 140 and 148.

LIST OF FIGURES

6. The requirements for RHR pump and heat exchanger operability/operation in Specifications 3.8.A.3 and 3.8.A.4 may be suspended during maintenance, modification, testing, inspection, repair or the performance of core component movement in the vicinity of the reactor pressure vessel hot legs. During operation under the provisions of this specification, an alternate means of decay heat removal shall be available when the required number of RHR pump(s) and heat exchanger(s) are not operable. With no RHR pump(s) and heat exchanger(s) operating, the RCS temperature and the source range detectors shall be monitored hourly.

7. The reactor $T_{avg}$ shall be less than or equal to $140^{o}F$.

8. Specification 3.6.A.1 shall be adhered to for reactor subcriticality and containment integrity.

B. With fuel in the reactor vessel and when:

i) the reactor vessel head is being moved, or

ii) the upper internals are being moved, or

iii) loading and unloading fuel from the reactor, or

iv) heavy loads greater than 2300 lbs (except for installed crane systems) are being moved over the reactor with the reactor vessel head removed,

the following specifications (1) through (12) shall be satisfied:

1. Specification 3.8.A above shall be met.

2. The minimum boron concentration shall be the more restrictive of either $\geq$2000ppm or that which is sufficient to provide a shutdown margin $\geq$5% $\Delta k/k$. The required boron concentration shall be verified by chemical analysis daily.

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

4. No movement of fuel in the reactor shall be made until the reactor has been subcritical for at least 174 hours.

5. A dead-load test shall be successfully performed on the spent fuel pit bridge refueling crane before fuel movement begins. The load assumed by the refueling crane for this event must be equal to or greater than the maximum load to be assumed by the refueling crane during the refueling operation. A thorough visual inspection of the refueling crane shall be made after the dead-load test and prior to fuel handling.

6. The fuel storage building charcoal filtration system must be operating whenever spent fuel movement is taking place within the spent fuel storage areas unless the spent fuel has had a continuous 35-day decay period.

7. Radiation levels in the spent fuel storage area shall be monitored continuously whenever spent fuel movement is taking place in that area.

8. The equipment door, or a closure plate that restricts direct air flow from the containment, and at least one personnel door in the equipment door or closure plate and in the personnel air lock shall be properly closed. In addition, at least one isolation valve shall be operable or locked closed in each line penetrating the containment and which provides a direct path from containment atmosphere to the outside.

9. Radiation levels in containment shall be monitored continuously.

10. A licensed senior reactor operator shall be at the site and designated in charge of the operation whenever changes in core geometry are taking place.

11. The minimum water level above the top of the reactor pressure vessel flange shall be at least 23 feet (El. 92′0″) whenever movement of spent fuel is taking place inside the containment.

12. If any of the conditions specified above cannot be met, suspend all operations under this specification (3.8.B). Suspension of operations shall not preclude completion of movement of the above components to a safe conservative position.

C. The following conditions are applicable to the spent fuel pit any time it contains irradiated fuel:

1. The spent fuel cask shall not be moved over any 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.

2. The spent fuel storage pit water level shall be maintained at an elevation of at least 93′2". In the event the level decreases below this value, all movement of fuel assemblies in the spent fuel pool storage pit and crane operations with loads over spent fuel in the spent fuel pit shall cease and water level shall be restored to within its limit within 4 hours.

D. The following conditions are applicable to the spent fuel pit anytime it contains fuel:

1. The spent fuel storage racks are categorized as either Region I or Region II as specified in Figure 3.8-2. Fuel assemblies to be stored in the spent fuel storage racks are categorized as either Category A, B or C based on burnup and enrichment limits as specified in Figure 3.8-3. The storage of Category A fuel assemblies within the spent fuel storage racks is unrestricted. Category B fuel assemblies shall only be stored in Region I or in a Region II spent fuel rack cell with one cell wall adjacent to a non-fuel area (a non-fuel area is the cask area or the area on the outside of a rack next to a wall). Category C fuel assemblies shall be stored only in Region I. The one exception to this shall be fuel assembly F-65 which shall be stored in Region I or in a Region II spent fuel rack cell with two cell walls adjacent to non-fuel areas.

   In the event any fuel assembly is found to be stored in a configuration other than specified, immediate action shall be initiated to:

   a. Verify the spent fuel storage pit boron concentration meets the requirements of Specification 3.8.D.2, and

   b. Return the stored fuel assembly to the specified configuration.

2. At all times the spent fuel storage pit boron concentration shall be at least 1500 ppm. With the boron concentration less than this value, all fuel movement within the spent fuel storage pit shall cease and immediate action shall be initiated to restore the boron concentration to at least the minimum specified.

3. During operations described in Specification 3.8.B, the spent fuel storage pit boron concentration shall be at least equal to that required in Specification 3.8.B.2. With the boron concentration less than the specified value either:

   a. Isolate the spent fuel storage pit from the refueling cavity, or

   b. Take actions required by Specification 3.8.B.12.

E. Specification 3.0.1 is not applicable to the requirements of Specification 3.8.

Basis

The equipment and general procedures to be utilized during refueling are discussed in the FSAR. Detailed instructions, the above-specified precautions, and the design of the fuel-handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety[1]. Whenever changes are not being made in core geometry, one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The residual heat removal pump is used to maintain a uniform boron concentration.

The shutdown margin requirements will keep the core subcritical. During refueling, the reactor refueling cavity is filled with borated water. The minimum boron concentration of this water is the more restrictive of either 2000 ppm or else sufficient to maintain the reactor subcritical by at least 5% $\Delta k/k$ in the cold shutdown condition with all rods inserted. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. Periodic checks of refueling water boron concentration ensure the proper shutdown margin. The specifications allow the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

In addition to the above safeguards, interlocks are utilized during refueling to ensure safe handling. An excess weight interlock is provided on the lifting hoist to prevent movement of more than one fuel assembly at a time. The spent fuel transfer mechanism can accommodate only one fuel assembly at a time.

The 174 hour decay time following plant shutdown and the 23 feet of water above the top of the reactor vessel flanges are consistent with the assumptions used in the dose calculations for fuel-handling accidents both inside and outside of the containment. The analysis of the fuel handling accident inside of the containment is based on an atmospheric dispersion factor (X/Q) of $5.1 \times 10^{-4}$ sec/$m^3$ and takes no credit for removal of radioactive iodine by charcoal filters. The requirement for the fuel storage building charcoal filtration system to be operating when spent fuel movement is being made provides added assurance that the offsite doses will be within acceptable limits in the event of a fuel-handling accident. The additional month of spent fuel decay time will provide the same assurance that the offsite doses are within acceptable limits and therefore the charcoal filtration system would not be required to be operating.

The spent fuel storage pit water level requirement in Specification 3.8.C.2 provides approximately 24 feet of water above fuel assemblies stored in the spent fuel storage racks.

The fuel enrichment and burnup limits in Specification 3.8.D.1 and the boron requirements in Specification 3.8.D.2 assure the limits assumed in the spent fuel storage safety analysis will not be exceeded.

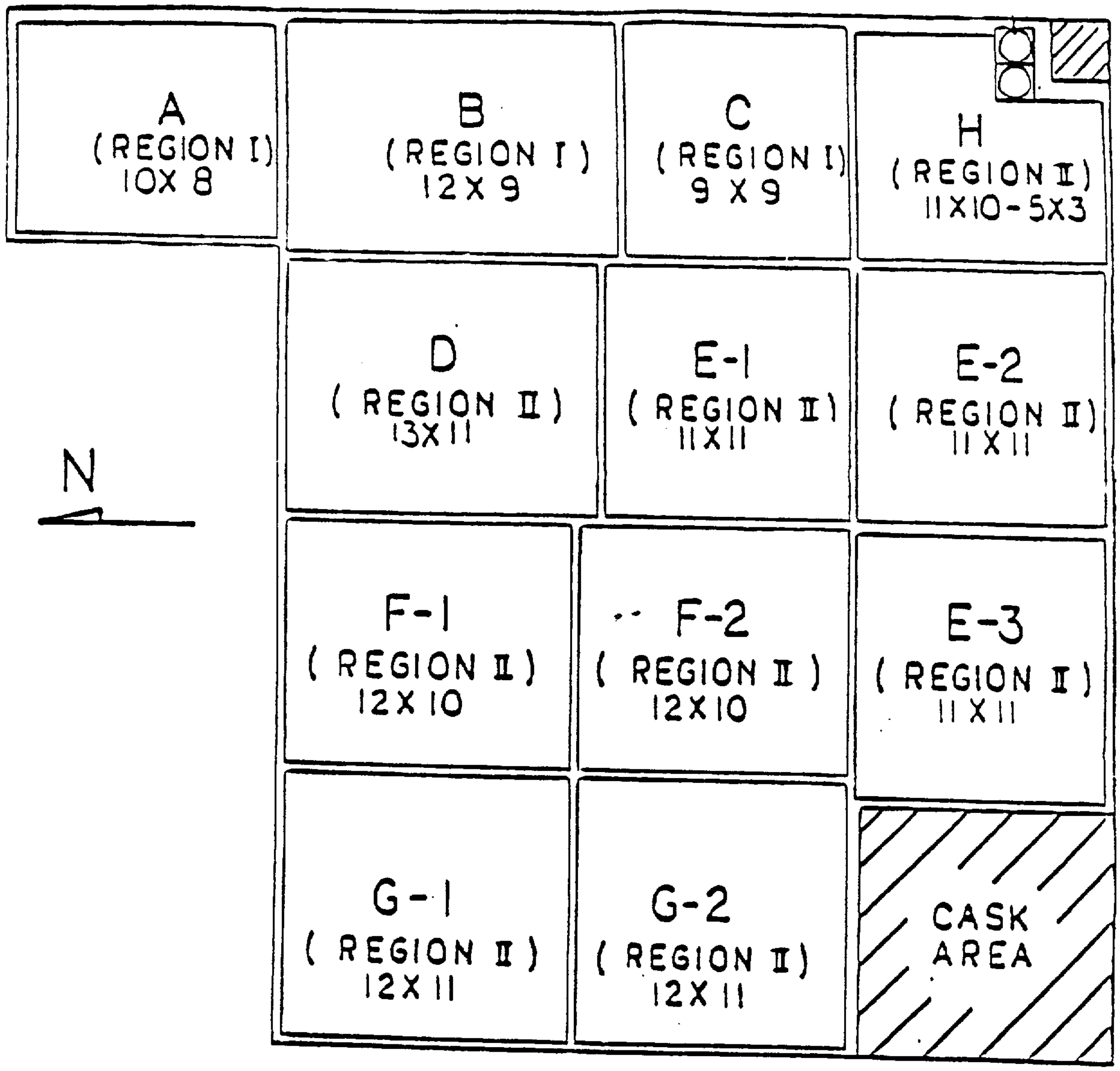
The requirement that at least one RHR pump and heat exchanger be in operation ensures that sufficient cooling capacity is available to maintain reactor coolant temperature below 140°F, and sufficient coolant circulation is maintained through the reactor core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR pumps and heat exchangers operable when there is less than 23 feet of water above the vessel flange ensures that a single failure will not result in a complete loss of residual heat removal capability. With the head removed and at least 23 feet of water above the flange, a large heat sink is available for core cooling, thus allowing adequate time to initiate actions to cool the core in the event of a single failure.

The presence of a licensed senior reactor operator at the site and designated in charge provides qualified supervision of the refueling operation during changes in core geometry.

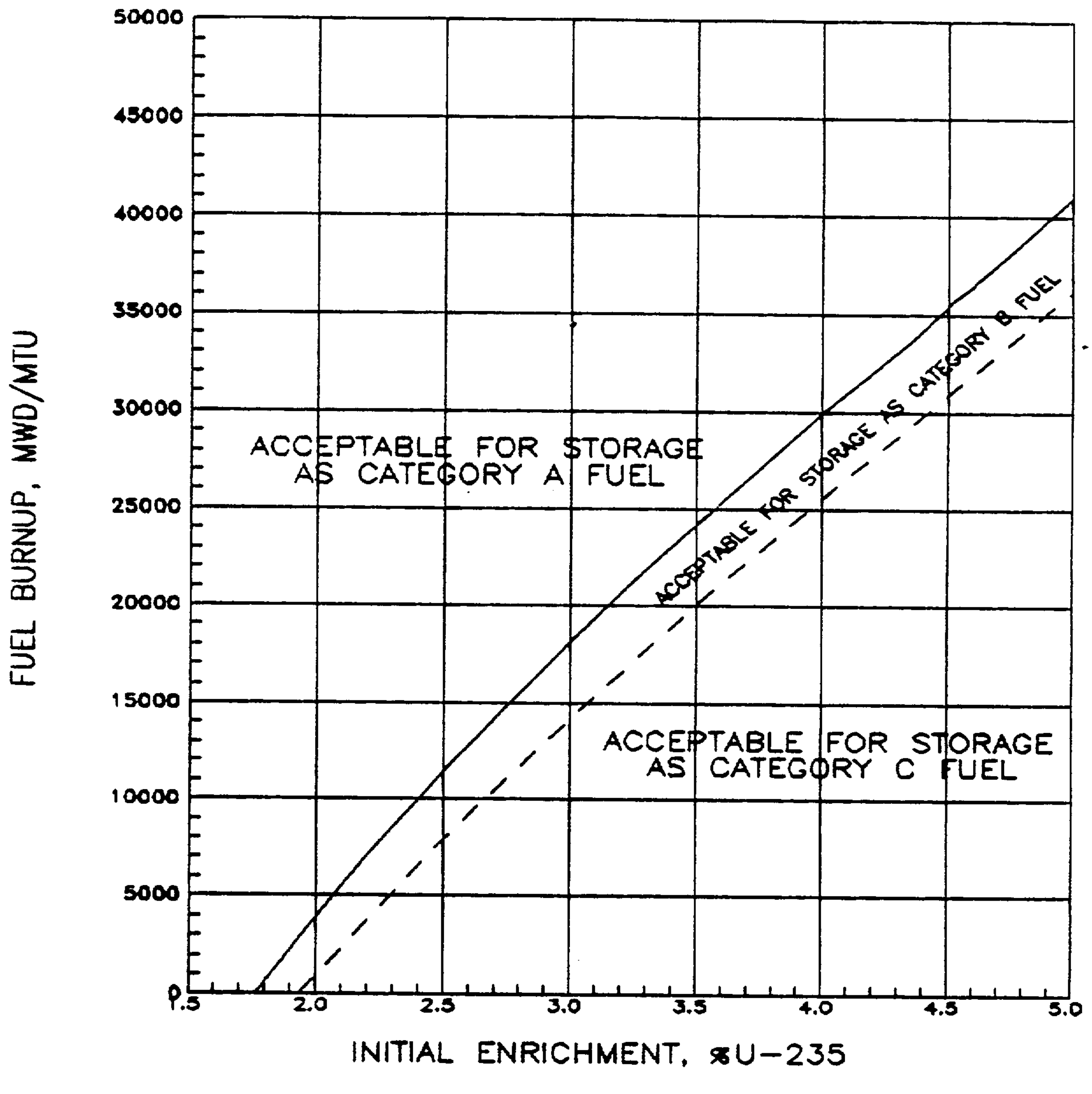
References

(1) FSAR Section 9.5.2




Spent Fuel Storage Rack Layout - IP2 Pool

Figure 3.8-2




Limiting Fuel Burnup
Versus
Initial Enrichment



Figure 3.8-3

## TABLE 4.1-2

## FREQUENCIES FOR SAMPLING TESTS

| | | Check | Frequency | Maximum Time Between Tests |
|---|---|---|---|---|
| 1. | Reactor Coolant Samples | Gross Activity (1)<br>Radiochemical (2)<br>$\bar{E}$ Determination<br>Tritium Activity<br>F, Cl & $O_2$ | 5 days/week (1)<br>Monthly<br>Semi-annually (3)<br>Weekly (1)<br>Weekly | 3 days<br>45 days<br>30 weeks<br>10 days<br>10 days |
| 2. | Reactor Coolant Boron | Boron Concentration | Twice/week | 5 days |
| 3. | Refueling Water Storage Tank Water Sample | Boron Concentration | Monthly | 45 days |
| 4. | Boric Acid Tank | Boron Concentration | Twice/week | 5 days |
| 5. | DELETED | | | |
| 6. | Spray Additive Tank | NaOH Concentration | Monthly | 45 days |
| 7. | Accumulator | Boron Concentration | Monthly | 45 days |
| 8. | Spent Fuel Pit | Boron Concentration | Monthly | 45 days |
| 9. | Secondary Coolant | Iodine-131 | Weekly (4) | 10 days |
| 10. | Containment Iodine-Particulate Monitor or Gas Monitor | Iodine-131 and Particulate Activity or Gross Gaseous Activity | Continuous When Above Cold Shutdown(5) | NA* |

Amendment No. 150

## 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.A. The new fuel storage rack is 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 54.33 grams of U-235 per axial centimeter of fuel assembly.

2.B. The spent fuel storage racks are designed and their loading maintained within the limits of Technical Specification 3.8.D.1, such that $K_{eff} \leq 0.95$ even if unborated water were used to fill the pit and with the fuel loading in the assemblies limited to 56.6 grams U-235 per axial centimeter of fuel assembly.

THIS PAGE IS BEING DELETED

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 150 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 letter dated June 20, 1989, as supplemented by letters dated August 25, 1989; October 23, 1989; January 19, 1990; January 24, 1990; February 9, 1990; February 23, 1990; and March 5, 1990; the Consolidated Edison Company of New York, Inc. (the licensee) requested an amendment to Facility Operating License No. DPR-26 which would add Sections 3.8.C.2, 3.8.D.2, 3.8.D.3, 3.8.E., Figure 3.8-2, Figure 3.8-3 and revise Sections 3.8.D.1, 5.4.2.B, 5.4.3, and Table 4.1-2 of the Technical Specifications for Indian Point Nuclear Generating Unit No. 2. The proposed changes would permit the replacement of the existing spent fuel storage racks with high density racks containing "Boraflex" as a neutron absorber. The changes would also permit the storage of fuel assemblies with enrichments of up to 5.0 weight percent U-235. The new high density racks would increase the storage capacity of the spent fuel pit from its current capacity of 980 fuel assemblies to 1376 fuel assemblies.

The June 20, 1989 request for license amendment was noticed in the Federal Register on August 25, 1989 (54 FR 35421) as a Consideration of Issuance of Amendment to Facility Operating License and Opportunity for Hearing.

The licensee's letters dated August 25, 1989; October 23, 1989; January 19, 1990; January 24, 1990; February 9, 1990; February 23, 1990; and March 5, 1990; provided supplemental information to the original submittal dated June 20, 1989 that did not alter the action as noticed in the Federal Register on August 25, 1989.

The replacement high density storage rack design is a free standing welded array of stainless steel cells arranged in two regions. Region I provides storage for 269 unirradiated or irradiated fuel assemblies with enrichments up to 5.0 weight percent U-235. Region II provides storage for up to 1105 irradiated fuel assemblies which have achieved sufficient burnup. In addition, two special locations are provided to store failed fuel canisters. The nominal center-to-center spacings between storage cells in Region I are 10.545 inches and 10.765 inches, respectively, in north-south and east-west directions. The nominal pitch in Region II is 9.04 inches. Boraflex is used as the neutron absorber material for both regions. In Region I, the Boraflex has a thickness of 0.102 ± 0.007 inch and a nominal boron-10 (B-10) areal density of 0.0324 g/cm$^2$. In Region II, the Boraflex thickness is 0.082 ± 0.007 inch with a nominal B-10 areal density of 0.0260 g/cm$^2$.

9004250215 900419
PDR ADOCK 05000247
P PDC

The following evaluation addresses the staff's review of the June 20, 1989 submittal and the seven supplemental submittals. The Updated Final Safety Analysis Report was also consulted as part of the review.

## 2.0 EVALUATION

### 2.1 Criticality Analysis

#### 2.1.1 Calculation Methods

The analyses of the reactivity effects of fuel storage in Regions I and II were primarily performed with the two-dimensional transport theory computer program CASMO-2E. Auxiliary calculations of the small incremental reactivity effects of eccentric fuel positioning were made with the diffusion theory program PDQ07. Independent verification calculations were made with the AMPX-KENO Monte Carlo computer package for several fuel rack designs. These programs are widely used in the analysis of fuel rack reactivity and have been verified against experiment by several users. The staff concludes that the analysis methods used are acceptable.

The analyses were performed with several assumptions which tend to maximize the rack reactivity. These include:

(1) Unborated pool water at a temperature yielding the highest reactivity;

(2) The absorption effect of the fuel assembly spacer grids is neglected;

(3) Assumptions of infinite extent in lateral and axial directions except for certain peripheral and abnormal assessments;

(4) Reactivity of design basis fuel assembly is higher than Westinghouse HIPAR, LOPAR, or OFA fuel.

The staff concludes that appropriately conservative assumptions were made.

#### 2.1.1 Treatment of Uncertainties

For the nominal storage cell design in Region I, uncertainties due to manufacturing tolerances in boron loading in the Boraflex, cell lattice spacing, stainless steel thickness, and fuel enrichment and density were considered as well as Boraflex shrinkage and eccentric fuel positioning. These uncertainties were appropriately determined at least at the 95 percent probability, 95 percent confidence level. In addition, a method bias and uncertainty were determined from benchmark calculations. The final Region I design when fully loaded with fuel enriched to 5.0 weight percent U-235 resulted in a bias-corrected $K_{eff}$ of 0.9401 when combined with all known uncertainties. This meets the staff's criterion of $k_{eff}$ no greater than 0.95 including all uncertainties at the 95 percent probability, 95 percent confidence level and is, therefore, acceptable.

For Region II, the same uncertainties mentioned above were considered. In addition, an additional factor for uncertainty in the burnup analyses and for the axial burnup distribution was included. Calculations were made for fuel of several different initial enrichments and, at each enrichment, a limiting $k_{eff}$ value of less than 0.95 was established. This method for obtaining the constant reactivity curve for required burnup as a function of enrichment is the standard one used for rack reactivity evaluations and is acceptable. The solid line in Figure 4.1 of the June 20, 1989 submittal indicates the acceptable domain for storage of fuel of various initial enrichments and discharge burnup in Region II. Because of large local neutron leakage, the peripheral cells in Region II facing non-fueled areas are capable of accommodating fuel assemblies of high reactivity. The dashed curve in Figure 4.1 of the June 20, 1989 submittal defines the acceptable domain for storage of fuel in these peripheral cells. In addition, one assembly (No. F-65) has been prematurely discharged and does not satisfy either of the two criteria for storage in Region II. The licensee's calculations have shown that this assembly may be safely stored in one of the four outside corner cells facing two non-fueled areas due to consideration of the neutron leakage in two directions.

Most abnormal storage conditions will not result in an increase in the $K_{eff}$ of the racks. However, it is possible to postulate events, such as an inadvertent misplacement of a fresh fuel assembly either into a Region II storage cell or outside and adjacent to a rack module, which could lead to an increase in pool reactivity. However, for such events credit may be taken for the Technical Specification requirement of at least 1500 ppm of boron in the spent fuel pool water. The reduction in $K_{eff}$ caused by the boron more than offsets the reactivity addition caused by credible accidents.

2.1.3 Summary - Criticality Analysis

Based on the review described above, the staff finds the criticality aspects of the proposed modifications to the Indian Point 2 spent fuel storage racks are acceptable and meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling. The staff concludes that fuel from Indian Point 2 may be safely stored in Region I provided that the U-235 enrichment does not exceed 5.0 weight percent (56.6 grams U-235 per axial centimeter of fuel assembly). Any of these fuel assemblies may also be stored in Region II provided it meets the burnup and enrichment limits specified in Figure 3.8-3 of the Indian Point 2 Technical Specifications.

## 2.2 Material Compatibility and Chemical Stability

### 2.2.1 Discussion

Nuclear reactor plants provide storage facilities for the wet storage of spent fuel assemblies. The safety function of the spent fuel storage pit is to maintain the spent fuel assemblies in a sub-critical array during all credible storage conditions. The staff has reviewed the compatability and chemical stability of the storage rack materials wetted by the pool water in accordance with Section 9.1.2 of the Standard Review Plan (NUREG-0800, July 1981).

The spent fuel storage pit at Indian Point 2 contains air saturated demineralized water which is borated to a minimum of 1500 ppm boron. The pit is lined with stainless steel. The proposed spent fuel racks are constructed of ASME SA240-304 austenitic stainless steel except for leveling screws which are SA-564-630 precipitation hardened stainless steel (to 1100°F). The racks utilize a neutron absorbing material, Boraflex, which is attached to each cell. Boraflex is a silicone-based polymer containing fine particles of boron carbide in a homogeneous, stable matrix.

Boraflex sheets are attached to the sides of the storage rack cell by "picture frame sheathing." The sheathing serves to locate and position the poison sheet accurately but will not subject the Boraflex to surface compression. Clearance is provided to allow unrestrained shrinkage of the Boraflex during irradiation without developing tears or cracks.

The licensee proposed a long-term surveillance program to monitor the performance of the Boraflex in the spent fuel pool environment. Surveillance coupons, representative of materials used in the racks, will be located adjacent to selected racks. The initial surveillance will be implemented after an exposure interval of about two cycles. Future surveillances will be based on results of initial measurements. The examination will include hardness testing, coupon dimensions, neutron absorption measurements, visual examination and photography. If degraded Boraflex is found, corrective actions that would be considered include: cessation of further fuel assembly loading into affected cells until there is reasonable assurance that $k_{eff}$ will be maintained no greater than 0.95, relocation of fuel assemblies from affected to non-affected cells, increasing the fuel pool boric acid concentration, addition of neutron absorbing material to the fuel assembly, or administrative controls on enrichment and/or fuel burnup of fuel assemblies placement in affected cells.

### 2.2.2 Evaluation

The stainless steel in the spent fuel storage pool liners and rack assemblies is compatible with the air saturated borated water and radiation environment of the spent fuel pit. In this environment, corrosion of Type 304L stainless steel is not expected to exceed a rate of $6X10^{-7}$ inch per year

(E. G. Brush and W. L. Pearl, "Corrosion and Corrosion Product Release in Neutral Feedwater," Corrosion, Volume 28, No. 4, page 129, April 1972). The corrosion rate is negligible for even the thinnest stainless steel in the pool liners or rack assemblies. Galvanic attack between the stainless steel in the pool liners or rack assemblies and the Inconel/Zircaloy in the fuel assemblies will not be significant since these materials are protected by passivating oxide films. Boraflex is composed of non-metallic materials and, therefore, will not develop a galvanic potential with the metal components.

Space is available to allow escape of any gas which may be generated from the silicone polymer binder in the Boraflex due to heat and irradiation, thus preventing possible bulging or swelling. Boraflex has undergone extensive testing to determine the effects of gamma irradiation in various environments and to verify its structural integrity and creditability as a neutron absorbing material (Bisco Products, Inc., Technical Report No. NS-1-001, "Irradiation Study of Boraflex Neutron Shielding Materials," August 12, 1981). The evaluation tests have shown that Boraflex is unaffected by the pool water environment and will not be degraded by corrosion. Tests were performed at the University of Michigan, exposing Boraflex in 2000 ppm boron solution to $1.03 \times 10^{11}$ rads of gamma radiation with a concurrent neutron flux of $8.3 \times 10^{13}$ neutrons/cm²/sec. These tests indicate that Boraflex maintains its neutron attenuation capabilities after being subjected to an environment of borated water and gamma and neutron irradiation. However, irradiation caused some loss of flexibility and shrinkage of the Boraflex.

Long-term water soak tests at high temperatures were also conducted, "Boraflex Neutron Shielding Material Product Performance Data," August 25, 1981. The tests show that Boraflex will withstand a temperature of 240°F in a solution of 3000 ppm boron for 251 days without visible distortion or softening. The Boraflex showed no evidence of swelling or loss of ability to maintain a uniform distribution of boron carbide. The spent fuel pool water temperature is normally maintained below 140°F which is well below the 240°F test temperature.

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

Recently, anomalies which range from minor physical changes in color, size, hardness, and brittleness to gap formation of up to four inches wide were observed in Boraflex panels that have been used in the spent fuel pools of four nuclear power plants. The exact mechanism that caused the observed physical degradation of Boraflex have not been confirmed. The staff postulates that gamma radiation from the spent fuel may induce crosslinking of the polymer in the Boraflex, producing shrinkage of the Boraflex material. When crosslinking becomes saturated, scissioning (a process in which bonds between atoms are broken) of the polymer predominates as the accumulated radiation dose increases. Scissioning may produce porosity which allows spent fuel pool water to permeate the Boraflex material, which may cause embrittlement. Gamma radiation from the spent fuel is the most probable cause of the observed physical degradation, such as color change, size, hardness, and brittleness. The staff does not have sufficient information to determine conclusively what caused the gap formation in some Boraflex panels. However, it is conceivable that if two ends of a full-length Boraflex panel are physically restrained, then shrinkage caused by gamma radiation may promote panel tearing and subsequent gap formation.

The staff determined that reasonable assurance exists that the Boraflex panels are not physically restrained in the design of the storage racks at Indian Point 2. The sheathing holds the Boraflex in place on the side of the storage rack cell without pinching, binding, sagging or buckling. Therefore, it is not likely that gaps will form to any significant extent in the Boraflex panels during the design life of spent fuel storage racks. However, minor physical degradation may take place due to irradiation of the Boraflex panels. The Boraflex panels are designed to allow for both shrinkage and edge deterioration and still meet criticality requirements.

The inservice surveillance of the Boraflex panels will monitor the performance of the neutron absorber material in the spent fuel environment. This program is based on EPRI NP-6159, "An Assessment of Boraflex Performance in Spent-Nuclear-Fuel Storage Racks," December 1988. In the unlikely event of gap formation in the Boraflex panels that would lead to loss of neutron absorbing capability, the monitoring program will detect such degraded panels, and the licensee would have sufficient time to perform a criticality evaluation.

### 2.2.3 Summary - Material Compatibility and Chemical Stability

Based on the above discussion, the staff concludes that corrosion of the proposed fuel storage racks due to the spent fuel pit environment should be of little significance during the life of the facility. The staff finds that implementation of the proposed surveillance program and the selection of appropriate materials of construction by the licensee, meet the requirements of 10 CFR Part 50, Appendix A, General Design Criterion 61, regarding the capability to permit appropriate periodic inspection and testing of fuel storage components, and General Design Criterion 62 regarding prevention of criticality by the structural integrity of components and of the boron absorber material and are, therefore, acceptable.

## 2.3 Spent Fuel Pit Load Handling and Cooling

### 2.3.1 Spent Fuel Pit Load Handling

#### 2.3.1.1 Damage to Spent Fuel Assemblies and Offsite Doses

The licensee proposes to move the spent fuel in the present racks in order to remove the existing racks out of the spent fuel pit. The new racks will be brought into the pit with a minimum spacing of 4 feet between a moving rack and the nearest spent fuel assembly being maintained at all times. This separation will minimize the possibility of damage to spent fuel during the reracking process. In addition, the licensee proposed to delay the reracking process until the spent fuel in the present racks has decayed a minimum of 6 months. That was a conservative proposal since in actuality the spent fuel will have decayed more than one year. The reracking is currently proposed to occur during the present fuel cycle (Cycle 10). The reactor was shut down for the last refueling on March 18, 1989.

As can be seen from Table 2.1-1 of NUREG-0612, at least 7500 spent fuel assemblies would need to be damaged in order to cause offsite doses as great as 1/4 of 10 CFR Part 100 limits. Since the IP-2 spent fuel pool will only be capable of storing 1376 assemblies, it is not possible to achieve offsite dosages in excess of staff guidelines by damaging spent fuel stored in the Indian Point 2 pool as long as it has decayed for 6 months or longer. Therefore, the staff finds that the concern of offsite release, resulting from the drop of a new or old spent fuel rack on spent fuel during the reracking process, is satisfactorily addressed.

#### 2.3.1.2 Damage to the Spent Fuel Pit as a Result of a Heavy Load Drop

The licensee noted that the Fuel Storage Building Crane which will be used to remove the old racks and install the new ones has a design capability of 40 tons with a factor of safety of 5. The weight of the heaviest rack to be handled by this crane during the reracking process will be 19.4 tons, effectively increasing the crane factor of safety to 10 during the reracking process. The licensee proposes additional protection by the design and use of special handling devices incorporating a stress design factor in excess of that specified in ANSI N-14.6 and utilizing four lift points instead of the three recommended for a stable lift as an added redundancy. These design features are intended to minimize the potential for a drop of a rack or other heavy load in the spent fuel pit.

In response to a question by the staff, the licensee reported that the drop of a heavy object (a spent fuel cask) onto the spent fuel pit floor had been analyzed in a previous submittal and been found acceptable in a previous evaluation by the staff (Safety Evaluation Report dated 11/16/70). In that Safety Evaluation Report, the staff noted that structural damage would not result in a rapid loss of water with pit makeup being supplied at a rate of 150 gpm. The licensee stated that this bounds the drop of any object during the reracking process since the cask is much heavier than any object being handled during reracking. The staff finds this to be acceptable and finds the issue of a rack drop on the spent fuel pit floor during reracking to be satisfactorily addressed.

2.3.1.3 Potential Criticality

A potential exists for a reduction in safety margin to criticality resulting from a heavy load handling accident in the spent fuel pit. In the Indian Point 2 pit, this potential is considerably reduced by employment of the following:

(1) Rack Handling Device

The licensee described the rack handling device as a box frame with four lift rods. The box, itself, is not designed to carry loads, but is merely intended to place the rods in the proper geometric position for carrying the spent fuel racks into and out of the spent fuel pit. The bottom of each rod is shaped like a bayonet, which allows insertion into a hole in the bottom of the rack. A feature on the top of the lift rod allows the bayonet to be rotated 90° and then locked into place. The rack handling device is then lifted with the rack to any desired location within the operating area of the fuel building crane. Rotation of the rod and locking of the bayonet may be accomplished remotely.

The licensee plans to design and manufacture the rack handling device in accordance with the section 5.1.1 (4) of NUREG-0612. While redundant devices are not being employed, the licensee notes that one lifting device rod could fail without affecting any rack lift during the reracking process.

The licensee also plans to conduct a test of the special lifting device with a test load of 150% of the rated load prior to its use.

(2) Slings for Lifting Racks

The load of a rack and rack handling fixture will be supported with a sling from each corner of the box to the crane rods. Each sling to be used for lifting the rack handling device will comply with the guidelines of section 5.1.6 (1)(b)(ii) of NUREG-0612, i.e., will be capable of carrying twice the load normally required and, hence, will meet the criteria for a single-failure proof sling.

(3) Plan for Rack Movement

In addition, the licensee has outlined the plan for removal of the old racks and replacement with the new ones (shown in Figures 1-6 of the licensee's submittal dated February 9, 1990). These figures show the minimum distance between stored fuel and a rack (empty) that is moved in or out of the spent fuel pit. It is noted that the present Indian Point 2 Technical Specifications permit racks to be moved in the pit, provided they are not moved over spent fuel assemblies.

(4) Fuel Storage Building Crane (FSBC)

It should be noted that a stress safety factor of 10 will be obtained when using the FSBC (see section 2.1.2, above).

In view of the foregoing, the staff finds the possibility of a spent fuel storage rack drop so as to increase Keff to a value in excess of 0.95 to be a highly unlikely event and, thus, considers this issue to be satisfactorily addressed.

2.3.1.4 Safe Load Paths

The licensee stated the following in its June 20, 1989 submittal:

> "There is no equipment which is essential in the safe shutdown of the reactor or employed to mitigate the consequences of an accident located beneath, adjacent to or otherwise within the area of influence of any loads that will be handled during the reracking."

Therefore, the staff considers the issue of safe load paths to be satisfactorily addressed.

2.3.1.5 Procedures

The licensee provided an outline of the procedures to be used during reracking. These included:

(a) Moving the spent fuel assemblies already in the spent fuel pit
(b) Removal of old spent fuel storage racks
(c) Installation of new racks
(d) Replacement of spent fuel assemblies in the new racks

The staff finds this satisfactory.

2.3.1.6 Operator Training

The licensee proposes to qualify operators in the reracking process by a course which includes the following topics:

Crane Operator qualifications/conduct
Communications
O.S.H.A./A.N.S.I. standards
Safety factors and precautions
Rigging methods, precautions and procedures
Rigging hardware
Scaffolding

The licensee stated that the course will include discussions, videotaped presentations and hands on field exercises designed to cover the course topics. The licensee will require the operators to undergo performance demonstrations and a written examination at the end of the course. The licensee intends to qualify operators in accordance with Chapter 2-3, "Qualifications for Operators," of ANSI B30.2.0-1976, "Overhead and Gantry Cranes". Personnel who have been trained in reracking procedures will control the reracking process. The staff finds this acceptable and considers this issue to be satisfactorily addressed.

2.3.1.7 Fuel Storage Building Crane Design

The FSBC was found to meet the intent of the guidelines for cranes used for handling heavy loads in a previous staff safety evaluation and, thus, no further consideration of crane design is necessary.

2.3.2 Spent Fuel Pit Cooling

2.3.2.1 Description of Spent Fuel Pit Cooling System (SFPCS)

The Indian Point 2 SFPCS consists of two pumps and one heat exchanger with one of the two pumps normally on standby. It also contains a filter, demineralizer, associated valves and piping.

The licensee recently submitted a proposal to increase the plant power output from 2758 to 3071.4 MWT. The licensee's proposal was approved by License Amendment No. 148 issued March 7, 1990. Thermal-hydraulic analysis of temperatures within the spent fuel pool have been conducted for spent fuel in the new racks with the new power level as the basis for these calculations.

2.3.2.2 Calculated SFP Coolant Temperatures

The licensee calculated the expected bulk pit coolant temperatures in accordance with applicable staff criteria. For the normal maximum heat load case, (one refueling load discharge and the remainder of the pit full with successive refueling discharges), the pit water temperature was calculated to be 138.4°F. For the abnormal maximum heat load (full core discharge), the coolant temperature was calculated to be 205.2°F. The recommended maximum temperatures for these cases are 140°F and less than 212°F (bulk boiling temperature), respectively. Therefore, the staff finds the fuel pit temperature acceptable and fuel pit cooling has therefore been satisfactorily addressed.

2.3.2.3 Spent Fuel Pit Makeup

The licensee calculated that, in the event spent pit cooling was lost, the amount of makeup water needed to maintain the pit level would be 62 gpm, assuming the temperature of the makeup water was 100°F. The licensee noted that there were three sources of pool makeup water: the primary water storage tank (PWST), the refueling water storage tank (RWST), or the fire protection system. The minimum makeup rate from these sources is 100 gpm. Of these, only the RWST is seismic Category I and borated (@ 2000 ppm boron) with a minimum water content of 345,000 gallons - enough to supply makeup to the SFP for almost 4 days. The PWST contains 165,000 gallons, enough makeup for almost 2 days. The primary tank in the fire protection system contains 1.5 million gallons with a 300,000 gallon backup tank and a virtually inexhaustible supply from the city water system. The staff finds these makeup sources to be acceptable and considers the issue to be satisfactorily addressed.

2.3.3.4 Spent Fuel Cladding Temperatures

The licensee assumed a maximum pit bulk water temperature of 180°F in calculating spent fuel assembly temperatures. The licensee calculated the natural convection flow by determining the thermal/hydraulic balance for a row of hot assemblies and associated down comers, ignoring coolant from other areas. The maximum outlet coolant temperature was determined to be less than 240.5°F (the local boiling point) with a fuel surface temperature approximately 10°F greater. The staff finds this to be acceptable and considers this issue to be satisfactorily addressed.

2.3.3 Technical Specification (TS) Changes

2.3.3.1 TS 3.8.C.2

The licensee proposed to add this TS which requires that the spent fuel storage pit water level be maintained at an elevation of at least 93'2". This level provides approximately 24 feet of water above fuel assemblies stored in the spent fuel storage racks.

The staff finds this to be acceptable since it provides the basis upon which both heat transfer and offsite dose calculations were made.

2.3.3.2 TS 3.8.D

TS 3.8.D.1 specifies the allowable fuel storage patterns permitted in the new racks. Added TS 3.8.D.2 specifies the minimum boron concentration in the spent fuel pool in order to maintain acceptable criticality limits. TS 3.8.D.3 specifies that the spent fuel pool boron concentration shall be at least 1500 ppm in the event fuel is in the reactor vessel and one of the following actions is taking place:

i) the reactor vessel head is being moved, or
ii) the upper internals are being moved, or
iii) loading and unloading fuel from the reactor, or
iv) heavy loads greater than 2300 lbs (except for installed crane systems) are being moved over the reactor with the reactor vessel head removed.

TS 3.8.D.3 replaces TS 5.4.3 which specified that, with fuel in the spent fuel pit, the spent fuel pit boron content must be equivalent to that in the refueling canal and reactor cavity during refueling operations. Note that TS 3.8.D.1 includes Figure 3.8-2 which shows the spent fuel and the classification required for storage in each rack.

2.3.3.3 TS 3.8.E

This new specification makes TS 3.0.1 inapplicable to TS 3.8, thus allowing continued reactor operation in the event any part of TS 3.8 is not met. This condition actually applies only to specifications 3.8.C and D since TS 3.8.A and 3.8.B only apply when either the reactor head is not fully tensioned or some movement is taking place with the reactor head off in which cases power operation has already been discontinued. Thus, the application of TS 3.8.E would permit reactor operation to continue with an inappropriate fuel storage pattern or when the spent fuel pool boron content was less than that required. However, both TS 3.8.C and 3.8.D contain requirements to correct these spent fuel pool maloperations without requiring reactor shutdown. Furthermore, TS 3.8.E complies with the Standard Technical Specifications for Westinghouse PWRs. Therefore, the staff finds TS 3.8.E to be acceptable.

2.3.3.4 TS 5.4.2.B

This TS replaces the present TS 3.8.C.1 with 3.8.D.1 for spent fuel assembly loading patterns in the pit and provides a new limit of 56.6 gm U-235 in lieu of 54.33 gm per axial centimeter of fuel in an assembly.

2.3.3.5 TS 5.4.3

This TS has been deleted and replaced by TS 3.8.D.3, as noted previously.

2.3.4 Summary - Spent Fuel Pit Load Handling and Cooling

Based on its review, the staff concludes that the licensee's plans to provide new racks in the Indian Point 2 spent fuel pit together with accompanying operations and changes in the plant Technical Specifications meet applicable staff criteria and are, therefore, acceptable from the point of view of heavy loads handling and spent fuel pool cooling.

2.4 Structural Analysis

2.4.1 Discussion

The primary areas of review associated with the structural analysis of the proposed change are focused towards assuring the structural integrity of the fuel, fuel cells, rack modules, and the spent fuel pool floor and walls under the postulated loads (Appendix D of SRP 3.8.4) and fuel handling accidents. The major areas of concern and their resolutions are outlined in the following paragraphs, which include fuel rack seismic analysis, multiple rack seismic analysis results, overall evaluation of seismic analysis, thermal analysis, accident analyses and spent fuel pit analysis.

2.4.2 Evaluation - Fuel Handling Building and Spent Fuel Storage Pit

The review of the analysis and design of the spent fuel pit was based on a preliminary meeting with the licensee in April 1989, at the NRC Headquarters, One White Flint North, Rockville, Maryland, to review the licensee's proposed schedule for submitting its licensing application to NRC and subsequent schedules relating to rack removal and installation, and a subsequent meeting with the licensee, its consulting engineer and vendors on October 26, 1989, at the vendor's site.

Detailed drawings of the concrete fuel storage building, and tank liner plates indicating the location of the piping in the spent fuel pit (the only safety-related equipment, except for the storage rack) were reviewed.

The spent fuel pit is designed as a Seismic Category I structure. The structure was reanalyzed with the proposed new racks. The foundation bedrock consists of hard limestone capable of supporting loads up to 50 tons per square foot. The foundation boring logs indicate limestone with unconfined compressive strength of 7810 psi in the vicinity of the spent fuel pit. The structural analysis was carried out using a finite element model of representative sections of the pit. The floor and walls were modeled using shell elements, and the foundation modeled using 3D brick elements. Sections were assumed to be fully loaded with the heaviest racks for structural analysis purposes. In addition to the superimposed loadings, the pit structure was also subjected to the temperature induced loadings. The thermal boundary conditions were conservatively specified as 180 degrees F pool water temperature and 0 degrees F outside ambient. The thermal moments computed by the finite element analyses were combined with those due to mechanical loads. The Finite Element Model consisted of 270 elements and 544 nodes. The contribution of the surrounding rock continuum is modeled using three dimensional solid elements. Load combinations were considered as per Appendix D of SRP 3.8.4. In addition to the specified load combinations, other assumptions in the pit structural analysis produced inherent margins of safety in the computed values. Despite these conservative assumptions, the licensee's calculations resulted in large margins between the factored loads and corresponding design strengths. (Reference Tables 2 and 3 of licensee's January 19, 1990 letter.)

A review of the stress factors resulting from these analyses demonstrates that an adequate design margin exists for the pit walls and basemat.

### 2.4.3 Evaluation - High Density Racks

Licensee's original seismic analysis in Section 6 of the Licensing Report attached to the licensee's June 20, 1989 letter is based on 4% structural damping and Housner's Ground Response Spectra as the basic input. The staff requested the licensee to utilize the damping factors as committed in the FSAR.

All governing loading cases reported in Section 6 of the Licensing Report have been re-run with 1% structural damping by the licensee as per FSAR Table 1.11-1. The responses have increased slightly over previously submitted calculations using 4% damping; however, there is no effect on the rack structural integrity conclusions presented in the licensing submittal.

Statistical independence of the three components of synthetic time histories was established by computing the normalized cross correlation of each pair of time histories (a total of three pairs).

The technical evaluation of the high density racks (HDRs) includes the fuel rack dynamic analysis with a 3-D model and consideration of parameters such as fluid coupling effects, bounding friction coefficients, conservative damping, seismic input motion, and various load cases. Once the rack seismic models are constructed, equations of motion of the system are written and solved using the Dynarack computer program. The Dynarack computer program has been reviewed previously by the staff and its use has been found acceptable.

The analysis resulted in (1) computed displacements and element forces at each instant of time during the earthquake, and (2) stresses at critical rack locations from known element forces at each time step. The limiting value of each stress factor is 1.0 for the OBE condition. For the SSE condition the limit is 2.0 for the rack material and upper part of the support feet and 1.54 for the lower part of the support feet. Maximum stress factors for the rack base and support feet, maximum rack displacements and floor loads were computed. The licensee also performed calculations to evaluate the adequacy of welds at various rack locations. The concern of the potential damage to the fuel resulting from fuel assembly to cell wall impacts led to the licensee's demonstration that the capacity of the fuel assembly to withstand lateral forces is greater than the maximum forces calculated by the licensee's rack analysis. This approach is acceptable.

The results of the licensee's seismic analysis indicated that all stresses in the racks met the corresponding allowables. Impact loads on fuel assemblies would not damage the fuel, and rack displacements would not be large enough to result in impacts with other racks or with the pit walls. The licensee performed additional studies to address the potentially unconservative modeling assumption regarding effective fuel mass, multiple rack effects and fluid coupling considerations. Compared to the minimum spacing (1 1/4 inches) provided between the racks and the racks and the walls, the maximum calculated displacements were small (.1854 inches), assuring no impacts between racks or between the racks and the walls. These results coupled with the conservatisms present in the analyses, and the significant margins provided in the design, demonstrate the adequacy of the fuel racks.

It is quite likely that the racks could move or tip during or after certain events, such as jammed fuel assembly or a seismic event. The licensee has stated that its Abnormal Operating Instruction 28.0.8 entitled "Earthquake Emergency" will be used to assure that any necessary repair procedures for damaged equipment or structures are implemented (see January 19, 1990 submittal at Response to Question V.2.). The staff finds this procedure acceptable.

### 2.4.4 Evaluation - Fuel Handling Accident

Structural analyses and evaluations were performed for the following events: (1) a fuel assembly and associated handling tool (weight approximately 2000 pounds) dropped from 36 inches above a storage location impacting the base of the module and (2) a fuel assembly and associated handling tool dropped from 36 inches above the rack impacting the top of the rack. The staff agrees that these assumptions are conservative since fuel assemblies will not be raised more than 36 inches above the top of racks. With respect to case (1), the results of the analyses showed that by using conservative estimates the baseplate will contain the drop with the possibility of some local baseplate to cell weld damage occurring adjacent to the cell in question. In case (2), some inelastic deformation may occur within top 6 inches of the rack. However, such an occurrence would not damage the fuel in the rack. Hence, this is not a safety concern.

Based on these bounding results, the licensee has demonstrated that the postulated fuel handling accident conditions do not cause unacceptable conditions in the fuel racks. Thus, the structural integrity of the rack and of the stored fuel is assured.

### 2.4.5 Summary - Structural Analysis

Based on the review and evaluation of the licensee's submittal, additional information and analysis provided by the licensee, discussions with the licensee at meetings, and the results of the staff audit at the vendor's facilities, we conclude that the licensee's structural analyses and design of the spent fuel rerack modules, and the spent fuel pool, are adequate to withstand the effects of the required environmental and abnormal loads. Furthermore, the analyses and design are in compliance with the current licensing practice and, therefore, are acceptable as the licensee commits to ensure that the design gaps between the racks/walls are maintained (1) during rack installation, (2) during fuel handling operations, and (3) after a seismic event equivalent to or exceeding OBE.

## 2.5 Radiation Protection and ALARA Considerations

### 2.5.1 Discussion/Evaluation

The NRC staff has reviewed the licensee's proposal to rerack the Indian Point 2 spent fuel storage pit. The licensee's submittals state that the additional occupational radiation exposure for reracking the spent fuel storage pit is estimated to be less than 10 person-rem. Operation of the spent fuel storage pit with an increased storage capacity will not contribute any significant increase in plant occupational exposure.

The existing SFP racks consist of twelve independent free standing modules which can be lifted out of the pool without the use of divers. Similarly the new high density racks are also independent free standing modules and will not need divers to install them in the SFP. In the event that divers are required during this reracking operation, the licensee has specific station procedures in place to ensure the radiation exposure received by the divers is ALARA. Each diver will be equipped with a remote-readout radiation detector, which will be continuously monitored by a health physics technician. Also, each diver will have a calibrated alarming dosimeter. Spent fuel will be relocated to minimize radiation exposure to divers. Radiation surveys in the pool will be performed daily (prior to diving) and whenever fuel is moved. Also, Quality Assurance personnel will independently witness and verify the locations of fuel assemblies whenever fuel is moved.

### 2.5.2 Summary - Radiation Protection and ALARA Considerations

Based on our review of the Indian Point 2 submittals, we conclude that the projected activities and estimated person-rem doses for this project appear reasonable. The licensee intends to take ALARA considerations into account, and to implement reasonable dose-reducing activities. We conclude that the licensee will be able to maintain individual occupational radiation exposures within the applicable limits of 10 CFR Part 20, and maintain doses ALARA, consistent with the guidelines of Regulatory Guide 8.8. Therefore, the proposed radiation protection aspect of the SFP rerack is acceptable.

## 2.6 Design Basis Accidents

### 2.6.1 Discussion

In its application, the licensee evaluated the possible consequences of postulated accidents and included means for their avoidance in the design and operation of the facility, and has provided means for mitigation of their consequences should they occur. The staff independently assessed such so-called design basis accidents (DBAs) and agrees with the licensee that no previously unconsidered DBA would be created by the installation and operation of the reracked spent fuel storage pool.

### 2.6.2 Evaluation

In its safety evaluation for License Amendment No. 86, issued on February 21, 1984, the staff conservatively estimated offsite doses due to exposures to radionuclides released to the atmosphere from a fuel handling accident. This is the staff's scoping DBA for the spent fuel storage pit. The staff concluded that the plant mitigative features would reduce the DBA doses to well below the doses specified in the applicable regulation at 10 CFR Part 100.

Since the applicant intends to utilize higher enrichment fuel, for which higher burnups are intended, the staff reanalyzed the fuel handling DBA for this case. Increased burnup could increase offsite doses from the fuel handling DBA by a factor of 1.2 (NUREG/CR-5009, February 1988). Burnup to 60,000 MWD/T would require the use of fuel initially enriched to about 5.3 weight percent U-235. Thus, we conservatively increased the previously estimated doses by a factor of 1.2. In Table 1, the new and old DBA doses are presented and compared to the guideline doses in 10 CFR Part 100.

### 2.6.3 Summary - Design Basis Accidents

As shown in Table 1, the DBA doses (even with the estimates for higher fuel burnup) are still well within the regulatory guideline values and are, therefore, acceptable.

TABLE 1

RADIOLOGICAL CONSEQUENCES OF
FUEL HANDLING DESIGN BASIS ACCIDENT

| | Exclusion Area | |
|---|---|---|
| | Thyroid | Whole Body |
| Revised Estimates (License Amendment No. 86) | 100 | 0.4 |
| Estimates for Higher Fuel Burnup* | 120 | 0.48 |
| Regulatory Requirement (10 CFR Part 100) | 300 | 25 |

*Factor of 1.2 greater than original estimates

## 3.0 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 April 6, 1990 (55 FR 12969). Based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

## CONCLUSION

The Commission published a Notice of Consideration of Issuance of Amendment to Facility Operating License and Opportunity for Hearing in the Federal Register on August 25, 1989 (54 FR 35421). No requests for hearing were received and the State of New York did not have any comments.

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.

PRINCIPAL CONTRIBUTORS:
J. L. Minns
R. Pedersen
L. Kopp
F. J. Witt
N. Wagner
D. S. Brinkman
W. N. Thompson

DATED: April 19, 1990