April 21, 1986

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. 111 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 November 19, 1985.

The amendment revises the Technical Specifications to permit the use of higher enrichment reload fuel assemblies and stroge of such assemblies both prior and subsequent to their loading in the reactor. The fuel enrichment is revised from 3.5 w/o U-235 to 4.3 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,

Marylee M. Slosson, Project Manager
PWR Project Directorate #3
Division of PWR Licensing-A

Enclosures:
1. Amendment No. 111 to DPR-26
2. Safety Evaluation

cc: w/enclosures
See next page

*SEE PREVIOUS PAGE FOR CONCURRENCES

PAD-3*	PAD-3* MMS	OELD*	D/PAD-3*
C. Vogan	M. Slosson;bs		S. Varga
04/9/86	04/9/86	04/11/86	04/21/86

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C. Vogan	M. Slosson;bs	M. Karman	S. Varga
04/9/86	04/9/86	04/11/86	04/21/86

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

AM07 BOOK

DO NOT REMOVE

April 21, 1986

Docket No. 50-247

Posted
Amdt. 111
to DPR-26

Mr. John D. O'Toole
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Marylee M Slosson

Marylee M. Slosson, Project Manager
PWR Project Directorate #3
Division of PWR Licensing-A

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2. Safety Evaluation

cc: w/enclosures
See next page

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

CONSOLIDATED EDISION COMPANY OF NEW YORK, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 111
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 November 19, 1985, 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-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. 111 , 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 to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Director
PWR Project Directorate #3
Division of PWR Licensing-A

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 21, 1986

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 111 TO FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
vii	vii
3.8-4	3.8-4
3.8-5	3.8-5
3.8-6	3.8-6
-	3.8-7
-	Figure 3.8-1
5.3-1	5.3-1
5.4-1	5.4-1
-	5.4-2

LIST OF FIGURES

B. If any of the specified limiting conditions for refueling is not met, refueling shall cease until the specified limits are met, and no operations which may increase the reactivity of the core shall be made.

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

1. Fuel assemblies to be stored in the spent fuel pit are categorized as either Category A, B or C based on burnup and enrichment limits as specified in Figure 3.8-1. The storage of Category A fuel assemblies within the pit is unrestricted. Category B fuel assemblies shall only be loaded into a spent fuel rack cell whose adjacent cells on all four sides either contain non-fuel materials or Category A fuel assemblies. The storage of Category C fuel assemblies within the pit is unrestricted except that they cannot be loaded adjacent to Category B fuel assemblies.

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 (2 above) and neutron flux provides immediate indication of an unsafe condition. The residual heat pump is used to maintain a uniform boron concentration.

The shutdown margin indicated in Part 5 will keep the core subcritical, even if all control rods were withdrawn from the core. During refueling, the reactor refueling cavity is filled with approximately 300,000 gallons of water from the refueling water storage tank with a boron concentration of 2000 ppm. The minimum boron concentration of this water at 1615 ppm boron is sufficient to maintain the reactor subcritical by at least 10% $\Delta k/k$ in cold shutdown with all rods inserted, and will also maintain the core subcritical even if no control rods were inserted into the reactor.[2] Periodic checks of refueling water boron concentration ensure the proper shutdown margin. Part 6 allows 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 then one fuel assembly at a time. The spent fuel transfer mechanism can accommodate only one fuel assembly at a time.

The 131 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 5.1×10^{-4} sec/m^{3} and takes no credit for removal of radioactive iodine by charcoal filters. The requirement for the spent 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 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.

The fuel enrichment and burnup limits in Specification 3.8.C.1 assures the limits assumed in the spent fuel safety analyses will not be exceeded. Within this specification adjacent location means those four locations directly contacting the four sides (faces) of a fuel assembly but excludes those four locations which contact the four corners of a fuel assembly.

<u>References</u>

(1) FSAR - Section 9.5.2

(2) Fuel Densification - Indian Point Nuclear Generating Station Unit No. 2, dated January 1973, Table 3.3.

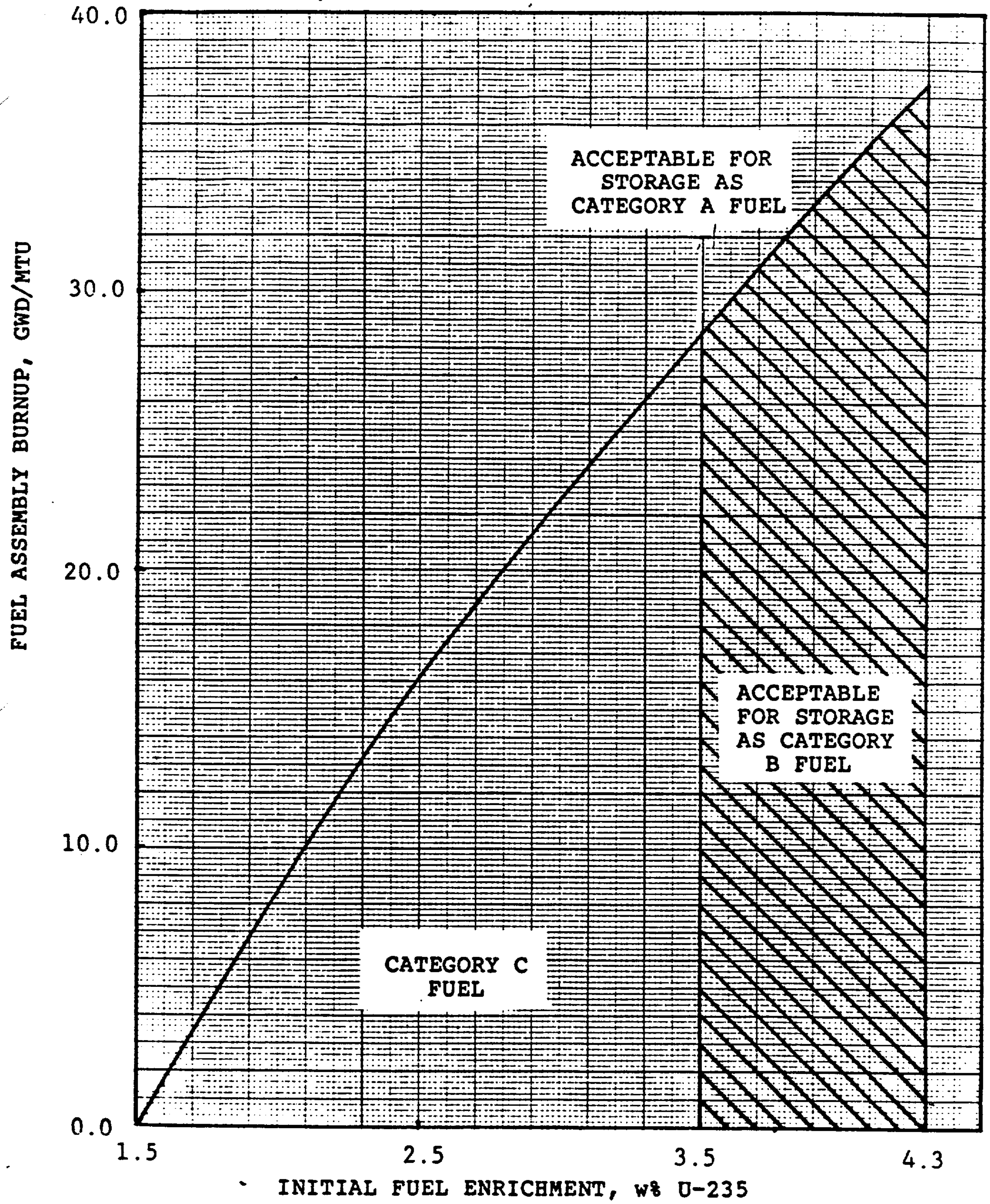

FIGURE 3.8-1

LIMITING FUEL BURNUP VERSUS INITIAL ENRICHMENT

CATEGORY A - AREA ALONG THE CURVE AND ABOVE
CATEGORY B - AREA BELOW THE CURVE AND 3.5 < w% U-235 ≤ 4.3
CATEGORY C - AREA BELOW THE CURVE AND w% U-235 ≤ 3.5

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. Deleted

3. The enrichment of reload fuel will be no more than 4.3 weight per cent U-235 and will be stored in accordance with Technical Specification 5.4.

4. Deleted

5. There are 53 control rods in the reactor core. The control rods contain 142 inch lengths of silver-indium-cadmium alloy clad with the stainless steel.(5)

B. Reactor Coolant System

1. The design of the reactor coolant system complies with the code requirements.(6) Design values for system temperature and pressure are 650^{o}F and 2485 psig, respectively.

2. All piping, components and supporting structures of the reactor coolant system are designed to Class I requirements, and have been designed to withstand the maximum potential seismic ground acceleration, 0.15g, acting in the horizontal and 0.10g acting in the vertical planes simultaneously with no loss of function.

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.C.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 54.33 grams 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. 111 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 November 19, 1985 (Reference 1) the Consolidated Edison Company (ConEd) of New York, licensee for the Indian Point Unit 2 reactor, requested a change to the Technical Specifications to permit the use of higher enrichment reload fuel assemblies and storage of such assemblies both prior and subsequent to their loading in the reactor. Additional information was submitted at the staff's request on January 30, 1986 (Reference 2) and on March 27, 1986 (Reference 3). Specifically it is requested that Technical Specifications 5.3.A.3, 5.4.2 and 3.8.C.1 be modified to change the fuel and spent fuel pool assembly enrichment limit from 3.5 to 4.3 w/o U-235. The method, utilized for the analyses used the CASMO-2E (Reference 4) and PDQ-07 (Reference 5) codes. However, the previous analyses (Reference 6) were performed using KENO-IV (Reference 7) with AMPX cross sections (Reference 8). The new method is benchmarked with respect to the (previous) KENO-IV calculations. The additional information dealt with the CASMO-PDQ uncertainties, procedures for the checkerboard loading scheme and potential future use of Gd as a fuel poison. The Reactor Systems Branch reviewed the submitted information and our evaluation follows.

2.0 EVALUATION

A discussion of each one of the proposed changes follows.

2.1 Technical Specification 3.8.C.1

The new specification distinguishes three categories of fuel assemblies (A, B and C) as a function of burnup vs enrichment. The three categories are shown in Fig. 3.8-1 of the Technical Specifications, which is reproduced on the next page. Technical Specification 3.8.C.1 proposes to load category B assemblies in the spent fuel pool in a checkerboard pattern, neighboring (on the sides) only with category A assemblies. The category C assembly storage within the pool is unrestricted except that they cannot be loaded adjacent to category B fuel assemblies. This arrangement with the burnups and enrichment specified in Fig. 3.8-1 assures that the limits assumed in the safety analyses of the spent fuel pool will not be exceeded. The substantive change is the increase of the initial fuel enrichment from 3.5 to 4.3 w/o U-235. The checkerboard loading pattern will maintain $k_{eff} \leq .95$ as provided in section 9.1.2 of the Standard Review Plan. The k_{eff} estimate is discussed in paragraph 2.4 below.

2.2 Technical Specification 5.3.A.3

The proposed change in this specification will allow the maximum initial fuel enrichment to increase from 3.5 to 4.3 w/o U-235.

2.3 Technical Specification 5.4.2

The proposed change in this specification allows the licensee to raise the amount of U-235 fuel/cm height of fuel assembly to 54.33 grams/cm (from 43.9 grams/cm) which corresponds to the 4.3 w/o U-235 enrichment.

2.4 Proposed Keff Evaluation

The arrangement of the Indian Point 2 spent fuel storage pool consists of twelve storage racks, having a total of 980 storage locations. The structural material is 304 stainless steel and each fuel assembly position has a borated steel plate (1.1 w/o boron) on each side. In such racks, the initial calculations showed that 3.5 w/o U-235 fuel at zero exposure in a infinite x-y array would yield $k_{eff} \leq 0.95$. No hardware changes have been proposed for the storage racks or the spent fuel pool.

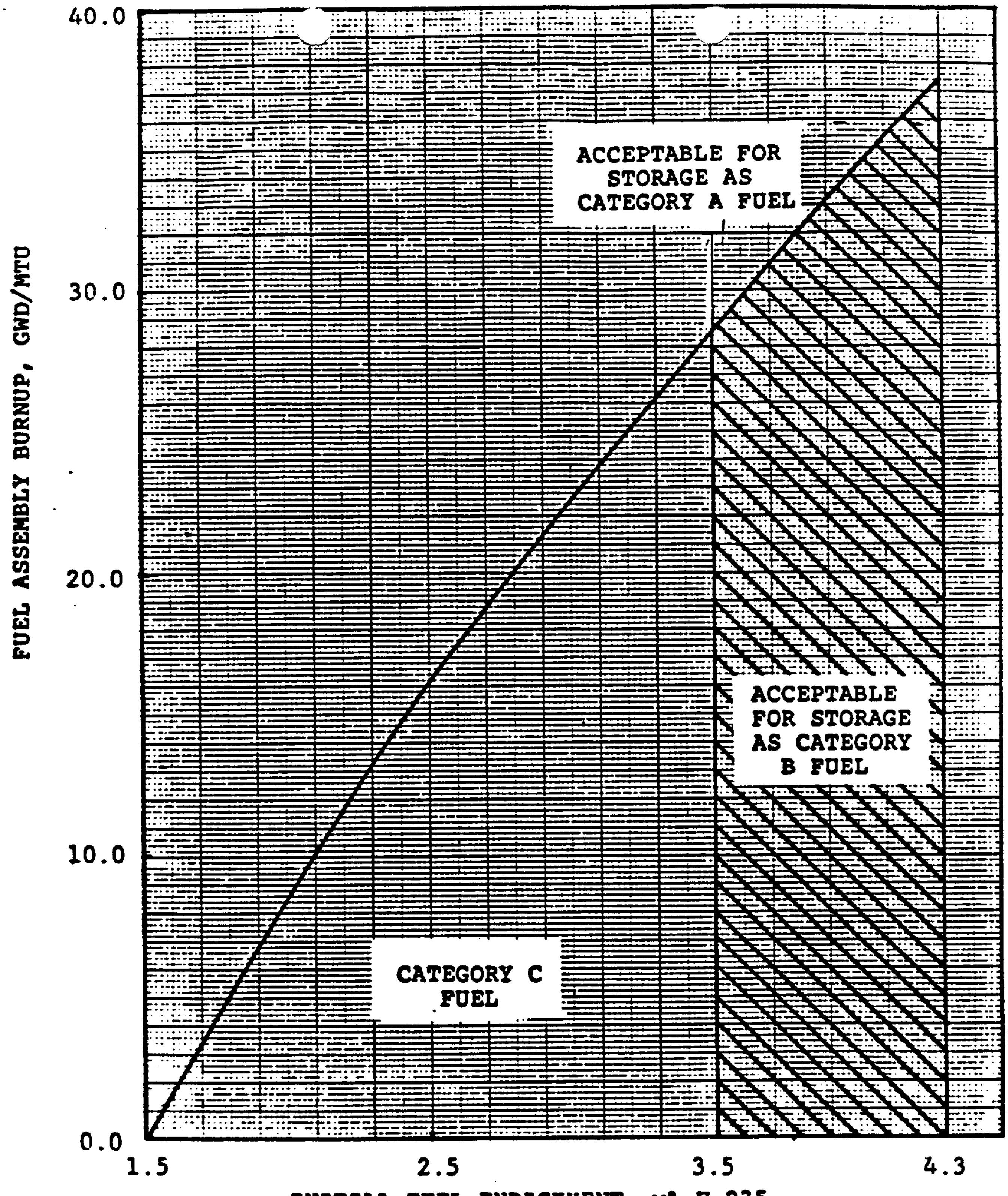

FIGURE 3.8-1

LIMITING FUEL BURNUP VERSUS INITIAL ENRICHMENT

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CATEGORY B - AREA BELOW THE CURVE AND 3.5 < w% U-235 ≤ 4.3
CATEGORY C - AREA BELOW THE CURVE AND w% U-235 ≤ 3.5

Amendment No.

1. The soluble boron in the pool water is ignored

2. The temperature of the pool water is assumed to be 68°F

3. The neutron absorption in the fuel assembly grid spacer material is ignored.

4. No credit is taken for burnable poison

5. The calculation assumes infinite array in all dimensions, hence, ignores neutron leakage

6. The 15x15 fuel assembly is assumed to be the Westinghouse LOPAR design (Zircalloy guide tubes) which is more reactive than the HIPAR fuel design.

The methodology, in this calculation used the CASMO-2E and PDQ-7 codes (References 4 and 5) to assess enrichment vs depletion. The CASMO-2E solution for the Indian Point 2 pool was used to benchmark the calculation against the original (reference) calculation and to provide macroscopic cross sections for non fuel regions for use in the PDQ-7 code. (References 9 and 10). Depletion dependent macroscopic cross sections as a function of initial enrichment, were used which in turn were used to derive burnup as a function of initial enrichment for a constant k_{eff}. CASMO fuel rack and fuel assembly models were developed, which generated the necessary cross sections for a PDQ-7 rack and fuel assembly model. The last was applied to determine the checkerboard arrangement of 4.3 w/o U-235 and depleted assembly loadings. This process yielded Fig 3.8-1 shown previously.

Using the 3.5 w/o U-235 loading for all locations as a reference condition, a correction factor of Δk_{eff} = .012 was estimated for the CASMO-2E model, which yielded a k_{eff} = .936. This compared well with the .933±.06 of the KENO-IV initial calculation.

The PDQ-7 fuel rack configuration calculations with 4.3 w/o U-235 and depleted

The following assumptions were made in the estimation of the k_{eff} in the new configuration which includes the 4.3 w/o U-235 enrichment.

assemblies was carried out on a 4 1/4 assembly configuration with reflecting boundary conditions. Two of the 1/4 assemblies were 4.3 w/o U-235 and the other two were depleted. The k_{eff} value in Figure 3.8-1 at 1.5 w/o U-235 and zero burnup is .906. Then this k_{eff} was held constant for increasing burnup and enrichment. In this manner the category A fuel is determined to be above the k_{eff} = .906 line. A justification of the biases and uncertianties of the calculation was submitted in the additional information (Reference 3). The uncertainties and biases were conservative and, hence, acceptable.

3.0 SUMMARY

We have reviewed the information submitted by the Consolidated Edison Co. in support of their request for modifications of the Technical Specifications to increase the initial enrichment from 3.5 to 4.3 w/o U-235. The high initial enrichment assemblies will be loaded in a checkerboard fashion with low enrichment high burnup assemblies. Procedures for the handling of the assemblies in the pool have been established. Conservative calculations have shown that the k_{eff} of the pool will at most equal .936 which is well below the .95 value identified as acceptable in the standard review plan. Therefore, the storage of fuel assemblies with initial enrichment from 3.5 to 4.3 w/o U-235 is acceptable. Likewise, the proposed changes to the Technical Specifications which allow storage of high initial enrichment fuel are acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

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.

Dated: April 21, 1986

PRINCIPAL CONTRIBUTOR:

L. Lois