ember 19, 1986

Docket No. 50-286

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Mr. John C. Brons
Senior Vice President - Nuclear Generation
Power Authority of the State of New York
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Dear Mr. Brons:

The Commission has issued the enclosed Amendment No. 70 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated June 13, 1986.

The amendment revises the Technical Specifications to permit storage of fuel having enrichment up to 4.3 weight percent U-235 in the fresh and spent fuel racks.

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,

Joseph D. Neighbors, Senior Project Manager
PWR Project Directorate #3
Division of PWR Licensing-A, NRR

Enclosures:
1. Amendment No. 70 to DPR-64
2. Safety Evaluation

cc: w/enclosures
See next page

PD#3	PD#3	OGC	PD#3
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Indian Point Nuclear Generating
Unit No. 3

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

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 70
License No. DPR-64

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

   A. The application for amendment by Power Authority of the State of New York (the licensee) dated June 13, 1986, 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. 70 , 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

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance:
November 19, 1986

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

ATTACHMENT TO LICENSE AMENDMENT NO. 70

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

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

6. The fuel storage building emergency ventialtion system shall be operable whenever irradiated fuel is being handled within the fuel storage building. The emergency ventilation system may be inoperable when irradiated fuel is in the fuel storage building, provided irradiated fuel is not being handled and neither the spent fuel cask nor the cask crane are moved over the spent fuel pit during the periods of inoperability.

7. Fuel assemblies to be stored in the spent fuel pit can be categorized as either Category 1, 2 or 3 based on burnup and initial enrichment as specified in Figure 3.8-1. Category 2 fuel shall be loaded into the spent fuel pool storage locations in a checkerboard fashion with the intermediate storage locations containing Category 1 fuel, non-fuel materials or left empty. Unless restricted by the above, Category 1 or 3 fuel can be stored in any location in the spent fuel pool.

Basis

The equipment and general procedures to be utilized during refueling, fuel handling, and storage 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, fuel handling, reactor maintenance or storage 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 indicated 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 342,000 gallons of water from the refueling water storage tank with a boron concentration of 2000 ppm. A shutdown margin of 10% $\Delta K/K$ in the cold condition with all rods inserted will also maintain the core subcritical even if no control rods were inserted into the reactor. [2] Periodic checks of refueling water boron concentration and residual heat removal pump operation insure the proper shutdown margin. The requirement for direct communications 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 than one fuel assembly at a time. The spent fuel transfer mechanism can accommodate only one fuel assembly at a time.

The 120-hour decay time following the subcritical condition and the 23 feet of water above the top of the reactor pressure vessel flange is consistent with the assumptions used in the dose calculation for the fuel-handling accident.

The waiting time of 400 hours required following plant shutdown before unloading more than one region of fuel from the reactor assures that the maximum pool water temperature will be within design objectives as stated in the FSAR.

The requirement for the fuel storage building emergency ventilation system to be operable is established in accordance with standard testing requirements to assure that the system will function to reduce the offsite dose to within acceptable limits in the event of a fuel-handling accident. The system is actuated upon receipt of a signal from the area high activity alarm or by a manually-operated switch. The system is tested prior to fuel handling and is in a standby basis.

When fuel in the reactor is moved before the reactor has been subcritical for at least 365 hours, the limitations on the containment vent and purge system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorbers prior to discharge to the atmosphere.

The limit to have at least two means of decay heat removal operable ensures that a single failure of the operating RHR System will not result in a total loss of decay heat removal capability. With the reactor head removed and 23 feet of water above the vessel flange, a large heat sink is available for core cooling. Thus, in the event of a single component failure, adequate time is provided to initiate diverse methods to cool the core.

The minimum spent fuel pit boron concentration and the restriction of the movement of the spent fuel cask over irradiated fuel were specified in order to minimize the consequences of an unlikely sideways cask drop.

Fuel assemblies whose initial enrichment is greater than 3.5 w/o U-235 but less than or equal to 4.3 w/o can be stored in the spent fuel pool providing they are placed in a checkerboard array with fuel whose initial enrichment and burnup are sufficient to ensure that K_{eff} is less than 0.95 with no soluble boron present. This is ensured by categorizing the fuel whose initial enrichment is greater than 3.5 w/o U-235 but less than or equal to 4.3 w/o and whose burnup is below the curve of Figure 3.8-1 as Category 2. This fuel can be stored by checkerboarding with Category 1 fuel which is defined as fuel whose initial enrichment and burnup place it on or above and to the left of the curve in Figure 3.8-1. Category 3 fuel which is less than or equal 3.5 w/o U-235 and below the curve of Figure 3.8-1 cannot be used in the checkerboard with Category 2 fuel. Any Category 1 or 3 fuel can continue to be stored on a repeating x-y array with other non-Category 2 fuel. For the purpose of storing Category 2 fuel, non-fuel material or empty locations can be utilized in place of Category 1 fuel.

When the spent fuel cask is being placed in or removed from its position in the spent fuel pit, mechanical stops incorporated in the bridge rails make it impossible for the bridge of the crane to travel further north than a point directly over the spot reserved for the cask in the pit. Thus, it will be possible to handle the spent fuel cask with the 40-ton hook and to move new fuel to the new fuel elevator with a 5-ton hook, but it will be impossible to carry any object over the spent fuel storage area with either the 40 or 5-ton hook of the fuel storage building crane.

Dead load test and visual inpsection of the hoists and cranes before handling irradiated fuel provide assurance that the hoists or cranes are capable of proper operation.

References

(1) FSAR - Section 9.5.2

(2) FSAR - Table 3.2.1-1

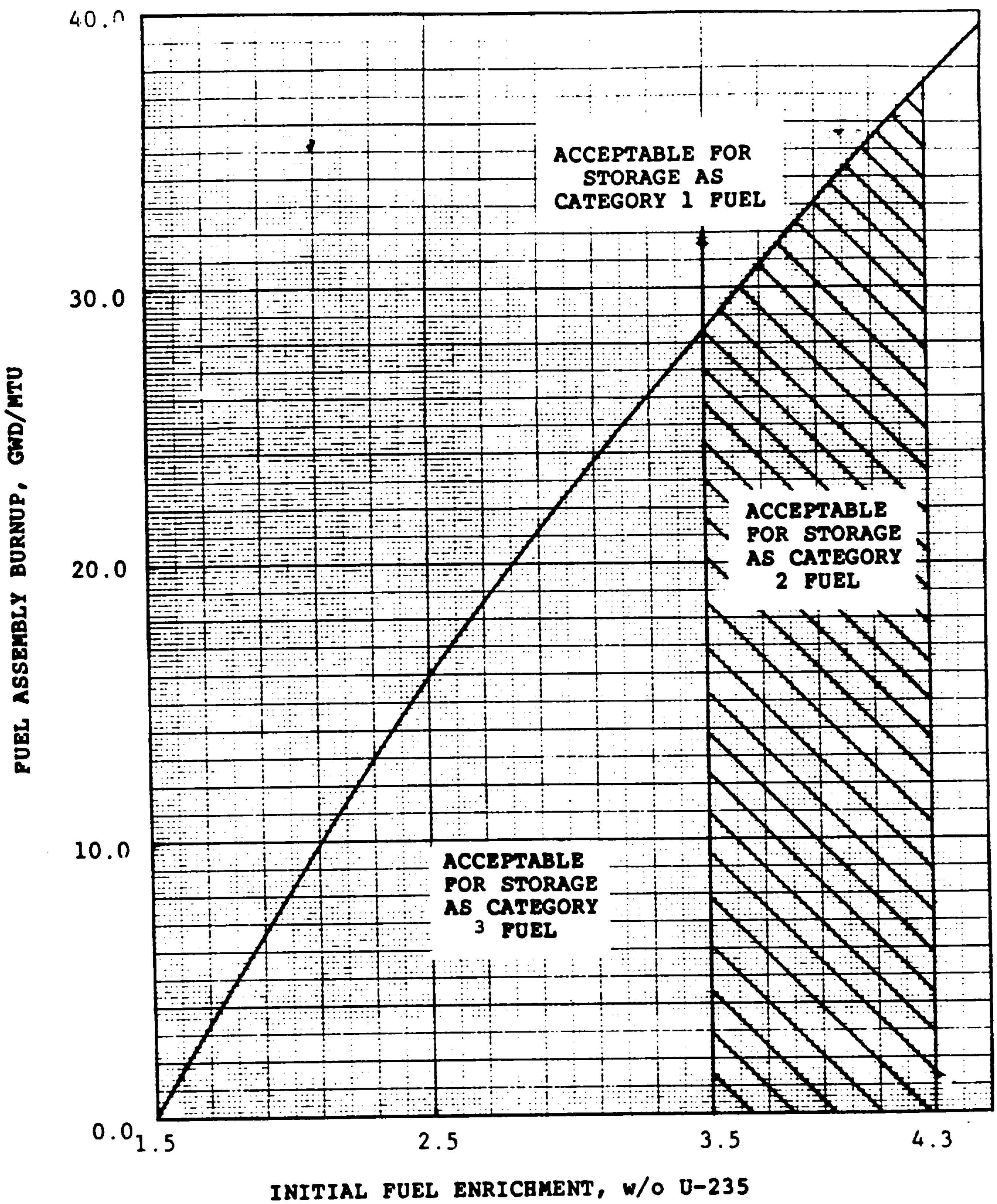

FIGURE 3.8-1

LIMITING FUEL BURNUP VERSUS INITIAL ENRICHMENT

CATEGORY 1 - AREA ALONG THE CURVE AND ABOVE
CATEGORY 2 - AREA BELOW THE CURVE AND $3.5 < \varepsilon \leq 4.3$ w/o U-235
CATEGORY 3 - AREA BELOW THE CURVE AND $\varepsilon \leq 3.5$ w/o U-235

5.3 REACTOR

Applicability

Applies to the reactor core, and reactor coolant system.

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 was a nominal 2.8 weight percent of U-235. Three fuel enrichments were used in the initial core. The highest enrichment was a nominal 3.3 weight percent of U-235.(2)

3. Reload fuel will be similar in design to the initial core. The enrichment of reload fuel will be no more than 4.3 weight percent of U-235.

4. Burnable poison rods were incorporated in the initial core. There were 1434 poison rods in the form of 8, 9, 12, 16, and 20-rod clusters, which are located in vacant rod cluster control guide tubes.(3) The burnable poison rods consist of borosilicate glass clad with stainless steel.(4)
   Burnable poison rods of an approved design may be used in reload cores for reactivity and/or power distribution control.

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 spent fuel storage racks are designed to assure $K_{eff} \leq 0.95$ if the assemblies are inserted in accordance with Technical Specification 3.8. The capacity of the spent fuel pit is 840 assemblies. The new fuel storage racks are designed to assure $K_{eff} \leq 0.95$ and their capacity is 72 assemblies.

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

4. Fuel assemblies that contain more than 54.6 grams of uranium -235, or equivalvent, per axial centimeter of fuel assembly shall not be stored in the spent fuel pit.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 70 TO FACILITY OPERATING LICENSE NO. DPR-64

POWER AUTHORITY OF THE STATE OF NEW YORK

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

INTRODUCTION

By letter dated June 13, 1986, The Power Authority of the State of New York (the licensee) requested a change to Facility Operating License DPR-64 which would change Sections 3.8, 5.3 and 5.4 of the Technical Specifications for Indian Point Nuclear Generating Unit No. 3 (IP3). The proposed changes would permit the reload of fuel assemblies with enrichments up to 4.3 weight percent U-235 and the storage of such assemblies prior to and subsequent to loading in the reactor.

DISCUSSION AND EVALUATION

Two types of storage racks of slightly different design are currently present in the spent fuel storage area at IP3. Each type is licensed to store fresh fuel having enrichments up to 3.5 weight percent U-235 without any restrictions on the location of the fuel in the racks. The storage of fuel with greater enrichment requires either that it have a burnup greater than some value which is dependent on initial enrichment or that it be stored in a "checkerboard" pattern in the racks. The submittal provides analyses to show that checkerboard storage is safe and presents the results of the analysis of the required burnup as a function of initial enrichment.

The analysis of the reactivity effects of the increase in fuel enrichment was performed as a perturbation on the original analysis. That analysis was performed with the KENO IV Monte Carlo code and is reported in Reference 1. The perturbation analyses presented in the present submittal were performed with the CASMO-2E and PDQ codes. Both codes are widely used in the analysis of fuel rack criticality and have been verified against experiment by several users. The CASMO/PDQ combination used for the present analysis was verified by comparison to KENO calculation of several fuel rack designs. This is an acceptable verification technique for well established codes. The verification calculation resulted in the conclusion that CASMO tends to overestimate the reactivity of the racks relative to KENO. That is the usual result from such comparisons. We conclude that the analysis methods used for the IP3 are acceptable.

Both the original and the present 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 grid spacers is neglected.

3. The Westinghouse Optimized Fuel Assembly (OFA) design is neglected.

4. Assumptions of infinite extent in lateral and axial directions.

We conclude that appropriately conservative assumptions are made.

The CASMO and PDQ codes are used to obtain a curve of required burnup as a function of initial enrichment for unlimited storage in the racks of fuel having enrichment between 3.5 and 4.3 weight percent U-235. These codes are also used to obtain the required burnup as a function of initial enrichment for fuel that may be used to fill the checkerboard when fresh fuel with enrichment of 4.3 weight percent U-235 or lower is stored in the racks. These codes are widely used to do burnup calculations for operating reactors and their use for obtaining the isotopic concentration as a function of burnup is acceptable.

The method used for obtaining the constant reactivity curve for required burnup as a function of enrichment is the standard one used for rack criticality evaluations and is acceptable. The application of this method to the evaluation of checkerboard loading is also acceptable. The required burnup curve is obtained by first evaluating the rack multiplication factor for a checkerboard array consisting of fresh 4.3 weight percent and 1.5 weight percent U-235 fuel in alternating (checkerboard) locations. The value obtained for the multiplication factor was 0.901 for the Type A racks and 0.906 for the Type B racks. Next burned fuel having an initial enrichment greater than 1.5% weight percent was substituted for the 1.5 weight percent fuel and the burnup of that fuel was varied until the same multiplication factor was obtained. This process was repeated for several initial enrichment values up to 4.3 weight percent U-235. A curve of required enrichment as a function of burnup was then constructed and is included in the proposed Technical Specifications.

The multiplication factor values cited above are nominal values - i.e. they do not include any uncertainties since the present analyses are perturbations on the original evaluation of the rack criticality. Most of the uncertainties are the same as those in the original analysis. Additional uncertainties due to treatment of burnup and bias in the CASMO-PDQ calculation are included. The slight positive value of the latter uncertainty is conservatively ignored. A value of 0.02 is assigned to the burnup uncertainty. This is consistent with the usual value for this quantity and is acceptable. The final values for the multiplication factors are 0.932 for the Type A racks and 0.937 for the Type B racks at the 95% probability, 95% confidence level.

A separate calculation was performed for the case of the fresh 4.3 weight percent fuel loaded into the racks without any fuel in the other half of the checkerboard array. As expected the multiplication factor was less than with burned fuel in these locations (< 0.90). The chief motivation for the calculation was the investigation of the effect of pool temperature on the multiplication factor. The results showed that the maximum reactivity occurs at 68°F.

The introduction of the checkerboard loading pattern means that misloading 4.3 weight percent fuel in any location must be considered. The limiting case is the loading of such fuel throughout the racks. In such misloading events credit may be taken for the borated water in the pool and the multiplication factor is well below the value for normal storage. The licensee calculated the multiplication factor for both unborated water and water containing 500 ppm boron. The results were 0.96 for unborated water and 0.89 for 500 ppm. Thus, even if loss of boron in the water were postulated along with the full misloading of 4.3 weight percent fuel, the fuel in the racks would remain subcritical.

The results of other accidents are not changed from the original analysis. We conclude that the treatment of accidents is acceptable.

The licensee has provided an analysis of the effective multiplication factor of the fresh fuel storage racks as a function of moderator density. The calculations were performed with the KENO code which was used and approved for the original analysis of the spent fuel racks. Its use for the fresh fuel racks is also acceptable. The results of the analyses showed that the maximum value of effective multiplication factor occurred at full density water and was less than 0.95 including all uncertainties. This meets our criterion for effective multiplication factors and is acceptable.

The licensee has proposed changes to Sections 3.8, 5.3 and 5.4 of the IP3 Technical Specifications. Section 3.8 has been amended to include the curve of required burnup as a function of initial enrichment and the restrictions on the storage of fuel with enrichment between 3.5 and 4.3 weight percent U-235. We have reviewed the proposed changes and conclude that they are consistent with the analysis provided and are acceptable. The changes to Sections 5.3 and 5.4 are merely to change the maximum permitted enrichment from 3.5 to 4.3 weight percent U-235 and are acceptable.

Based on our review, which is described above, we conclude that fuel assemblies having initial enrichments up to 4.3 weight percent U-235 may be safely stored in the fresh and spent fuel racks if the requirements of the Technical Specifications are met. This conclusion is based on the following:

1. Analyses were performed with well established codes that were properly verified.

2. Conservative input assumptions were used.

3. The analysis results meet our criteria for acceptance with respect to effective multiplication factor.

4. The consequences of the limiting accident are acceptable.

5. The proposed Technical Specifications are consistent with the analyses provided.

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: November 19, 1986

PRINCIPAL CONTRIBUTOR:

W. Brooks

REFERENCES:

1. Letter, G. T. Berry to E. G. Case, "Application for Amendment to Operating License, Spent Fuel Pool Modification Description and Safety Analysis", Docket No. 50-286, September 1, 1977.