

September 19, 1990

Docket No. 50-286

Mr. John C. Brons
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

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Dear Mr. Brons:

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT 3 (TAC NO. 77201)

The Commission has issued the enclosed Amendment No. 104 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 July 26, 1990.

The amendment revises the Technical Specifications to incorporate a cycle-specific change regarding the substitution of two failed fuel rods, located in assembly T53 at the core center, with two stainless steel rods.

Also included in this amendment is a correction to Technical Specification page 5.3-2 which incorporates text previously approved by Amendment No. 86 but inadvertently deleted by Amendment No. 101.

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:

Joseph D. Neighbors, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:

1. Amendment No. 104 to DPR-64
2. Safety Evaluation

cc: w/enclosures
See next page

PDI-1
CVogan *cu*
9-5-90

PDI-1 *NAB*
DNeighbors:rsc
9/5/90

OGC *JRH*
J. Hill
9/7/90

PDI *Roe*
RACapra
9/19/90

DOCUMENT NAME: AMENDMENT IP3/77201

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

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Sincerely,

A handwritten signature in cursive script, reading "Joseph D. Neighbors".

Joseph D. Neighbors, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:

1. Amendment No. 104 to DPR-64
2. Safety Evaluation

cc: w/enclosures
See next page

Mr. John C. Brons
Power Authority of the State
of New York

Indian Point 3 Nuclear Power Plant

cc:

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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. 104
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 July 26, 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:

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P FDC

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No.104 , 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 and shall be implemented within 30 days.

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: September 19, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 104.

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

Remove Pages

5.3-1

5.3-2

Insert Pages

5.3-1

5.3-2

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⁾ except during Cycle 8 operation. For Cycle 8 operation only, fuel assembly T53 will contain two stainless steel filler rods in place of two fuel rods.
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 4.5 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.

5.3-1

Amendment No. 67, 70, 90, 104

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⁾
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.
3. The nominal liquid volume of the reactor coolant system, at rated operating conditions and with 0% equivalent steam generator tube plugging, is 11,522 cubic feet.

Basis

The DNBR for Cycle 8 reconstituted fuel assembly T53 will be conservatively determined by assuming the stainless steel replacement rods are operating at the highest power in the reconstituted fuel assembly.

References

- (1) FSAR Section 3.2.2
- (2) FSAR Section 3.2.1
- (3) FSAR Section 3.2.1
- (4) FSAR Section 3.2.3
- (5) FSAR Sections 3.2.1 & 3.2.3
- (6) FSAR Table 4.1-9



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 104 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 July 26, 1990, the Power Authority of the State of New York (the licensee) submitted an application for amendment to the Technical Specifications for the Indian Point Nuclear Generating Unit No. 3. The changes involve the use of two stainless steel dummy (filler) rods replacing two fuel rods in the fuel assembly T53 at the core center. However, in order to justify the use of stainless steel dummy rods in the core center, cycle-specific reload analysis is required. The licensee has committed to provide such an analysis. Our evaluation follows.

EVALUATION

The dummy rods (Zircaloy-4 or stainless steel rods) were originally used in fuel assemblies to replace those fuel rods damaged by the baffle jetting problem in the Westinghouse reactors. The concept was extended further to replace failed rods during reconstitution of fuel assemblies in other locations. However, in order to satisfy generic fuel design criteria as described in the SRP, the dummy rods require thermal-hydraulic analyses to demonstrate that inclusion of the dummy rods in fuel assemblies with the specific configurations and core locations chosen for a specific fuel cycle is acceptable with respect to the overall fuel performance and safety-significant conclusions.

The licensee indicated that the dummy rods will be analyzed by assuming that dummy rods operate at power levels equal to the highest power in any of the fueled rods in the reconstituted assembly. The licensee stated that this results in a conservative analysis with less margin than actually exists to the Departure from Nucleate Boiling Ratio (DNBR) acceptance limit in the reconstituted assembly. The staff agrees that this analysis procedure should be sufficiently conservative to offset uncertainties associated with application of the approved DNBR correlation to reconstituted fuel assemblies which have fuel rod configurations slightly different than those represented in the DNBR test data base. However, core wide analyses will result in a non-conservative calculation with erroneous redistribution of flow from the reconstituted fuel assemblies to other assemblies in the core. This effect should be small and is

probably negligible because only one reconstituted fuel assembly is involved. We, therefore, accept the licensee's approach of cycle-specific reload analysis for dummy rods in the reconstituted fuel assembly.

As for seismic and Loss of Coolant Accident (LOCA) loading conditions, the two dummy rods have outside diameters identical to the fuel rod diameter and the rod length is also the same. The grid strength will remain unchanged since the dummy rods will provide the same support in the grid cells as the fuel rods. For the proposed reconstitution with only two dummy rods, the change in mass and stiffness of the fuel assembly will be insignificant. There will be negligible effects on fuel assembly dynamic properties, such as fuel fundamental frequency. Thus, the load carrying capability of the fuel assembly and grid spacers is not affected under the seismic and LOCA design loading conditions for the reconstituted fuel. The staff concludes that this assessment is reasonable and acceptable.

The licensee stated that the reload that contains reconstituted assemblies will be evaluated using approved methods described in WCAP-9273A, "Westinghouse Reload Safety Evaluation Methodology," dated July 1985. The effect of the actual reconstitution on core performance parameters, peaking factors, core average linear heat rate, and LOCA-related analyses will be evaluated to ensure that the existing safety criteria and design limits and the original fuel assembly design criteria are satisfied. The staff finds that this approach is acceptable since the analysis methods have previously been approved except for the DNBR evaluation for the reconstituted assemblies. A method for the latter evaluation, assuming that the dummy rods are operating at the highest power in the reconstituted assembly, is approved for Indian Point 3 by incorporation in the Technical Specification Basis.

TECHNICAL SPECIFICATION CHANGES

(1) Section 5.3.A.1 Reactor Core

In Section 5.3.A.1 the fuel assembly T53 may consist of 202 fuel rods clad with Zircaloy-4, and two stainless steel dummy rods. This use of the two dummy rods is approved only during Cycle 8 operation and only if justified by cycle-specific analysis. The staff concurs in this change to the Technical Specification.

(2) Section 5.3 Basis

As basis for the change, the licensee states that the DNBR for the reconstituted assembly T53 will be conservatively determined by assuming that the two dummy rods are operating at the highest power in the reconstituted fuel assembly previously described. The staff approves this method of determination.

SUMMARY

The staff has reviewed the licensee's submittal of Technical Specification changes for Indian Point 3 and the proposed approach to safety analyses to assure that fuel assembly design changes will not result in failure to meet the pertinent design safety criteria. We conclude that the proposed Technical Specification revisions are acceptable and that the required cycle-specific

evaluation approach, including DNBR evaluation of the reconstituted fuel assembly as described in the proposed Technical Specification Basis, is acceptable when the use of dummy rods is limited to two rods in the fuel assembly T53 in the core center.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to 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 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: September 19, 1990

PRINCIPAL CONTRIBUTOR:

Larry Phillips, Reactor Systems Branch