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

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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