ATTACHMENT I TO IPN-90-033

PROPOSED TECHNICAL SPECIFICATION CHANGES

RELATED TO

ALTERNATIVE REQUIREMENTS FOR FUEL ASSEMBLIES

IN ACCORDANCE WITH GENERIC LETTER 90-02

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

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 with each fuel assembly nominally containing 204 fuel rods. [1] Substitution of Zircaloy-4 or stainless steel filler rods or open water channels for fuel rods may be made in fuel assemblies if justified by cycle-specific reload analyses using an NRC-approved methodology. Should more than 30 rods in the core or 10 rods in any assembly be replaced per refueling, a special report describing the number of rods replaced shall be submitted to the Commission pursuant to Specification 6.9.2 within 30 days after cycle startup.

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. 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, 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 feet.

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

ANNUAL REPORTS

6.9.1.5 A report of specific activity analysis results in which the primary coolant exceeded the limits of Specification 3.1.D. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Data providing the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator-Region 1 within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification;

a. Sealed source leakage on excess of limits (Specification 3.9)

b. Inoperable Seismic Monitoring Instrumentation (Specification 4.10)

c. Seismic event analysis (Specification 4.10)

d. Inoperable plant vent sampling, main steam line radiation monitoring or effluent monitoring capability (Table 3.5-4, items 5, 6 and 7)

e. The complete results of the steam generator tube inservice inspection (Specification 4.9.C)

f. Inoperable fire protection and detection equipment (Specification 3.14)

g. Release of radioactive effluents in excess of limits (Appendix B Specifications 2.3, 2.4, 2.5, 2.6)

h. Inoperable containment high-range radiation monitors (Table 3.5-5, Item 24)

SPECIAL REPORTS (con't)

i. Radioactive environmental sampling results in excess of reporting levels (Appendix B Specification 2.7, 2.8, 2.9)

j. Operation of Overpressure Protection System (Specification 3.1.A.8.c.)

k. Replacement of more than 30 rods in the core or 10 rods in any assembly per refueling (Specification 5.3.A.1).

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspection, repair and replacements of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of changes made to Operating Procedures.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all source material of record.

i. Records of reactor tests and experiments.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records of any drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

ATTACHMENT II TO IPN-90-033

SAFETY EVALUATION

RELATED TO

ALTERNATIVE REQUIREMENTS FOR FUEL ASSEMBLIES

IN ACCORDANCE WITH GENERIC LETTER 90-02

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

Section I - Description of Changes

The proposed change to Indian Point 3 Technical Specification 5.3.A.1 permits the replacement of fuel rods with Zircaloy-4 or stainless steel filler rods, or with open water channels, if justified by cycle-specific reload analyses. Additionally, a special report will be required if more than 30 rods in the core or 10 rods in any assembly are replaced per refueling. The proposed changes are in accordance with the guidance provided by Generic Letter 90-02, "Alternative Requirements for Fuel Assemblies in the Design Features Section of Technical Specifications."

Section II - Evaluation of Changes

Existing Specification 5.3.A.1 states that each assembly contains 204 fuel rods. The proposed change to Specification 5.3.A.1 will provide for the flexibility to deviate from the nominal number of fuel rods per assembly without the need to request future amendments to the Technical Specifications. This reduces the burden of processing changes for both the NRC and the Authority. Additionally, the change will permit the timely removal of fuel rods that are found to be leaking during a refueling outage or are determined to be probable sources of future leakage. This will provide for reductions in future occupational radiation exposure and plant radiological releases.

The replacement of fuel rods with filler rods or open water channels would be justified by a cycle-specific reload analysis using an NRC approved methodology to ensure that the existing safety criteria and design limits are met.

In accordance with the Generic Letter, a special report shall be submitted to the Commission if more than 30 rods in the core or 10 rods in any assembly are replaced per refueling. The report shall state the number of rods replaced per assembly. This requirement is included in the proposed changes to Specifications 5.3.A.1 and 6.9.2.

Section III - No Significant Hazards Evaluation

Consistent with the requirements of 10 CFR 50.92, the enclosed application is judged to involve no significant hazards based on the following information:

(1) Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response:

The proposed changes regarding the number of fuel rods in a fuel assembly will not adversely affect plant system operations, functions or setpoints. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The replacement of failed fuel rods with filler rods or open water channels would be justified by a cycle-specific reload analysis. The reconstituted assembly will be evaluated using an NRC approved methodology. The reload analysis will evaluate the effect of the actual reconstitution on core performance parameters, peaking factors, and core average linear heat rate effects to ensure that the existing safety criteria and design limits are met, and original fuel assembly design criteria are satisfied.

(2) Does the proposed license amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response:

The proposed changes regarding the number of fuel rods in a fuel assembly will not adversely affect plant system operations, functions or setpoints. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The replacement of failed fuel rods with filler rods or open water channels would be justified by a cycle-specific reload analysis. The reconstituted assembly will be evaluated using an NRC approved methodology. The reload analysis will evaluate the effect of the actual reconstitution on core performance parameters, peaking factors, and core average linear heat rate effects to ensure that the existing safety criteria and design limits are met, and original fuel assembly design criteria are satisfied.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

Response:

The proposed changes regarding the number of fuel rods in a fuel assembly will not adversely affect plant system operations, functions or setpoints. The proposed change does not involve a significant reduction in a margin of safety. The replacement of failed fuel rods with filler rods or open water channels would be justified by a cycle-specific reload analysis. The reconstituted assembly will be evaluated using an NRC approved methodology. The reload analysis will evaluate the effect of the actual reconstitution on core performance parameters, peaking factors, and core average linear heat rate effects to ensure that the existing safety criteria and design limits are met, and original fuel assembly design criteria are satisfied.

Section IV - Impact of Change

This change will not adversely impact the following:

ALARA Program
Security and Fire Protection Programs
Emergency Plan
FSAR or SER Conclusions
Overall Plant Operations and the Environment

Section V - Conclusions

The incorporation of this change: a) will not increase the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report; b) will not increase the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report; c) will not reduce the margin of safety as defined in the bases for any Technical Specification; d) does not constitute an unreviewed safety question; and e) involves no significant hazards considerations as defined in 10 CFR 50.92.

Section VI - References

a) IP-3 FSAR
b) IP-3 SER