ATTACHMENT I TO IPN-92-025

PROPOSED TECHNICAL SPECIFICATION CHANGE

RELATED TO

RECONSTITUTED FUEL ASSEMBLIES

NEW YORK POWER AUTHORITY
INDIAN POINT 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 or ZIRLO™ 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 9 and Cycle 10 operation. For Cycle 9 and Cycle 10 operation only, fuel assemblies W51 and W06 will each contain one stainless steel filler rod in place of a fuel rod.

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.

Amendment No. ~~61~~, ~~70~~, ~~90~~, ~~104~~, ~~117~~,

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 Cycles 9 and 10 reconstituted fuel assemblies W51 and W06 will be conservatively determined by assuming the stainless steel replacement rods are operating at the highest power in the reconstituted fuel assemblies.

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

ATTACHMENT II TO IPN-92-025

SAFETY EVALUATION

RELATED TO

RECONSTITUTED FUEL ASSEMBLIES

TECHNICAL SPECIFICATION CHANGE

NEW YORK POWER AUTHORITY
INDIAN POINT NUCLEAR POWER PLANT
DOCKET NO. 50-286
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Section I - Description of Change

This application for amendment to the Indian Point 3 Technical Specifications proposes to change the description of the reactor fuel assemblies given in Specification 5.3.A.1 to allow (for cycles 9 and 10 only) the substitution of a stainless steel filler rod in place of a fuel rod in fuel assemblies W51 and W06. This proposed change also deletes the allowance for two stainless steel filler rods in one fuel assembly during cycle 8, because cycle 8 has ended.

Section II - Evaluation of Change

During the current cycle 8/9 refueling outage, the Authority noticed damage (a separation or space) to a corner of the lower grid of fuel assembly W51, and damage (a separated brazed joint) to an upper grid corner of assembly W06. This damage leads to the concern that there is a potential (in the next two operating cycles) for vibration and fuel clad fretting due to the grids rubbing against the corner fuel rod. In order to prevent possible damage to the corner fuel rod in assemblies W51 and W06 during the next two operating cycles, it is prudent to remove the corner fuel rods and replace them with stainless steel filler rods.

On May 4, 1991, while defueling the reactor, damaged grid straps were noted on two fuel assemblies (including assembly W51). Fuel offload was completed on May 6. At this time, the exact nature and extent of the grid strap damage was not known. Ultrasonic testing (UT) of fuel assemblies was started on May 8, and was completed on May 11. After UT, the assemblies were visually inspected. Visual inspections were completed on May 13. Since the fuel inspections, the Authority has been evaluating alternative courses of action. On Thursday, May 14, the Authority decided to use a stainless steel rod in place of a fuel rod in assembly W51. The fuel rod in question was satisfactorily tested (UT), but the possibility of fretting in the area of the damaged grid led to the decision to use a stainless steel rod. The proposed Technical Specification change necessary to use the stainless steel rod was reviewed and approved by the onsite and offsite review committees on Monday, May 18, and sent to the Nuclear Regulatory Commission by letter dated May 19, 1992.

On Friday, May 22, while attempting to repair a previously noted raised section of the upper grid strap of assembly W06, the brazed joint in that upper grid was noticed to be separated. Prior to this, the Authority believed the raised grid section could be repaired, and use of a stainless steel rod would not be necessary. Because of the separated brazed joint, the Authority decided to use a stainless steel rod in place of the corner fuel rod in assembly W06, because of the potential for fretting that was described above. The proposed Technical Specification change necessary to use a stainless steel rod in each of the two fuel assemblies (W51 and W06) was reviewed and approved by the onsite and offsite review committees on Friday, May 22.

This proposed Technical Specification change is required prior to core reload during the present cycle 8/9 refueling outage. **The Authority requests that this application be approved on an emergency basis, in accordance with 10 CFR 50.91(a)(5), prior to May 29, 1992. An emergency situation exists because core reload is scheduled to begin on June 1, 1992, and failure to act in a timely way will prevent resumption of plant operation on the scheduled date. Because of the sequence of events listed above, the Authority could not have avoided the need for this emergency Technical Specification change request.**

Existing Technical Specification 5.3.A.1 states that each assembly contains 204 fuel rods, except for a stainless steel filler rod allowance for cycle 8. For cycles 9 and 10, the Authority proposes to use reconstituted fuel assemblies W51 and W06, each containing 203 fuel rods and 1 stainless steel rod. The Technical Specification allowance for two stainless steel filler rods during cycle 8 is being deleted, because cycle 8 has ended.

The Authority has a reasonable assurance that these substitutions can safely be made based on past industry experience with stainless steel filler rods that have performed acceptably. The IP3 cycle 8 core contained two stainless steel filler rods that were evaluated as having no effect on the fuel assembly structural integrity, fuel assembly dynamic properties, control rod worths, core peaking factors, or peak power levels. The acceptability of replacing the corner rod in fuel assemblies W51 and W06 with stainless steel rods will be justified as part of the cycle specific reload evaluation process using a Nuclear Regulatory Commission (NRC) approved methodology to confirm that all existing safety criteria and design limits are met, prior to plant startup with the reconstituted fuel assemblies (W51 and W06) loaded in the reactor core.

As part of the cycle specific Reload Safety Evaluation (RSE) process to be performed by Westinghouse Electric Corp. for the Authority, the impact of the reconstituted assemblies on the Departure from Nucleate Boiling Ratio (DNBR) will be evaluated. Westinghouse will determine the DNBR for the reconstituted assemblies by assuming the filler rods are operating at the highest power in the reconstituted fuel assemblies. Using this extremely conservative assumption, the predicted DNBR for the reconstituted assemblies will be shown to satisfy the minimum DNBR acceptance limit. This approach is consistent with the methodology Westinghouse uses to evaluate reloads, as described in the NRC approved topical report WCAP-9273A (Reference 3). This approach is identical to that used to evaluate the effect of stainless steel filler rods on the DNBR for IP3 cycle 8, and approved by the NRC by the issuance of IP3 Technical Specification Amendment No. 104. The results of the DNBR evaluation will be documented as part of the Indian Point 3 RSE process for cycles 9 and 10.

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 any accident previously evaluated?

Response:

The proposed change does not involve a significant increase in the probability or consequences of any accident previously analyzed. The acceptability of replacing fuel rods with stainless steel filler rods will be justified as part of the cycle specific reload evaluation process using an NRC approved methodology to confirm that all existing safety criteria and design limits are met. The reload evaluation process will address 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.

As part of the cycle specific Reload Safety Evaluation (RSE) process to be performed by Westinghouse Electric Corp. for the Authority, the impact of the reconstituted assemblies on the DNBR will be evaluated. Westinghouse will determine the DNBR for the reconstituted assemblies by assuming the filler rods are operating at the highest power in the reconstituted fuel assemblies. Using this extremely conservative assumption, the predicted DNBR for the reconstituted assemblies will be shown to satisfy the minimum DNBR acceptance limit. This approach is consistent with the methodology Westinghouse uses to evaluate reloads, as described in the NRC approved topical report WCAP-9273A (Reference 3). This approach is identical to that used to evaluate the effect of stainless steel filler rods on the DNBR for IP3 cycle 8, and approved by the NRC by the issuance of IP3 Technical Specification Amendment No. 104. The results of the DNBR evaluation will be documented as part of the Indian Point 3 RSE process for cycles 9 and 10.

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

Response:

The proposed change does not create the possibility of a new or different kind of accident from any previously evaluated. The acceptability of replacing fuel rods with stainless steel filler rods will be justified as part of the cycle specific reload evaluation process using an NRC approved methodology to confirm that all existing safety criteria and design limits are met. The reload evaluation process will address 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 change does not involve a significant reduction in a margin of safety. The acceptability of replacing fuel rods with stainless steel filler rods will be justified as part of the cycle specific reload evaluation process using an NRC approved methodology to confirm that all existing safety criteria and design limits are met. The reload evaluation process will address 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 and 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. This conclusion is predicated on the assumption that, as stated previously, the cycle specific Reload Safety Evaluation process will confirm that the reconstituted assemblies satisfy the minimum DNBR acceptance limit, as well as all existing safety criteria and design limits.

Section VI - References

1) IP3 SER
2) IP3 FSAR
3) WCAP-9273A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.