ATTACHMENT A

PROPOSED TECHNICAL SPECIFICATION CHANGES

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
DECEMBER, 1993

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5.3 REACTOR

Applicability

Applies to the reactor core, reactor coolant system, and emergency core cooling systems.

Objective

To define those design features which are essential in providing for safe system operations.

A. REACTOR CORE

1. The core shall contain 193 fuel assemblies. Each fuel assembly shall consist of 204 Zircaloy-4 or ZIRLO clad fuel rods. Limited substitutions of Zircaloy-4, ZIRLO, or stainless steel filler rods for fuel rods, in accordance with NRC approved applications of fuel rod configurations, may be used. Fuel assembly configurations shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by test or cycle-specific reload analyses to comply with all fuel safety design basis. Each fuel rod shall have a nominal active fuel length of 144 inches. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

2. Deleted

3. The enrichment of reload fuel will be no more than 5.0 weight percent U-235 and will be stored in accordance with Technical Specification 5.4.

4. Deleted

5. There are 53 control rods in the reactor core. The control rods contain 142 inch lengths of silver-indium-cadmium alloy clad with stainless steel (1).

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT: REACTOR CORE

Applicability

Applies to the limiting combinations of thermal power, Reactor Coolant System pressure, and coolant temperature during four-loop and three-loop operation, and reactor coolant flow during four-loop operation.

Objective

To maintain the integrity of the fuel cladding.

Specifications

The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure 2.1-1. The safety limit is exceeded if the point defined by the combination of Reactor Coolant System average temperature and power level is at any time above the appropriate pressure line.

Basis

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by operating the hot region of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters: thermal power, reactor coolant temperature and pressure have been related to DNB through correlations which have been developed to predict the DNB

flux and location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The DNB design basis is as follows: There must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the design DNBR limit.

In meeting the DNB design criterion, uncertainties in operating parameter, nuclear and thermal parameters, fuel fabrication parameters, and computer codes must be considered. As described in the FSAR, the effects of these uncertainties have been statistically combined with the correlation uncertainty. Design limit DNBR values have been determined that satisfy the DNB design criterion.

Additional DNBR margin is maintained by performing the safety analyses to a higher DNBR limit. This margin between the design and safety analyses limit DNBR values is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility.

The curves of Figure 2.1-1 show the loci of points of thermal power Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the Safety Limit DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid. These curves are based on a peak nuclear hot channel factor of 1.62 for the LOPAR fuel and a 1.65 for the OFA and a 1.55 cosine axial power shape.

B. REACTOR COOLANT SYSTEM

1. The design of the reactor coolant system complies with the code requirements (2). Design values for system temperature and pressure are 650°F and 2485 psig, respectively.

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% Steam Generator tube plugging is 11,350 cubic feet.

References

(1) UFSAR Section 3.2
(2) UFSAR Table 4.1-9

ATTACHMENT B

SAFETY ASSESSMENT

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
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SECTION I - Description of Changes

This application for amendment to the Indian Point 2 Technical Specifications seeks to amend 1) Section 5.3.A (Reactor Core) to allow the use of VANTAGE + fuel with ZIRLO cladding and limited substitution of Zircaloy-4, ZIRLO or stainless steel filler rods for fuel rods, in accordance with NRC approved applications of fuel rod configurations, (these limitations are provided to reflect the guidance described in Generic Letter 90-02, Supplement 1), and 2) the basis to Section 2.1 (Safety Limit: Reactor Core) to allow the use of DNB Correlations applicable to VANTAGE + fuel.

SECTION II - Evaluation of Changes

Con Edison seeks to utilize Westinghouse 15 X 15 VANTAGE + fuel in its Indian Point 2 reactor. The VANTAGE + fuel uses ZIRLO as its cladding material rather than Zircaloy-4. Therefore, a change is necessary in Section 5.3.A. to include ZIRLO as an acceptable cladding material. Westinghouse has previously made a number of submittals to the NRC on the VANTAGE + fuel. This submittal only addresses operation of VANTAGE + fuel up to a rod-average burnup of 60,000 MWd/MTM. Separate submittals will address higher burnups and Loss of Coolant Accident analyses for VANTAGE + fuel. The NRC has accepted the use of ZIRLO clad, as documented in reference 1.

Section 5.3 provides a description of the fuel assemblies used in the reactor core. The description does not address fuel rod replacement. The NRC has since issued Generic Letter 90-02 and Supplement 1 to clarify the limitations on the application of current NRC-approved analytical methods used in the analysis of reconstituted fuel.

The use of reconstituted fuel assemblies has been evaluated by Westinghouse as described in WCAP-13060-P.A., entitled "Westinghouse Fuel Assembly Reconstitution Evaluation Methodology". The evaluation was performed to address the issues identified in the generic letter and has been reviewed and approved by the NRC (Reference 2). The methodology described in the WCAP or other approved methodologies will be used for each cycle where reconstituted fuel assemblies are used. Section 5.3 has been modified to incorporate the wording provided in Generic Letter 90-02, Supplement 1, to ensure consistency with NRC requirements.

Fuel assembly reconstitution involves replacing leaking or damaged fuel rods with filler rods of either ZIRLO, stainless steel, or zirconium alloy. This allows continued operation of fuel assemblies for further power production, and also permits the use of fuel assemblies that would otherwise have to be discharged from the core. The technical specifications do not prohibit the reuse of fuel assemblies that have one of more leaking fuel rods. However, good operating practice would suggest fuel assembly reconstitution to remove a potential source of radiation exposure from an increase in primary coolant fission product inventory.

Con Edison seeks to modify the existing requirements of Section 5.3 to reflect the wording recommended in Generic Letter 90-02, Supplement 1, and to incorporate Section 4.2.1 of the new Standard Technical Specifications (STS). This new wording is consistent with the NRC staff position with respect to meeting the analytical requirements for fuel assembly reconstitution to ensure compliance with General Design Criteria (GDC) 10. The proposed revisions to Section 5.3 will allow fuel assembly reconstitution in accordance with NRC-approved codes and methods and, as shown by tests and analyses, to comply with all fuel safety design basis. The change also allows the use of lead test assemblies without requiring a specific technical specification change. These limitations will ensure that those fuel configurations employing reconstituted fuel are consistent with previously approved designs and are within the existing acceptance criteria.

The FSAR was reviewed and it was determined that for purposes of transient analyses a reconstituted fuel assembly is not significantly different from other fuel assemblies, and would not introduce an unreviewed safety question in the operation of the core. In addition, industry experience has shown that plant operation with reconstituted fuel assemblies is safe and that conformance to the technical specifications provides sufficient assurance of the safe operation of the plant. Safety analyses for each core reload containing reconstituted fuel assemblies will be performed in accordance with the approved methodology (Reference 2).

Con Edison seeks to utilize DNB correlations applicable to Vantage + fuel. Therefore, the basis of Section 2.1 would be changed to more accurately describe the basis of the DNB correlations and how they are applied to ensure that the design criteria are met.

SECTION III - No Significant Hazards Evaluation

Consistent with the requirements of 10 CFR 50.92, the enclosed application involves 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:

Neither the probability nor the consequences of an accident previously analyzed is increased due to the proposed changes. As discussed in Reference 1, the fuel containing ZIRLO clad will meet all the same material and mechanical design criteria as the Zircaloy clad fuel. The use of approved Westinghouse Methodology for fuel assembly reconstitution as documented in Reference 2 will ensure that all criteria are met. The change to the basis of Section 2.1 more accurately describes DNB methodology and application.

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

Response:

The changes will not create the possibility of a new or different kind of accident. The proposed changes involve approved methodology which have been shown to meet design and safety criteria. In addition, approved procedures will be used to implement the changes.

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

Response:

The proposed amendment does not involve a significant reduction in the margin of safety. The changes involve the use of approved methodology which meet design and safety criteria. The change to the Section 2.1 basis is descriptive and will more accurately describe the DNB methodology used in conjunction with the use of VANTAGE + fuel.

SECTION IV - Impact of Changes

These changes 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 these changes: a) will not increase the probability or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the safety analysis report; b) will not create the possibility for an accident or malfunction of a new or different kind from 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.

The proposed changes have been reviewed by both the Station Nuclear Safety Committee and the Con Edison Nuclear Facilities Safety Committee. Both committees concur that the proposed changed do not represent a significant hazards consideration.

References:

1) Letter from Thadani to Tritch, "Acceptance for Referencing of Topical Report WCAP-12610, VANTAGE + Fuel Assembly Reference Core Report" (TAC No. 77258) July 1, 1991.

2) Letter from Thadani to Tritch, "Acceptance for Referencing of Topical Report WCAP-13060-P, Westinghouse Fuel Assembly Reconstitution Evaluation Methodology" (TAC No. M821391), March 30, 1993.