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July 13, 1984

Mr. H. R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555

Attention: Mr. J. R. Miller, Chief
Operating Reactors, Branch 3

Gentlemen:

DOCKET NOS. 50-266 AND 50-301
TECHNICAL SPECIFICATION CHANGE REQUEST NO. 87
ADDENDUM TO SAFETY EVALUATION FOR OPTIMIZED FUEL
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

Technical Specification Change Request No. 87 covered use of the Westinghouse Optimized Fuel Assembly design (OFA) at Point Beach Units 1 and 2. Application for the related changes was initiated by our letter dated March 14, 1983. Additional information and other editorial Technical Specification changes were contained in a subsequent letter dated September 6, 1983. In that letter, it was also noted that the "small-break" LOCA analysis had not been completed pending NRC review and approval of the latest Westinghouse models. The purpose of this transmittal is to report on the results of the small-break LOCA analysis, to submit other licensing changes related to the use of OFA fuel and to explain our plans in respect to Unit 2 Cycle 11. Specifically, the following licensing items are addressed:

1. Results of the small-break LOCA analysis performed for use of OFA fuel.

2. Results of our analysis covering the boron dilution incident under cold shutdown at half-pipe and/or without reactor coolant pumps running (reduced effective volume).

3. Figure 15.3.10-1 of the September 6, 1983 submittal is relabeled.

4. Figure 15.3.10-3 of the September 6, 1983 submittal is relabeled.

5. The Startup of an Inactive Loop incident was reviewed to cover the situation where one reactor coolant pump is lost consistent with current Technical Specifications bases (p. 15.3.3-7). This was not discussed in the Safety Evaluation (attachment B to the September 6, 1983 transmittal).

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Small-Break LOCA Analysis

Because of recent concerns that the NRC review and acceptance of the latest Westinghouse models would not be complete in time it was decided that the small-break LOCA analysis for Point Beach OFA fuel should be done using the currently accepted Westinghouse small-break LOCA analysis.

The small-break LOCA analysis for Point Beach applicable to transition and full OFA core cycles was reanalyzed due to the differences between Westinghouse standard and OFA designs. The currently approved October 1975 small break ECCS evaluation model was utilized for a spectrum of cold-leg breaks.

When assessing the transition core impact on small-break LOCA, the only mechanism available to cause a transition core to have a greater calculated peak clad temperature (PCT) than a full core of either fuel is the possibility of flow redistribution due to fuel assembly hydraulic resistance mismatch.

The W-FLASH computer code is used to model the core hydraulics during a small-break LOCA event. Only one core flow channel is modeled in W-FLASH since the core flowrate during a small-break LOCA is relatively low and this provides enough time to maintain flow equilibrium between fuel assemblies (i.e., cross-flow). Therefore, hydraulic resistance mismatch is not a factor for small-break LOCA. Thus it is sufficient to reference the small-break LOCA for the full core of the OFA design.

Results/Conclusion

The small-break OFA LOCA analysis for Point Beach utilizing the currently approved 1975 Small-Break Evaluation model resulted in a PCT of 992°F for the 6-inch diameter cold-leg break. The analysis assumed the worst small-break power shape consistent with a LOCA F_q envelope of 2.32 at core midplane elevation and 1.5 at the top of the core.

Analyses results show that the high and low head portions of the ECCS, together with the accumulators, provide sufficient core flooding to keep the calculated PCT well below the required limits of 10 CFR 50.46. Adequate protection is therefore afforded by the ECCS in the event of a small-break LOCA.

Technical Specifications Figure 15.3.10-1

The difference between this figure and that of the earlier OFA submittals is the change in the units on the ordinate axis from "steps withdrawn" to "% withdrawn". This is done to eliminate confusion and to provide labeling consistent with the discussion presented in our recent Technical Specification change request letter dated June 8, 1984.

Technical Specification Figure 15.3.10-3

The calculated values of the break-points on the curve are presented for completeness. Previous submittal figures contained rounded or truncated values.

Boron Dilution Cold Shutdown, Reduced Effective Volume

Your letter of September 30, 1980 expressed concern that the FSAR Treatment of the Boron Dilution incident at cold shutdown did not cover the condition at half-pipe or with no reactor coolant pumps running (reduced effective primary system volume).

In response, our letter of December 23, 1980 presented the results of our analysis and established operational constraints for the conditions of concern using standard design 14 x 14 Westinghouse fuel. Under these constraints the one percent shutdown margin specified in the Technical Specifications insured that criticality would not occur within 15 minutes. A similar analysis was performed to cover utilization of OFA fuel in transition and full OFA reload cores. The analysis assumed bounding values for initial boron concentration and boron worth based on Westinghouse calculations of transition and full OFA core nuclear characteristics. These values will be verified as bounding values during our review of the nuclear characteristics calculated for each cycle specific core.

For the postulated Boron Dilution at cold shutdown with reduced effective volume the use of OFA fuel will tend to reduce the time to criticality because under the worst case either the boron worth is higher (at end of cycle) or the dilution rate is higher (beginning of cycle) when boron concentration is higher than calculated for standard fuel. Accordingly, greater shutdown margin is required for cold shutdown conditions with reduced effective volume if OFA fuel is used. The attached figure, Figure 1, presents the amount of shutdown margin required as a function of initial boron concentration for dilution flow rates in terms of 1, 2 or 3 charging pumps in operation for a "worst case" full OFA core. Meeting these conditions will ensure that under the specified operating constraints the operators will have more than 15 minutes to take action before criticality is achieved.

Startup of an Inactive Reactor Coolant Loop

Startup of an idle reactor coolant pump results in the injection of relatively cold water into the core resulting in a reactivity increase. This accident is not sensitive to a positive MTC since the limiting conditions exist with a negative MTC. This accident is not impacted by OFA and does not employ ITDP. Thus, the startup of an inactive loop need not be reanalyzed.

Unit 2 Cycle 11

The safety evaluation for Unit 2 Cycle 11, the first core utilizing a full reload region of OFA fuel, has been completed by Westinghouse. The refueling shutdown of the end of Unit 2 Cycle 10 is expected to begin on September 28, 1984 at a Cycle burnup of about 13,700 MWD/T. Startup of Cycle 11 is expected to occur in early November following refueling with the first reload region of OFA fuel.

The Unit 2 Cycle 11 reload core is designed to operate under current nominal transition core design parameters, Technical Specifications (incorporating Technical Specification Change No. 87), related bases, and transition core setpoints.

The reload fuel mechanical and thermal-hydraulic design for the Cycle 11 reload core will be unchanged from that discussed in the Safety Evaluation transmittal of September 6, 1983. The reload core meets the F_Q^TxP limit of less than 2.21 which is consistent with OFA reload nuclear designs. The current F_{Δ}^NH limit of less than 1.58 ensures that the DNB ratio will be greater than 1.30.

Based on the Westinghouse safety evaluation it is concluded that the Unit 2 Cycle 11 design does not cause the previously acceptable safety limits to be exceeded. This conclusion is based on the following:

1. Cycle 10 burnup is between 12,650 and 14,150 MWD/T.

2. Cycle 11 burnup is limited to 11,250 MWD/T (includes 1000 MWD/T of coastdown).

3. The analyses and proposed Technical Specification changes submitted in Technical Specification Change Request No. 87 are approved by the NRC.

4. There is adherence to the plant operating limitations given in the Technical Specifications, including the changes included in Technical Specification Change Request No. 87.

Verification of the core design will, of course, be performed by means of the standard startup physics tests normally performed at the start of each cycle.

We trust that the foregoing completes the application for Technical Specification Change Request No. 87. As required by the Commission regulations, we have enclosed three signed originals and forty copies of this submittal which includes a modification to our previous Technical Specification Change Request. Please note that we have considered these modifications to our proposed specification change in light of the requirements of 10CFR50.92 and have concluded that these administrative changes do not alter or negate the determination reached in our previous application that these changes involve no significant hazards consideration. Please contact us as early as possible if you have any requests for additional information.

Very truly yours,

CW Fay

Vice President-Nuclear Power

C. W. Fay

Copy to NRC Resident Inspector
C. F. Reiderer

Subscribed and sworn to before me this 13th day of July, 1984.

Gl- a J. Monsoor
Notary Public, State of Wisconsin
My Commission expires 6-12-88.