TOLEDO EDISON

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Docket No. 50-346

License No. NPF-3

Serial No. 631

July 16, 1980

Director of Nuclear Reactor Regulation
Attn: Mr. Thomas M. Novak
Assistant Director Operating Reactors
Division of Licensing
United States Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Novak:

This is in response to your letter of July 1, 1980 (Log No. 574) requesting further information concerning Cycle 2 operation of the Davis-Besse Nuclear Power Station, Unit No. 1 (DB-1).

Attachment A to this letter addresses your staff's two questions on nuclear design and transient analysis. Attachment B addresses your staff's questions on startup testing at Davis-Besse Unit 1.

As always, if you have any questions on the information please contact us.

Very truly yours,

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Attachments

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Attachment A

Responses to Questions on Nuclear Design
and Transient Analysis for Davis-Besse
Nuclear Power Station, Unit 1 (DB-1)
From NRC Letter of July 1, 1980 (Log No. 574)

Question 1

"The Technical Specification on total peaking factor permits Fq values as high as 2.94/P where P is the fraction of full power. The Bases for Specifications 2.1.1 and 2.1.2 state that a total peaking factor of 2.56 was used at a core power of 112% of full power to show compliance with DNBR limits. But if normal operation is permitted with a peaking factor of 2.94 and an overpower transient (such as rod bank withdrawal) occurs DNBR limits appropriate for anticipated transients would be violated. Please explain the apparent discrepancy or resolve it."

RESPONSE

The Fq value for the Davis-Besse Technical Specifications (2.94) has historically been the 6 foot level LOCA limit (18.4 kw/ft for Davis-Besse) divided by the densified average linear heat rate at 102%FP (6.14 kw/ft x 1.02). The LOCA kw/ft limits for Davis-Besse 1, Cycle 2 are given as a function of core height in Table 7-2 of BAW-1598, Davis-Besse Nuclear Power Station, Unit 1, Cycle 2 Reload Report. The normal operating controls are set so that they preserve the LOCA kw/ft limits at all elevations. In practice the peaks which determine the normal operating controls occur near the 2 and 10 foot elevations and not at the core midplane. As stated in BAW-10122A, Normal Operating Controls, page vii, response to NRC question 232.1, the normal operating controls on regulating rod bank position, power imbalance and APSR position preserve the most restrictive of the initial DNB and LOCA kw/ft criteria. The 2.56 value of total peaking quoted in the Bases for Technical Specification 2.1.1 and 2.1.2 preserves the limiting DNBR of 1.30 at 112%FP. However, as stated in the Bases, the combination of radial and axial peak which produces this DNBR varies such that a considerably higher total peak is allowed for some positions of the axial peak. The objective is to preserve the 1.30 DNBR. The normal operating controls are set so that, should an overpower transient occur, they would preserve a DNBR of 1.30 at 112 percent power. This is accomplished by considering a family of allowable peaks which preserve a much higher DNBR during normal operation at or below 102%FP.

Since, within the normal operating controls, no peaks are allowed which would violate either the LOCA kw/ft or initial condition DNBR criterion, the Technical Specification on Fq is unnecessary, redundant, and provides no real initial condition control. The normal operating controls provide continuous assurance that the initial conditions for both DNB limiting accidents and LOCA are preserved.

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Attachment A

Question 2

"In the FSAR the fuel misloading error was treated by asserting that it was very unlikely that it could occur due to the great care taken in the manufacture and loading of assemblies. Further it was asserted that serious misloading errors would be discovered during startup testing. Report BAW-10028, "Effect of Asymmetries in Fuel Loading," shows that certain misloadings can result in violation of thermal limits. Have these conclusions been confirmed for the second cycle of Davis-Besse? Has an analysis similar to that performed for the Midland Plant (Docket 50-329/330, Response to RAI 232.12) been done?"

RESPONSE

The conclusions stated in the FSAR regarding the likelihood of fuel misloading and the discovery of serious misloading errors apply equally to reload cycles. Misloading the fuel pins in an assembly is prevented by loading controls and procedures. Each fuel rod is identified by an enrichment code,and the design of the reactor is such that only one enrichment is used per assembly. The manufacturing process relies on administrative procedures and quality control checks to ensure that fuel rods are placed in the proper assembly. One such administrative procedure that is practiced to the extent practical is the "campaigning" of enrichments to ensure that only a single enrichment is handled at a given time in fuel fabrication.

Gross fuel assembly misplacement in the core is prevented by administrative core loading procedures and the prominent display of identification markings on the upper end fitting of each assembly. The fuel handling bridges and grapple mechanisms are designed for accurate indexing and positioning. To ensure seating of assemblies on the fuel grid, hoist tape readings are verified. In addition to this, visual verifications are utilized.

(a) Prior to the commencement of loading operations, a check is made to ensure that fuel assemblies have been stored in their proper storage locations in the spent fuel pool and that control components are inserted in the proper fuel assemblies.

(b) During loading or unloading of each assembly into or out of the reactor core the following visual verifications are made:

i. When picking up the assembly, two persons independently visually verify that it has been picked up from the proper location in the spent fuel pool or the reactor core.

ii. While in transit, the identification markings on the assembly are checked.

iii. When the assembly is placed in the final location, two persons independently visually verify that it has been placed in the proper location in the core or spent fuel pool.

RESPONSE to Question 2 (Continued)

(c) Prior to installing the reactor upper internals, the core loading is independently verified by two persons visually surveying the core and recording the fuel assembly numbers versus core location. This record is then compared to the core loading plan.

The Fuel/Control Component Handling Operating Procedure (PP-1502.04) specifies how the fuel assemblies are to be oriented in the spent fuel pool and in the reactor with respect to fixed references. In order to identify the fuel assemblies in the verifications mentioned in items b and c above, it is necessary for the observer to look at the fuel assembly identification number, which is located only on one side of the assembly.

A detailed evaluation was made of the detectability of a misloaded fuel assembly for the Midland Plant (Docket 50-329/330, Response to RAI 232.12). Although a similar analysis was not performed specifically for Cycle 2 of Davis-Besse, the conclusions reached as a result of the Midland study are considered to apply to all BAW 177 fuel assembly cores in general, and to Cycle 2 of Davis-Besse in particular. Any fuel assembly misloading of significance from a nuclear standpoint will result in identifiably anomalous beginning of cycle (BOC) measured data. Conversely, the perturbation produced by a minor fuel loading error that is not detectable during BOC testing does not violate thermal limits when operating at 102% of full power. In addition, any such perturbation will diminish with increasing cycle burnup.

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Attachment B

Responses to Questions on Startup Testing for
Davis-Besse Nuclear Power Station, Unit 1 (DB-1)
From NRC Letter of July 1, 1980 (Log. 574)

Question 1

"Section 9.2.1 critical boron concentration states that the acceptance criteria for this test will be ±100 ppm. Please state the review criteria."

RESPONSE

Toledo [illegible] established and does not plan to establish "review criteria" for star[illegible] detailed in an internal NRC memorandum from Paul S. Cheek dated No[illegible] 1978. The reasons include:

(a) Currently an on-site evaluation is performed on these startup test results by the performance staff and by the fuel vendor's (B&W) test engineer prior to escalating power. Additionally the fuel vendor (B&W) evaluates all test data, both on-site and in Lynchburg.

(b) The vendor test engineer is on-site at all similar reactor refueling startups.

The level of review by the Davis-Besse performance staff and its vendor exceeds that being performed by other reactor vendors and should therefore be considered a more than adequate substitute for review criteria.

Question 2

"Section 9.2.2 Temperature Reactivity Coefficient states an acceptance criteria of $\pm 0.4 \times 10^{-4}$ $\Delta k/k/^{o}F$. Please state the review criteria for this test."

RESPONSE

See Toledo Edison's response to Question 1 above.

Question 3

"Please state what further rod worth tests will be performed if the sum of the measured values for groups 5, 6 and 7 is more than 10% less than the predicted value for this sum."

RESPONSE

If the sum of the measured values for Groups 5, 6 and 7 is more than 10% less than the predicted value for the sum, the test program requires measurement of Group 4, with an acceptance criteria that requires the sum of Groups 4, 5, 6, and 7 to be within 10% of the predicted value.