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December 19, 1983

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Mr. Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Serial No. 706
WBR/NAS:bhg/0011N
Docket Nos.: 50-280
50-281
License Nos.: DPR-32
DPR-37

Gentlemen:

SUPPLEMENT 2 TO AN AMENDMENT TO OPERATING LICENSES DPR-32 AND DPR-37
PROPOSED REDUCTION IN BORON CONCENTRATIONS
SURRY POWER STATION UNITS 1 AND 2

In our letter dated September 13, 1983 (Serial No. 521), Vepco requested an amendment to Operating Licenses DPR-32 and DPR-37 to allow operation of Surry Unit Nos. 1 and 2 at reduced boron concentrations. This letter provides in Attachment 1, supplemental information in response to questions forwarded informally to Vepco earlier this month.

Should you have any further questions, please contact us at your earliest convenience.

Very truly yours,

W L Stewart

W. L. Stewart

Attachment

(1) Response to Reactor Systems Branch questions for the proposed reduction in boron concentrations for Surry Power Station

cc: Mr. James P. O'Reilly
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ATTACHMENT 1

RESPONSE TO REACTOR SYSTEMS BRANCH QUESTIONS

FOR THE PROPOSED REDUCTIONS IN BORON CONCENTRATION

FOR SURRY POWER STATION

NRC Question 1:

Provide the details and justification of the reactivity feedback model and the mixing coeffients utilized in the Main Steam Line Break Analysis contained in your submittal of September 13, 1983.

Response:

Vepco's RETRAN model represents interloop flow mixing phenomena with a set of cross-flow mixing junctions, as described in our topical report (Reference 1). The downcomer, inlet plenum and outlet plenum of the reactor vessel are each divided into two azimuthal regions or sectors. One sector represents the 1/3 portion of the reactor adjacent to the faulted loop (i.e., the loop which cools down most rapidly during the steam line break). The other sector represents 2/3 of the reactor adjacent to the model loop which represents the two intact loops in the plant.

Mixing between the loops is then represented by a pair of flow junctions which extend from the inlet of one loop to the inlet region of the other loop. The loss coefficients for these junctions are determined during steady state initialization by specifying equal and opposite mixing flows and a reactor vessel inlet pressure distribution based on steady state hydraulic calculations.

The degree of interloop flow mixing in the inlet region of the reactor vessel may be characterized by a mixing parameter, fmi, which is defined in Reference 2 as fmi = fraction of the coolant flow emerging from an inlet nozzle which flows up the radial part (per loop) of the core nearest the inlet nozzle.

Scale model mixing tests have shown that fmi is about 0.66 to 0.75, as discussed in Reference 2. Vepco sensitivity studies have shown that high values of fmi (minimum inlet mixing) produce the maximum peak heat fluxes and the most thermally limiting conditions for steam line break. As a result, the Reference 3 analyses are based on a value of fmi = 0.75.

The Vepco RETRAN model also accounts for mixing in the core exit/outlet plenum region of the core. The core exit mixing is modeled by cross flow paths between the two radial core exit regions and the opposing outlet plenum volumes. The loss coefficients are determined during steady state initialization to provide an overall vessel mixing fraction fm numerically equal to 0.7, where fm is defined as

fm = fraction of flow entering the vessel from one loop which returns to the same loop.

The numerical value of fm chosen again represents the results of scale model mixing tests, as discussed in Reference 2. The steamline break results are less sensitive to core exit mixing than to core inlet mixing.

The RETRAN mixing model is thus a representation of observed vessel average mixing behavior. The effects of more detailed core inlet mixing phenomena (i.e., assembly-to-assembly inlet temperature variations) are accounted for in the calculations performed to verify the conservatism of the RETRAN reactivity feedback coefficient, as discussed below.

The RETRAN reactivity feedback model used in Vepco's steamline break calculations may be described by the following expression

$$\rho_{total} = \rho_{initial} + \rho_{mod}\,(T_{mod}) + \rho_{power}\,(P) + \alpha_B C_B \quad (1)$$

where

$\rho_{initial}$ = reactivity corresponding to design shutdown margin 1.77% dK/K

ρ_{mod} (Tmod) = is the moderator feedback. The feedback function used is derived from the curve presented as Figure 1, Section B of Attachment 1 to our September 13, 1983 submittal (Reference 3). Tmod is a weighted core average moderator temperature which will be discussed further below.

ρ_{power} (P) = is a power reactivity feedback function which accounts for the effects of doppler reactivity from the high fuel temperatures near the stuck control rod assembly. For cases where steam generation occurs in the high flux regions of the core, the effects of void formation on the reactivity has also been included. The effect of power generation in the core on overall reactivity is shown in Figure 2, Section B, of Attachment 1 to Reference 3. As seen in that figure, separate curves are used for full RCS flow and low RCS flow cases, since the relationship between local voiding in the vicinity of the stuck rod and core power level is strongly influenced by the core flow rate.

α_B = is the differential boron worth

C_B = is the core average boron concentration; as discussed in Reference 3, this concentration is calculated using a detailed submodel in conjunction with the RETRAN control system modeling capability.

The weighted average moderator temperature Tmod is calculated from the expression

$$Tmod = \frac{1-RWF}{4}\sum_{i=1}^{4} Tc^{i} + \frac{RWF}{4}\sum_{i=1}^{4} Th^{i}$$

Where

Tc^i are the average core node moderator temperatures on the "cold" 1/3 of the core (nearest the break)

Th^i are the average core node coolant temperatures on the "hot" 2/3 of the core (nearest the intact loops) and

RWF is a radial weighting factor. The value of RWF is chosen to conservatively represent the effects of having a stuck rod cluster control assembly (RCCA) fully withdrawn in the region of the core which receives the coldest water.

The values of the functions ρ_{mod} (Tmod), ρ_{power} (P), and the weighting factor RWF have been selected based on past Surry reload core experience to provide a conservative representation of the reactivity feedback phenomena.

The conservatism of these functions is checked for several carefully chosen statepoints for each case using the neutronics models and methodology described in Reference 4 and 5. These check calculations were performed for a recent Surry reload core design. In performing the checks, no credit was taken for the fact that recent reload cores have a minimum end of cycle shutdown margin which is well in excess of the design value assumed in the RETRAN analysis. The check calculations included the effects of inlet temperature variations across the core, as discussed in Reference 4, the effects of boron concentration and of thermal-hydraulic feedback.

In each case, it has been confirmed that the increase in reactivity from time zero to the statepoint under investigation as calculated by RETRAN is greater than that calculated by the neutronics models.

Thus, the overall conservatism of the RETRAN reactivity feedback model used for these calculations has been confirmed.

References:

1. Letter from W. N. Thomas (Vepco) to H. R. Denton (NRC), Serial No. 215, April 14, 1981, Attachment 1, VEP-FRD-41, "Vepco Reactor System Transient Analyses Using the RETRAN Computer Code".

2. WCAP-7909, "MARVEL-A Digital Computer Code for Transient Analysis of a Multiloop PWR System", H. G. Hargrove, Westinghouse Electric Corporation, October 1972.

3. Letter from W. L. Stewart (Vepco) to H. R. Denton (NRC), Serial No. 521, "Amendment to Operating Licenses DPR-32 and DPR-37, Surry Power Station Units 1 and 2, Proposed Technical Specifications Change", September 13, 1983.

4. WCAP-9272, "Westinghouse Reload Safety Evaluation Methodology", F. M. Bordelon, et. al., Westinghouse Electric Corporation, March 1978.

5. WCAP-9227, "Reactor Core Response to Excessive Secondary Steam Releases", S. D. Hollingsworth and D. C. Wood, Westinghouse Electric Corporation, January 1978.

NRC Question 2:

Clarify your proposed heat tracing requirements for the Safety Injection (SI) lines assuming implementation of your recommendation to reduce the boric acid concentration. On Page 9 of your submittal, you state that lowering the boric acid concentration in the BIT to 0 ppm eliminates the need to maintain heat tracing on the associated SI piping. Discuss whether the normally stagnant sections of the SI piping between the normal charging pump discharge and the closed isolation valves upstream of the BIT can contain concentrated boric acid due to operator error or equipment failure (e.g. valve leakage). If this can occur explain how precipitation of the boric acid and consequent pipe blockage is avoided. Include the type and frequency of checks performed.

Response:

Except for Reactor Coolant System (RCS) boration operations, the main charging header will normally contain non-stagnant borated water at a concentration between 0 and 2000 ppm. At these concentrations, boron precipitation is not a concern; however, charging lines located outside the containment or auxiliary building (such as the suction lines from the refueling water storage tank) are heat traced to prevent freezing. The normal charging header and the stagnant SI piping between the BIT inlet valves and the main charging header are not currently heat traced, since these lines are located in buildings (i.e., auxiliary building and containment). Although the charging header will at times carry concentrated boric acid from the concentrated boric acid system to the RCS, the transport time is of short enough duration to prevent solution temperatures from dropping enough to cause boron precipitation in the insulated piping. This has been verified by over 10 years of operating experience. During this time, boron precipitation in the charging header and stagnant lines off the charging header has not been a problem and it is not expected to be a problem after the implementation of the reduced boron concentration requirements for the BIT.

Although not required to prevent boron precipitation, the stagnant SI piping between the main charging header and the BIT inlet valves is flushed monthly, in accordance with Technical Specification 4.1.E (Flushing of Sensitized Stainless Steel Pipe).

NRC Question 3:

Provide the results of the offsite dose calculations as a consequence of your revised main steam line break analysis for both the case of a preaccident iodine spike and concurrent iodine spike. Provide pertinent assumptions utilized in your calculations, including primary to secondary leakage.

Response:

As discussed in Reference 1, the main steam line break analyses submitted in support of the proposed boron concentration reduction have shown that departure from nucleate boiling (DNB) will not result as a consequence of this accident. Consequently no radioactivity is released to the environment because of a steam line break unless there is or has been primary to secondary system leakage in a steam generator. Current Technical Specifications limiting primary to secondary leakage and reactor coolant system activity will not change as a result of the revised boron concentration limits. The leakage and RCS activity limits are well below the values assumed in the currently applicable licensing analysis of offsite doses resulting from steam line rupture events (Reference 2).

In addition, in Reference 3, the NRC staff concluded that Vepco's consideration of the iodine spiking phenomenon with respect to accident calculations for fuel enrichments at up to 4.1 wt % and burnups of up to 37,000 MWD/MTU is conservative. This conclusion was based on the observation that Vepco was not requesting changes to plant technical specifications for 1) the magnitude of the equilibrium or the "spike" iodine concentration, 2) the surveillance requirements or 3) the restrictions on the total time the plant may operate above the equilibrium concentration. The above operational constraints remain unaffected by the proposed changes in boron concentration requirements submitted by Reference 1. We therefore conclude that the offsite dose calculations associated with the currently docketed safety analyses remain bounding.

In summary, Vepco has concluded that the effects of the proposed boron concentration reductions on the offsite doses associated with the main steam line break event are bounded by the calculations of Reference 2, and that the calculations adequately cover the effects of a potential preaccident or concurrent iodine spike.

References:

1. Letter from W. L. Stewart (Vepco) to H. R. Denton (NRC), Serial No. 521, Amendment to Operating Licenses DPR-32 and DPR-37, Surry Power Station Units 1 and 2, Proposed Technical Specifications Change, September 13, 1983.

2. Surry Power Station, Units 1 and 2, Updated Final Safety Analysis Report, Virginia Electric and Power Company.

3. "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 73 to Facility Operating License No. DPR-32 and Amendment No. 74 to Facility Operating License No. DPR-37, Virginia Electric and Power Company, Surry Power Station Unit Nos. 1 and 2, Docket Nos. 50-280 and 50-281", U. S. Nuclear Regulatory Commission, January 19, 1982.