TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401
400 Chestnut Street Tower II

March 15, 1983

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Denton:

In the Matter of the	)	Docket Nos. 50-259
Tennessee Valley Authority	)	50-260
		50-296

In response to verbal requests from your staff, we are providing the enclosed additional information regarding Browns Ferry RETRAN methodology in our reports TVA-TR81-01 and TVA-MDS-553. Enclosure 1 provides information discussed in a March 2, 1983 conference call with your staff and enclosure 2 provides information discussed in a March 8, 1983 conference call. It is our understanding that the enclosed information resolves all NRC concerns on this matter.

As indicated in my January 20, 1983 letter to you, your immediate approval of our submittal is requested. Any further delays in your approval beyond our original requested date of November 1, 1982 will result in schedule complications for TVA in preparation of our in-house reload core design for unit 3 cycle 5.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills

L. M. Mills, Manager
Nuclear Licensing

Subscribed and sworn to before
me this 15th day of March 1983.

Paulette H. White

Notary Public

My Commission Expires 9-5-84

A001

Enclosures
cc: See page 2

Mr. Harold R. Denton March 15, 1983

cc (Enclosures):
U.S. Nuclear Regulatory Commission
Region II
ATTN: James P. O'Reilly, Regional Administrator
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30303

Mr. R. J. Clark
Browns Ferry Project Manager
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, Maryland 20814

ENCLOSURE 1
BROWNS FERRY NUCLEAR PLANT RETRAN
MARCH 2, 1983 CONFERENCE CALL

Q1. Justify the assumption that the values of the random variables in the response surface equation are normally distributed.

A1. The distribution of scram speeds (variable SS in equation 8-2) and response surface fitting errors (variable URS) were tested and no reason to reject the assumption of normality was found. Insufficient data is available to test the distribution of initial steam flow (SF) and model uncertainty (URM), but the use of a normal distribution is a common practice in the absence of contrary information. The distributions of scram speed and model uncertainty have the dominant effect on the statistical adjustment factors. Since the model uncertainty can be expected to be due to the combined uncertainty in a large number of independent components, the central limit theorem of statistics would indicate that the assumption of a normal distribution for URM is reasonable.

Q2. The model uncertainty of 25 percent was arrived at primarily from comparisons to the Peach Bottom turbine trip tests. However, this uncertainty was used not only for the load rejection event but for the feedwater controller failure also. Justify.

A2. The model uncertainty was set by examining both the Peach Bottom turbine trip test results and the results of sensitivity studies shown in tables 7-1 and 7-2. The sensitivity study results for the FWCF generally showed the same trends as for the GLRWOB but with the magnitude of the sensitivity reduced by approximately 1/2 to 1/3. Since the RCPR for the FWCF is approximately 1/3 less than for the GLRWOB, the fractional uncertainties are roughly equivalent.

It should also be noted that a FWCF event evolves into a turbine trip with operation of the bypass valves and the major change in CPR occurs following the turbine trip. Since the Peach Bottom tests were turbine trips with bypass operation, they are appropriate for evaluating uncertainties in model predictions of FWCF events.

Q3. Justify the use of initial steam flow instead of initial power level as a random variable in the response surface equation.

A3. It makes little difference whether initial steam flow or power is selected as the random variable in the response surface equation since they are directly related (for a given operating pressure set point and feedwater heating characteristic). When the initial values are expressed as percent of rated values, then both steam flow and power are very nearly equal numerically as shown in the table below:

Initial Steam Flow (%NBR)	Initial Power (% NBR)
90	90.7
95	95.1
100	99.5
105	104.5
110	109.5

However, for pressurization events resulting from stopping the steam flow to the turbine, the severity of the event is most closely related to the pressurization rate which is, in turn, more directly related to the initial steam flow than to the initial power. Therefore, initial steam flow was selected as the variable for the response surface but results are not appreciably affected by use of initial power.

Q4. Please provide the A_i coefficients in equation 8-2 for each response surface.

A4. The response surface fitting coefficients are listed below for the four response surfaces for which statistical adjustment factors were calculated.

	GLRWOB at EOC	GLRWOB at MOC	FWCF at EOC	FWCF at MOC
A_0	1.13970 E-1	6.25919 E-2	9.54929 E-2	6.83487 E-2
A_1	1.21866 E-3	1.33998 E-3	-4.04006 E-4	-3.42683 E-4
A_2	-1.30147 E-5	1.34697 E-5	-2.16423 E-5	4.79854 E-5
A_3	2.41622 E-1	1.35571 E-1	8.00299 E-2	1.07703 E-1
A_4	1.50950 E-3	3.23100 E-3	9.37314 E-4	2.40229 E-3
A_5	5.63364 E-3	-5.90817 E-2	-9.20517 E-2	-1.52479 E-1

ENCLOSURE 2
BROWNS FERRY NUCLEAR PLANT RETRAN
MARCH 8, 1983 CONFERENCE CALL

Q1. In collapsing from 3D to 1D for various cross sections, either flux or flux-adjoint weighting may be used. Apparently flux weighting is normally used. Is flux-adjoint weighting ever used? If so, under what conditions?

A1. Our current practice is to use flux-adjoint weighting. However, sensitivity studies have shown little difference in transient results for the licensing basis pressurization events when flux weighting is employed. For example, the GLRWOB events described in chapter 6 of TVA-TR81-01 yields the results shown below when analyzed identically, except for the radial weighting used in cross section collapsing.

<u>RETRAN Results for GLRWOB</u>

	Weighting used in 3D to 1D Cross Section Collapse	
	Flux	Flux* Adjoint
Peak power (%NRB)	378.32	382.42
time (sec)	0.635	0.635
Max core avg. heat flux (% NBR)	119.75	119.85
time (sec)	0.850	0.845
Peak dome pressure (psia)	1207.18	1207.13
time (sec)	2.425	2.425
Max ΔCPR	0.221	0.222

Q2. In table 3 of MDS-353, the standard deviation in the percent error in reactivity change is noticeably high for the all rods out configurations PB, TT2 and TT3 tests. Why?

A2. The initial conditions for the Peach Bottom tests employed rod patterns with a large number of partially inserted control rods. During the transients, the control distribution will vary between the initial distribution and the all control rods fully inserted configuration. The states between the initial distribution and all rods out will not be encountered. Therefore, in developing the RETRAN 1D cross section fits, the initial and all control rods inserted states were more heavily weighted to improve the accuracy of the fits in the range they would actually be evaluated. This results in larger fitting errors in the unachievable all rods out configuration.

Q3. The 13 percent uncertainty in the void coefficient used to evaluate ΔRCPR in table 7-1 (page 269) is based on a change of 75 psi in the system pressure. However, a 160 psi change in pressure occurred in the GLRWOB event (figure 6-4). Should the 13 percent be scaled to account for this or is it assumed that the uncertainty does not depend on the magnitude of the pressure increase?

A3. The results in table 7-1 do not need to be scaled since the cross section polynomials were modified to obtain a 13 percent more negative void coefficient, and thus, the results in table 7-1 reflect the actual pressure change, not an assumed 75 psi change. The magnitude of the uncertainty in void coefficient was assumed to be independent of the size of the pressure change, but the 13 percent is a conservative estimate of the uncertainty.

Q4. Certain sources of uncertainties in the scram reactivity, such as the basic cross sections and assembly modeling uncertainties, are common to both KENO and LATTICE calculations. Are these considered or are they assumed to be covered by other conservatisms?

A4. There are potentially some sources of uncertainty common to both the KENO and LATTICE calculations; however, these should be smaller in magnitude than the uncertainties due to approximations in the neutron transport solution. Also, the maximum difference in control strength between KENO and LATTICE was less than 5 percent while a 10 percent uncertainty in scram reactivity was employed. This was judged to be an adequate allowance for the common uncertainties not identified by the KENO-LATTICE comparisons.

Q5. Neglecting radial distribution changes during transients affects the scram reactivity. How was this treated?

A5. As discussed in section 9 of TVA-MDS-553, the major shortcoming identified for the 1D control representation employed in RETRAN was a tendency to underestimate the worth of bank insertion movement for configurations with control rods initially inserted to different levels. The underestimate of bank movement worth is due to the decreased flux perturbation caused by a control rod tip at an axial plane with high void content relative to the perturbation caused at a lower axial plane with lower voids. This tendency to underestimate scram reactivity is conservative for licensing basis analyses with initially partially inserted control rods.

Q6. How were RETRAN modeling uncertainties included as opposed to input uncertainties?

A6. The sensitivity studies presented in chapter 7 of TVA-TR81-01 examined several sources of uncertainty, including those arising from RETRAN modeling (examples are the subcooled void model and slip models). These uncertainties were combined into the 0.041 penalty applied to the calculated operating limit CPR (equation 8-1) for option A. The modeling uncertainties are accounted for in the option B operating limit CPR by the URM (equation 8-2) variable employed in the response surface analysis for the statistical adjustment factors.

Q7. There appears to be some ambiguity in the treatment of uncertainties. For example, the 13 percent void reactivity. Uncertainty used in table 7-1 is apparently a one-sigma value (although this is not clear) but is treated as a 95/95 uncertainty in table 8-1 on page 295. Please clarify.

A7. The 13 percent void reactivity uncertainty is a bounding value; however, the method used to arrive at this value in section 7.1.1.1 of TVA-TR81-01 is inappropriate for several reasons. The conservatism of the assumed 13 percent uncertainty in void reactivity can be demonstrated by examining the difference in peak excess reactivity inferred by inverse point kinetics from the measured data and calculated by the RETRAN model for the Peach Bottom turbine trip tests. The results in the table below indicate a 95 percent confidence upper bound (from χ^2 test) on the standard deviation between measured and calculated peak reactivity of 4.8 percent. Thus, a 95/95 reactivity uncertainty of 9.6 percent is indicated by the Peach Bottom turbine trip tests confirming the conservatism in the assumed 13 percent uncertainty.

Peak Excess Reactivity ($)

Test	Data	Calculation	% Difference
TT1	0.803	0.790*	-1.62
TT2	0.793	0.797	+0.50
TT3	0.829	0.821	-0.97
		Average	-0.70
		Standard Deviation	1.10
		95% Confidence Standard Deviation	4.80

*based on calculation driven with measured upper plenum pressure to eliminate reactivity difference due to uncertainties in pressure prediction of TT1. TT2 and TT3 results are from base calculation (stop and bypass valve position boundary conditions) since no significant differences in measured and calculated upper plenum pressures were observed.

Q8. Are the different perturbations listed in table 2-7 (page 66) uncorrelated?

A8. At many axial planes in the core (especially near the top), the relative perturbations in nodal water density produced by separately perturbing system pressure, power, flow, etc., will be correlated.