400 Chestnut Street Tower II

October 25, 1979

Director of Nuclear Reactor Regulation
Attention: Mr. Thomas A. Ippolito, Chief
Branch No. 3
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Ippolito:

In the Matter of the	)	Docket No. 50-296
Tennessee Valley Authority	)	

This is in response to your request for additional information concerning the reload analyses for Browns Ferry unit 3 cycle 3 operation. Enclosed is the information requested by your September 19, 1979, letter to H. G. Parris. If you have any further questions, please get in touch with us.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills, Manager
Nuclear Regulation and Safety

Enclosure

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TVA RESPONSE TO NRC'S
REQUEST FOR ADDITIONAL INFORMATION
FOR BROWNS FERRY UNIT NO. 3
RELOAD FOR CYCLE 3

Question 1: Your letter dated August 6, 1979 (TVA BFNP TS 127) transmitted analysis for operation of Browns Ferry Unit No. 3 (BF-3) in the third fuel cycle. During the current outage, we understand you are modifying the end-of-cycle recirculation pump trip (RPT) system on Unit No. 3 to be the same as the RPT systems on Units Nos. 1 and 2. The transient analyses in your August 6, 1979 submittal, which include credit for the RPT system in establihsing operating limits, has been performed with the REDY code. During the past year, there have been a number of discussions and correspondence between the TVA and NRC staffs on whether the REDY Code vs. ODYN Code is the better predictor of plant behavior as transient severity is reduced (by the RPT system). Our position - with which we believe you concur - is that the ODYN Code, which uses a more physically correct model of the plant is probably a better predictor of changes in critical power ratios. During the previous Unit No. 3 reload submittal, as well as the subsequent reload submittals on Units Nos. 1 and 2, we had requested an ODYN analysis of the limiting pressurization transients to establish operating limit minimum critical power ratios (OLMCPRs). This subject has been discussed at length in our safety evaluations supporting the most recent reload amendments on Units Nos. 1, 2, and 3. We would still prefer and are requesting, for the current Unit No. 3 reload a reanalysis of the load rejection without bypass (LRWOBP) and the feedwater controller failure transients with the proposed licensing basis ODYN Code, as applied in the letter from E. D. Fuller, General Electric Company, to D. F. Ross, NRC, dated June 26, 1979, "Impact of ODYN Transient Model on Plant Operating Limits". Your previous position has been that you would perform any reload analysis with either REDY or ODYN - but not both - because of the time and cost required for duplicate analysis. You also raised the question of the acceptability of ODYN analyses until such time as the ODYN Code is approved by NRC. On the most recent reload amendments for Units Nos. 1, 2, and 3 we have resolved this issue by adding a margin to the OLMCPRs calculated by the REDY Code to account for possible lack of conservatism at the end of the fuel cycle when transient effects are most severe. You are requested to provide an ODYN analysis of the limiting pressurization transients as discussed above. If you do not propose to provide these ODYN analyses, explain the basis for your position and propose appropriate margins to the OLMCPR's with justification therefore.

Response 1: Until NRC's review of ODYN is complete, TVA's transient analyses will be based on the accepted REDY code. TVA acknowledges that some uncertainty may exist in the ability of REDY to accurately predict pressurization rate in all cases. However, overall, we believe that REDY provides a conservative calculation for the current licensing basis transients. Nevertheless, in order to account for any possible nonconservatisms of REDY, TVA commits to implement a ΔCPR penalty of .03 for the pressurization events for EOC-2000 MWd/t.

The unit 3 cycle 3 operating limit MCPR's should be revised as follows:

8x8 fuel; BOC 3 through EOC 3: 1.28
8x8R fuel; BOC 3 through EOC 3-2000 MWd/T: 1.22
EOC 3-2000 MWd/t through EOC 3: 1.25
P8x8R fuel; BOC 3 through EOC 3-2000 MWd/t: 1.23
EOC 3-2000 MWd/t through EOC 3: 1.26

Question 2: The proposed Technical Specifications for Cycle 3 include a change (pg. 75) which would allow BF-3 to operate at up to 85% power with neither RPT operable for an indefinite period of time. The present Technical Specifications require that the unit be brought to below 30% power within 24 hours if both RPT systems are inoperable. Provide justification for this change via a plant and cycle specific analysis of the most severe pressurization transients occurring from 85% power without taking credit for the RPT feature. Show that the safety limit MCPR will not be violated by any fuel type assuming the respective proposed operating limit MCPRs. If the proposed specification is intended to be applied to all future cycles of BF-3 the analysis should bound expected future core characteristics, otherwise cycle specific analyses may be required.

Response 2: Performance studies of a BWR at reduced power and flow shows that the ΔCPR for pressurization events (load rejection without bypass, etc.) can be reduced to 60 percent of the value based on the 100 percent power/flow stated when the core power has been reduced to approximately 85 percent by flow control. These studies were performed for several transient events including the Load Rejection Without Bypass using a standard computer program and licensing conservatism. The study did not include Recirculation Pump Trip (RPT) but other studies relating improvements with RPT have shown that RPT reduces ΔCPR to approximately 60 percent of that without RPT. The study was performed at end-of-cycle conditions (i.e., all rods out) at other operating conditions (lesser exposures, etc.), where events such as Rod Withdrawal Error which are not affected by the operation of the RPT system would be limiting. Thus, reducing core power to 85 percent without RPT by flow control would result in a trade-off not requiring MCPR operating limit changes.

Additional conservatism will also appear in a form of higher operating Minimum Critical Power Ratio when the core power is reduced. This additional conservatism is worth approximately 7 percent at the reduced core power and flow described previously. Thus, should the unlikely pressurization event (Load Rejection Without Bypass) actually occur with RPT inoperative, the safety limit MCPR would not be violated as long as operating power was limited to 85 percent.

Question 3: Describe or reference the physics startup test program which will be used for the restart of BF-3 for cycle 3 operation.

Response 3: The startup test program that will be conducted at the beginning of unit 3 cycle 3 operation is the startup program that was presented by BWR utilities to the Reactor Safety Branch of the Division of Operating Reactors in a March 29, 1979, meeting. This test program has been submitted by Nebraska Public Power District for Cooper Nuclear Station cycle 5 and was approved by the NRC. These tests are in addition to the technical specification requirements for startup and are attached for your information.

1. CORE LOADING VERIFICATION

I. PURPOSE

The purpose of this test is to visually verify that the core is loaded as intended.

II. DESCRIPTION

An underwater television camera or suitable viewing device will be employed to verify both proper orientation and location of each fuel assembly in the reactor core. At least one independent person must also either participate in performing the verification or review a videotape of the verification prior to startup.

III. CRITERIA AND ACTIONS

The as-loaded core must conform with the referenced core upon which the licensing analysis was performed. Any discrepancies discovered in the loading will be promptly corrected and the affected areas re-verified to be properly loaded prior to startup.

Conformance to the reference loading will be demonstrated by a permanent core serial number map, and documented by the signatures of the verifiers.

2. CONTROL ROD OPERABILITY AND SUBSCRITICALITY CHECK

I. PURPOSE

This test is performed to ensure that no gross local reactivity irregularities exist and that all operable control rods are functioning properly.

II. DESCRIPTION

The control rod mobility test will be performed after the four bundles surrounding the given control rod are loaded. The subcriticality check will be performed after the core loading has been completed. The control rod mobility check may be performed concurrent with the subcriticality check after core loading has been completed. Performance of this test will provide assurance that criticality will not occur due to the withdrawal of a single rod. Each control rod in the core will be withdrawn and inserted one at a time to ensure its mobility with drive pressure. Also, the nuclear instrumentation will be monitored during the movement of each control rod to verify subcriticality.

III. CRITERIA AND ACTIONS

For those control rods that will not move under drive pressure, appropriate repairs or adjustments will be made or the rod will be declared inoperable as described in the technical specifications. If criticality were to be achieved by the withdrawal of a single control rod, the control rod would be inserted and all further rod movements would cease and an investigation would be conducted to determine the cause.

3. TIP SIGNAL UNCERTAINTY TEST

I. PURPOSE

The purpose of this test is to determine the Traversing In-Core Probe (TIP) system total uncertainty using a statistical analysis.

II. DESCRIPTION

Total TIP signal uncertainty consists of geometric and random noise components. Data to perform the analysis is obtained at intermediate power levels and/or power levels greater than 75% with the reactor operating at steady state in an octant symmetric rod pattern (if possible). This data will be additionally used to perform a gross TIP symmetry check, which is a comparison of integrated readings from symmetrically located TIP's.

III. CRITERIA AND ACTIONS

A. The total TIP signal uncertainty (random noise plus geometric uncertainties) obtained by averaging these uncertainties for all data sets should be less than 9%. A minimum of two or up to six data sets may be used to meet the above criterion. If the 9% criterion is not met and the calculations have been rechecked, the calibration of TIP system (e.g. axial alignment) shall be checked. It may be necessary to omit data pairs from the analysis if exact octant symmetry is not attainable in fuel loading or control rod patterns. In such cases, offline code predictions of exposure or control rod induced asymmetry may prove useful in explaining the uncertainty.

B. The gross check of TIP signal symmetry should yield a maximum deviation between symmetrically located pairs of less than 25%. If the criterion cannot be met, the cause of the asymmetry must be investigated and an explanation attempted as per Criterion A.

Question 4: The staff stated in Section 6.2.2 of its safety evaluation of the Generic Reload Fuel Application (which you have referenced in your reload application) that "Additional data should be submitted by GE to the staff for review, to justify the conservatism of the GEXL correlation for the second and subsequent cycles of operation of the retrofit 8x8 bundles, when local peaking factors may increase sufficiently to cause non-conservative CPR calculations." Your reload submittal has not addressed this issue. Accordingly, we request you provide either directly or through reference, adequate information which speaks to this concern. Your response should include:

- The extent to which individual heater rods are instrumented in steady-state critical power tests for the retrofit fuel design.
- For each test bundle provide measured and predicted results in tabular form for the various test conditions.
- Provide trend plots (measured critical power/predicted critical power vs h_{IN}, G, P, critical power, test bundle)
- Maximum R-Factor for each test bundle (new and old R-Factor definitions)
- Thermocouple locations (rod-by-rod axially)
- Spacer-grid locations
- Provide power and heat flux for all plots of transient CPR cases.

Response 4: This question was asked during the review of the Cooper Nuclear Station Unit 1 Reload 4. General Electric provided a generic response to this question in Reference 4-1. This referenced letter supplied additional information similar to that given in the approved GETAB Licensing Topical Reports NEDO-10958-A and NEDE-10958-P-A, and also demonstrated that the additive constants used in the GEXL correlation for the 8x8R fuel design was conservatively derived using the methods approved in these two documents. The NRC Safety Evaluation Report for the Cooper Reload 4 (Reference 4-2) concluded that ". . . when viewed over its range of applicability, the 8x8R GEXL correlation (with new additive constants) has somewhat better precision in predicting 8x8R critical bundle powers than the 7x7 and 8x8 GEXL formulations are for predicting 7x7 and 8x8 critical bundle powers, respectively. Furthermore, from these results it may also be concluded that the 3.6 percent standard deviation and best estimate assumption of the GEXL correlation (which were actually used in the GETAB statistical analysis to derive the 1.07 safety limit MCPR) bound the

statistical characteristics associated with the subject 8x8R GEXL correlation." Therefore, the conservatism of GETAB for the 8x8R fuel design based on the previously approved licensing topical report has been recognized by the NRC staff.

Reference 4-1

Letter, Ronald Engel to Darrell G. Eisenhut and Robert L. Tedesco, "Additional Information, 8x8R Fuel GETAB R-Factors," March 30, 1979.

Reference 4-2

"Safety Evaluation by the Office of Nuclear Reactor Regulation support Amendment No. 55 to Facility Licensing No. DPR-46, Nebraska Public Power District, Cooper Nuclear Station, Docket No. 50-298," April 27, 1979.