TENNESSEE VALLEY AUTHORITY
CHATTANOOGA, TENNESSEE 37401
400 Chestnut Street Tower II

May 2, 1984

Director of Nuclear Reactor Regulation
Attention: Ms. E. Adensam, Chief
Licensing Branch No. 4
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Ms. Adensam:

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| In the Matter of the Application of | ) | Docket Nos. 50-390 |
| Tennessee Valley Authority | ) | 50-391 |

By your letter dated March 9, 1984 to H. G. Parris, TVA was requested to provide justification for deletion of startup tests 3.9, 4.1, and 4.10 from Watts Bar Nuclear Plant (WBN) Final Safety Analysis Report (FSAR) Table 14.2-2A in Amendment 49. Enclosed is a response related to each of these startup tests in addition to discussion of additional changes to FSAR Table 14.2-2A.

If you have any questions concerning this matter, please get in touch with D. P. Ormsby at FTS 858-2682.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

D S Kammer

D. S. Kammer
Nuclear Engineer

Sworn to and subscribed before me
this 2nd day of May 1984
Paulette H. White
Notary Public
My Commission Expires 9-5-84

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

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ENCLOSURE

WATTS BAR NUCLEAR PLANT UNITS 1 AND 2
INITIAL TEST PROGRAM

1. SU 3.9 - Will be reinstated in Table 14.2-2A.

2. SU 4.1 - Previous measurements of power coefficients at other Westinghouse plants such as Sequoyah units 1 and 2 and McGuire unit 1 have at best been estimates of the power coefficient. The reactivity change as measured on a reactivity computer is not attributable to one specific factor since the reactivity change measured is caused by a combination of isothermal temperature coefficient, moderator temperature coefficient, Doppler, Xenon, and rod movement. Since this measurement of the power coefficient is not a true measurement, and the results of the measurement would not verify design values, we do not believe this test should be required. This test is also time consuming when performed at all various testing plateaus. This test was not performed during initial startup of McGuire unit 2. However, we do plan on performing a power coefficient verification factor measurement at one power level for comparison to a design verification factor. This will be reinstated in Table 14.2-2A.

3. SU 4.10 - Reference: Letter from C. O. Thomas, NRC, to E. P. Rahe, Westinghouse Electric Corporation, dated March 31, 1983, "Acceptance for Referencing of Licensing Topical Report WCAP-10297(P), WCAP-10298 - (NS-EPR-2545) Entitled "Dropped Rod Methodology for Negative Flux Rate Trip Plants."

Westinghouse Electric Corporation (W) identified to TVA in November 1979, a concern with regard to certain assumptions employed in the dropped rod safety analysis. The concern came primarily from the potential for an unanalyzed power overshoot while in automatic rod control following selected dropped rod events without immediate reactor trip.

This item was a significant deficiency under 10 CFR 50.55(e). W recommended plant operating restrictions for reactor controls as an interim measure which would keep our safety analysis valid while a long-term solution was determined.

The proposed interim solution was discussed at a November 19, 1979 meeting with NRC and involved changes in plant operating procedures. The calculated consequences for this event were dependent upon whether the reactor was being operated in an automatic or manual mode. The concern was

limited to reactor operation in the automatic mode. The analysis in the SARs for the rod drop event with the reactor in a manual mode remained valid. This analysis indicated that the DNB limit was not exceeded. If a rod drop event occurred when the reactor was in the automatic mode, the reactor control system would respond to both the reactor power drop (mismatch between turbine power and reactor power) and the decrease in the core average temperature and will attempt to restore both quantities to their original values. This restoration of reactor power by the reactor control system might result in some power overshoot depending upon the excore power signal that was used; therefore, the simple and straightforward way to prevent power overshoot was to either operate the reactor in manual rod control or limit the potential overshoot by restricting rod insertion at high power levels.

The proposed changes were as follows:

1. In manual mode of reactor coolant from 0- to 100-percent power, there is no change from current procedures.

2. In automatic mode of reactor control from 0- to 90-percent power, there is no change from current procedures.

3. In automatic mode of reactor control above 90-percent of reactor power, control bank D must be withdrawn greater than or equal to 215 steps.

By implementing these changes, a dropped rod event during automatic rod control would not result in an overshoot above rated thermal power. For power levels greater than or equal to 90 percent, a dropped rod event would result in a withdrawal demand from the rod control system. Since differential rod worth of the D bank while above 215 steps is negligible, the reactivity required for a power overshoot following a rod drop is not available. For rod drops below 90-percent power, analysis has been performed to show that the reactor will not overshoot above rated thermal power and thus the DNB design basis is met. The above procedures resulted in no overshoot for a dropped rod event.

The negative rate reactor trip is intertwined with this issue. It was thought that this trip was needed to prevent an unanalyzed power overshoot, but this was before W had instituted the interim operating restrictions or completed its long-term evaluation. Initially, W thought the plants with negative rate trip circuitry might not experience a reactor trip as a consequence of rod drop due to a reduction of conservatism in the error allowances and the application of a more conservative core physics model than previously utilized.

W has n[illegible]completed its long-term evaluat[illegible]n, WCAP-1029[illegible] (P), "Dropped Rod Methodology [illegible]or Negative Flux Rate Trip Plant," and notified NRC of its conclusion that, based on a considerable quantity of work, the interim restrictions on operation above 90-percent power could be removed. NRC has completed their review of the W report and agrees the interim operating restrictions are no longer needed (See March 31, 1983 reference) and, in fact, has removed these restrictions on Sequoyah. Additional test data was provided by letter NS-EPR-2895 dated March 14, 1984 from E. P. Rahe to C. H. Berlinger.

The analytical work (WCAP-10297) W has completed on the rod drop issue also provides the following technical justification for the deletion of SU 4.10, since (1) the Watts Bar technical specifications use the generic nominal negative flux rate trip setpoint of 5-percent rated thermal power in two seconds and this will result in a reactor trip for dropped rod worths in excess of the minimum detectable value; (2) the methodology developed in WCAP-10297 provides a means by which to evaluate departure from nucleate boiling (DNB) for dropped rod events which do not result in reactor trip; and (3) successful plant/cycle specific application of the WCAP-10297 methodology will be completed prior to Watts Bar startup. This will be sufficient to confirm that the DNB design basis is met for all dropped rod events initiated from full power. In addition, the rod drop plant trip test has been successfully performed on W. B. McGuire unit 1. WBN FSAR table 4.1-1 presents a comparison of the nuclear, thermal-hydraulic, and mechanical design parameters between WBN units 1 and 2 and W. B. McGuire units 1 and 2, and they are very similar. Further, the core loading pattern and rod control pattern are identical for WBN and W. B. McGuire. Also, all the test objectives of the rod drop plant trip test are met by either required surveillance testing or other startup tests.

The results of a plant specific analysis will be provided as soon as the W evaluation is complete.

4. SU-3.8C, SU-4.10A, and SU-4.11 - These instructions were previously portions of test sequence documents. They have been removed from the sequence documents and made into individual tests without changing the scopes or methods.

5. SU-3.1 and SU-2.5 - These tests were incorporated into SU-3.2 and SU-2.1 respectively to provide better ease of test conduct by staying in one test instruction. No test objective or methods have been changed.

6. SU-3.4, SU-3.7, and SU-4.2 - These tests have been incorporated into PORC reviewed technical instructions. SU-3.4 requirements have been incorporated into TI-7 (Reactivity Computer Checkout) and SU-3.7 and SU-4.2 methods have been incorporated into TI-41 (Incore Flux Mapping).

7. SU-4.3 - Added back to SU Test Program to be performed at the [illegible] and 50-percent power plateaus.

8. Figure 14.2-3A - Changes on Figure 14.2-3A were made to reflect the changes made to Table 14.2-2A.