TENNESSEE VALLEY AUTHORITY
CHATTANOOGA, TENNESSEE 37401
500C Chestnut Street Tower II

MAR 16 1979

Director of Nuclear Reactor Regulations
Attention: Mr. S. A. Varga, Chief
Light Water Reactors Branch No. 4
Division of Project Management
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Varga:

In the Matter of the Application of	)	Docket Nos. 50-327
Tennessee Valley Authority	)	50-328

Enclosed is TVA's revised response to question 2 of your September 20, 1978, request for additional information. Our original response was transmitted to you in my January 26, 1979, letter. This completes our response to the Sequoyah Nuclear Plant Safety Evaluation Report outstanding issue item 5.

Very truly yours,

for J. E. Gilleland
Assistant Manager of Power

Enclosure

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Enclosure

Response to QAB Question 2 of September 20, 1978, Letter

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With regard to the initial test program, our review of the test descriptions modified in amendment 48 disclosed that the description of SU-9.4, Plant Trip from 100% Power, is not sufficiently descriptive to conclude that satisfactory acceptance criteria have been established. Modify the test abstract to: (1) identify the variables or parameters to be monitored for each trip, (2) provide assurance that test results will be compared with predicted results for each trip, and (3) provide quantitative acceptance criteria and their bases for the required degree of convergence of actual test results with predicted results for the monitored variables and parameters for each trip.

Response

The following plant parameters represent the minimum monitoring requirements for each trip test: RCS average temperature, one RCS loop hot leg temperature, one RCS loop hot leg to cold by temperature difference, pressurizer level, pressurizer pressure, nuclear power, one steam generator level, one steam generator feedwater flow, steam header pressure, and turbine speed.

Test results acceptance criteria and their bases are provided below for SU-9.4, Plant Trip from 100% Power.

SU-9.4A, Turbine and Reactor Trip from 100% Power

(1) Pressurizer and steam generator safety valves shall not lift and Safety Injection shall not be initiated.

Control system functioning should provide smooth response to transients and avoid unnecessary actuation of protection system components. This is basically an operational concern, since actuation of the Engineered Safety Features Actuation System (ESFAS) results in significantly increased plant downtime. The pressurizer safety valves, steam generator safety valves, and the Safety Injection system are all parts of ESFAS and thus are not expected to be actuated.

(2) Measured overall hot leg RTD response time shall not exceed the expected maximum response time.

The expected maximum hot leg RTD time response is based on a computer simulation of reactor trip from full power. The model accounts for FSAR Chapter 15 assumptions of transport delay, heatup time, and RTD response time. The time response is calculated on the same basis on which the test is performed. This test verifies that the thermal hydraulic characteristics of the RCS are at least as good as those assumed in the FSAR. The Reactor Protection System Time Response Test measures the time response of the electronics from the RTD through the gripper coils. Thus, the two tests verify the conservatism of the FSAR Chapter 15 assumed channel response time.

(3) Nuclear power shall decrease to 15% within 2.5 seconds after turbine trip.

This criteria gives the expected nuclear power versus time for the RTD time response test. If the power decrease was slower, there would be a corresponding slower rate of change in RCS hot leg temperature. This has the effect of slowing RTD time response. Thus, this criterion is a benchmark to determine whether a valid test of RTD time response exists.

(4) All full-length control rods shall release and drop.

As in virtually all safety analyses, it is assumed that the control rods trip when required, thus this criteria.

SU-9.4B, Net Load Loss from 100% Power

(1) See Acceptance Criterion (1) above for SU-9.4A.

(2) Turbine speed shall not exceed the overspeed trip setpoint.

Proper operation of the turbine control system precludes a turbine trip on overspeed, thus this acceptance criterion.

Test results review criteria and their bases are provided below for SU-9.4, Plant Trip from 100% Power. These review criteria are derived primarily from control system studies and characterize expected plant response beyond that required for overall test acceptance by the criteria presented above. Failure to satisfy a review criteria does not necessarily imply improper control system operation, but will require an evaluation of control system settings and appropriateness of review criteria to actual test conditions for satisfactory resolution.

SU-9.4A, Turbine and Reactor Trip from 100% Power

(1) Pressurizer level should not decrease below 20% of level span and pressurizer pressure should not decrease below 1950 psig.

The basis for these criteria is to monitor the pressurizer level and pressurizer pressure control systems. If these systems operate as designed, these minimum level and pressure values should not be exceeded. If the plant goes below these minimum values, Safety Injection actuation on low pressurizer level and low pressurizer pressure can occur. If the level decreases below 20% of span, heater cutout on low-low level or letdown isolation can occur.

(2) Feedwater flow should go to full shutoff prior to RCS average temperature reaching its no load value of 547° F, RCS average temperature should steady out at or above its no load value without manual intervention on feedwater flow, and the steam dump valves should trip open, modulate closed and not cycle repeatedly.

These criteria describe expected or desirable plant behavior. The feedwater control system is designed to close the feedwater control valves following a reactor trip when RCS average temperature decreases below 554° F. Thus, full shutoff prior to reaching a no load RCS average temperature is expected. The criteria on RCS average temperature is based on the expected behavior of the feedwater control system. If a large mismatch of feedwater flow above steam flow exists on the order of several minutes, excessive cooldown of the RCS would occur. The steam dump valves are controlled on temperature error between measured RCS average temperature and its no load value of 547° F. These valves should open rapidly to minimize peak steam pressure and then modulate closed to reduce RCS average temperature to its no load value. If the gains on the steam dump control system were too large, the valves would step open and closed and result in oscillations in RCS average temperature. A properly tuned steam dump system results in the valves modulating closed providing a smooth approach to no load RCS average temperature.

(3) The generator should remain connected to the grid for approximately 30 seconds following turbine trip.

The 30-second time delay is based on demonstrating the loss of flow protection via delayed fast bus transfer. This is allowed if turbine-generator protective functions such as bearing failure indication or generator electrical faults do not occur.

SU-9.4B, Net Load Loss from 100% Power

(1) Pressurizer pressure should not increase above 2350 psig.

Sequoyah Nuclear Plant has no direct turbine trip on opening of the generator breakers. Therefore, a reactor trip would be generated after some time delay from overpower ΔT, overtemperature ΔT, or low steam generator level. Because of this delay to reactor trip, the power mismatch between the RCS and the turbine-generator causes energy storage in the RCS as evidenced by an increase in RCS average temperature. Because of this increase, the coolant expands and the steam space in the pressurizer is compressed. The ensuing increase in pressurizer pressure is suppressed by pressurizer spray and relief valves.

(2) After reactor trip occurs, the behavior of the plant is similar to that of SU-9.4A, hence review criteria (1) and (2) above of SU-9.4A will be applied to SU-9.4B.

For economic reasons, TVA is currently investigating the feasibility of performing one plant trip test from full power which would meet the test objectives of both SU-9.4A and SU-9.4B. If this is determined to be acceptable, the acceptance and review criteria will be appropriately modified.