September 15, 19

Mr. Oliver D. Kingsl , Jr.
President, TVA Nucle and
Chief Nuclear Officer
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: REVISION TO THE TECHNICAL SPECIFICATION BASES
SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2

Dear Mr. Kingsley:

By letter dated September 8, 1995, the Tennessee Valley Authority (TVA) informed the NRC of changes (designated by TVA as Revisions BR-6 for Unit 1, BR-7 and BR-8 for Unit 2) that have been made by TVA to the Bases of the Sequoyah Nuclear Plant Units 1 and 2 Technical Specifications (TS). The changes designated BR-6 and BR-8 are being made to Bases Section 3/4.7.8 to clarify the intent of the vacuum relief flow indication when the auxiliary building gas treatment system is tested in accordance with Surveillance Requirement 4.7.8.d.3. The change designated BR-7 is being made to Bases Section 4.7.1.2 to indicate the current operation of the turbine driven auxiliary feedwater level control valves that were modified during the Unit 2 Cycle 6 refueling outage. They are now non-modulating valves that open automatically upon receipt of an accident signal.

The purpose of this letter is to distribute the attached revised TS pages to the appropriate TS manual holders.

Sincerely,
ORIGINAL SIGNED BY:

David E. LaBarge, Sr. Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 50-328

Enclosures: 1. Revised Unit 1 TS page
B3/4 7-5
2. Revised Unit 2 TS pages
B3/4 7-2 and B3/4 7-5

cc: See next page

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ENCLOSURE 1

SEQUOYAH NUCLEAR PLANT UNIT 1

TECHNICAL SPECIFICATIONS PAGE B3/4 7-5

3/4.7.8 AUXILIARY BUILDING GAS TREATMENT SYSTEM

The OPERABILITY of the auxiliary building gas treatment system ensures that radioactive materials leaking from the ECCS equipment following a LOCA are filtered prior to reaching the environment. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the accident analyses. ANSI N510-1975 will be used as a procedural guide for surveillance testing. Cumulative operation of the system with the heaters on for 10 hours over a 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

The minimum vacuum relief flow requirement in TS Surveillance Requirement 4.7.8.d.3 is for test purposes only. It is intended to demonstrate an acceptable level of ABGTS performance margin by simulating an ABSCE boundary breach. The inability to meet the specified minimum test condition under other circumstances does not challenge the operability of the ABGTS. BR-6

3/4.7.9 SNUBBERS

Snubbers are designed to prevent unrestrained pipe or component motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping or components as a result of a seismic or other event initiating dynamic loads. It is therefore required that all snubbers required to protect the primary coolant system or any other safety system or component be operable during reactor operation.

Because the snubber protection is required only during relatively low probability events, a period of 72 hours is allowed to replace or restore the inoperable snubber(s) to operable status and perform an engineering evaluation on the supported component or declare the supported system inoperable and follow the appropriate limiting condition for operation statement for that system. The engineering evaluation is performed to determine whether the mode of failure of the snubber has adversely affected any safety-related component or system. R16

Safety-related snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation, adequate fluid level if applicable, and attachment of the snubber to its anchorage. The removal of insulation or the verification of torque values for threaded fasteners is not required for visual inspections.

The inspection frequency is based upon maintaining a constant level of snubber protection. Thus, the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during a required inspection determines the time interval for the next required inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25 percent) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

ENCLOSURE 2

SEQUOYAH NUCLEAR PLANT UNIT 2

TECHNICAL SPECIFICATIONS PAGES B3/4 7-2 AND B3/4 7-5

R187

Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat), Mwt

K = Conversion factor, 947.82 $\frac{\text{(Btu/sec)}}{\text{Mwt}}$

w_s = Minimum total steam flow rate capability of the operable MSSVs on any one steam generator at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, in lb/sec. For example, if the maximum number of inoperable MSSVs on any one steam generator is one, then w_s should be a summation of the capability of the operable MSSVs at the highest capacity MSSV operating pressure, excluding the highest capacity MSSV. If the maximum number of inoperable MSSVs per steam generator is three then w_s should be a summation of the capacity of the operable MSSVs at the highest operable MSSV operating pressure, excluding the three highest capacity MSSVs.

h_{fg} = heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, Btu/lbm

N = Number of loops in plant

The valves calculated from this algorithm must then be adjusted lower to account for instrument and channel uncertainties.

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of off-site power.

The steam driven auxiliary feedwater pump is capable of delivering 880 gpm (total feedwater flow) and each of the electric driven auxiliary feedwater pumps are capable of delivering 440 gpm (total feedwater flow) to the entrance of the steam generators at steam generator pressures of 1100 psia. At 1100 psia the open steam generator safety valve(s) are capable of relieving at least 11% of nominal steam flow. A total feedwater flow of 440 gpm at pressures of 1100 psia is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F where the Residual Heat Removal System may be placed into operation. The surveillance test values ensure that each pump will provide at least 440 gpm plus pump recirculation flow against a steam generator pressure of 1100 psia.

Each motor-driven auxiliary feedwater pump (one Train A and one Train B) supplies flow paths to two steam generators. Each flow path contains an automatic air-operated level control valve (LCV). The LCVs have the same train designation as the associated pump and are provided trained air. The turbine-driven auxiliary feedwater pump supplies flow paths to all four steam generators. Each of these flow paths contains an automatic opening (non-modulating) air-operated LCV, two of BR-1 BR-7

3/4.7.8 AUXILIARY BUILDING GAS TREATMENT SYSTEM

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The minimum vacuum relief flow requirement in TS Surveillance Requirement 4.7.8.d.3 is for test purposes only. It is intended to demonstrate an acceptable level of ABGTS performance margin by simulating an ABSCE boundary breach. The inability to meet the specified minimum test condition under other circumstances does not challenge the operability of the ABGTS. BR-8

3/4.7.9 SNUBBERS

Snubbers are designed to prevent unrestrained pipe or component motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping or components as a result of a seismic or other event initiating dynamic loads. It is therefore required that all snubbers required to protect the primary coolant system or any other safety system or component be operable during reactor operation.

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Safety-related snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation, adequate fluid level if applicable, and attachment of the snubber to its anchorage. The removal of insulation or the verification of torque values for threaded fasteners is not required for visual inspections.

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Mr. Oliver D. Kingsley, Jr.
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SEQUOYAH NUCLEAR PLANT

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