ENCLOSURE 1
PROPOSED TECHNICAL SPECIFICATION CHANGES
SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2
(TVA SQN TS 68)

PROPOSED CHANGES TO DELETE TABLE 3.4-1
FROM THE SEQUOYAH TECHNICAL SPECIFICATIONS

LIST OF EFFECTIVE PAGES

UNIT 1

3/4 4-14
3/4 4-15
3/4 4-15a

UNIT 2

3/4 4-18
3/4 4-19
3/4 4-20

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 1 GPM total primary-to-secondary leakage through all steam generators and 500 gallons per day through any one steam generator,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 40 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 $\pm$ 20 psig.

f. 1 GPM leakage at a Reactor Coolant System pressure of 2235 $\pm$ 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in ~~Table 3.4-1.~~ appropriate plant instructions.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

SURVEILLANCE REQUIREMENTS (Continued)

a. Monitoring the lower containment atmosphere particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment pocket sump inventory and discharge at least once per 12 hours.

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 $\pm$ 20 psig at least once per 31 days with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into Mode 3 or 4.

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours.

e. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified by LCO 3.4.6.2.f ~~in Table 3.4-1~~ shall be demonstrated OPERABLE pursuant to Specification 4.0.5, except that in lieu of any leakage testing requirements required by Specification 4.0.5, each valve shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months.

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months.

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve.

d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

DELETE

VALVE NUMBER	FUNCTION
63-560	Accumulator Discharge
63-561	Accumulator Discharge
63-562	Accumulator Discharge
63-563	Accumulator Discharge
63-622	Accumulator Discharge
63-623	Accumulator Discharge
63-624	Accumulator Discharge
63-625	Accumulator Discharge
63-551	Safety Injection (Cold Leg)
63-553	Safety Injection (Cold Leg)
63-557	Safety Injection (Cold Leg)
63-555	Safety Injection (Cold Leg)
63-632	Residual Heat Removal (Cold Leg)
63-633	Residual Heat Removal (Cold Leg)
63-634	Residual Heat Removal (Cold Leg)
63-635	Residual Heat Removal (Cold Leg)
63-641	Residual Heat Removal/Safety Injection (Hot Leg)
63-644	Residual Heat Removal/Safety Injection (Hot Leg)
63-558	Safety Injection (Hot Leg)
63-559	Safety Injection (Hot Leg)
63-543	Safety Injection (Hot Leg)
63-545	Safety Injection (Hot Leg)
63-547	Safety Injection (Hot Leg)
63-549	Safety Injection (Hot Leg)
63-640	Residual Heat Removal (Hot Leg)
63-643	Residual Heat Removal (Hot Leg)
87-558	Upper Head Injection
87-559	Upper Head Injection
87-560	Upper Head Injection
87-561	Upper Head Injection
87-562	Upper Head Injection
87-563	Upper Head Injection
FCV-74-1*	Residual Heat Removal
FCV-74-2*	Residual Heat Removal

*These valves do not have to be leak tested following manual or automatic actuation or flow through the valve.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 1 GPM total primary-to-secondary leakage through all steam generators and 500 gallons per day through any one steam generator,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 40 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig.

f. 1 GPM leakage at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in ~~Table 3.4-1.~~ appropriate plant instructions.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

a. Monitoring the lower containment atmosphere particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment pocket sump inventory and discharge at least once per 12 hours.

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig at least once per 31 days with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into Mode 3 or 4.

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours.

e. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified by L.C.O. 3.4.6.2.f ~~in Table 3.4-1~~ shall be demonstrated OPERABLE pursuant to Specification 4.0.5, except that in lieu of any leakage testing requirements required by Specification 4.0.5, each valve shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months.

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months.

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve.

d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

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DELETE

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	FUNCTION
63-560	Accumulator Discharge
63-561	Accumulator Discharge
63-562	Accumulator Discharge
63-563	Accumulator Discharge
63-622	Accumulator Discharge
63-623	Accumulator Discharge
63-624	Accumulator Discharge
63-625	Accumulator Discharge
63-551	Safety Injection (Cold Leg)
63-553	Safety Injection (Cold Leg)
63-557	Safety Injection (Cold Leg)
63-555	Safety Injection (Cold Leg)
63-632	Residual Heat Removal (Cold Leg)
63-633	Residual Heat Removal (Cold Leg)
63-634	Residual Heat Removal (Cold Leg)
63-635	Residual Heat Removal (Cold Leg)
63-641	Residual Heat Removal/Safety Injection (Hot Leg)
63-644	Residual Heat Removal/Safety Injection (Hot Leg)
63-558	Safety Injection (Hot Leg)
63-559	Safety Injection (Hot Leg)
63-543	Safety Injection (Hot Leg)
63-545	Safety Injection (Hot Leg)
63-547	Safety Injection (Hot Leg)
63-549	Safety Injection (Hot Leg)
63-640	Residual Heat Removal (Hot Leg)
63-643	Residual Heat Removal (Hot Leg)
87-558	Upper Head Injection
87-559	Upper Head Injection
87-560	Upper Head Injection
87-561	Upper Head Injection
87-562	Upper Head Injection
87-563	Upper Head Injection
FCV-74-1*	Residual Heat Removal
FCV-74-2*	Residual Heat Removal

~~*These valves do not have to be leak tested following manual or automatic actuation or flow through the valve.~~

ENCLOSURE 2

PROPOSED TECHNICAL SPECIFICATION CHANGES

SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2

(TVA SQN TS 68)

JUSTIFICATION FOR DELETION OF TABLE 3 4-1

FROM SEQUOYAH TECHNICAL SPECIFICATIONS

Description of Change

This proposed change would delete Table 3.4-1 of LCO 3.4.6.2, listing pressure isolation valves, from the technical specifications for both units 1 and 2. This table would be placed in the appropriate plant instructions to ensure proper control.

Reason for Change

NRC pointed out in the referenced safety evaluation report (SER) the need for inclusion of valves FCV-87-7 and FCV-87-8 to both units 1 and 2 technical specifications. This SER was a written response to TVA submittals describing Sequoyah's Inservice Test Program (IST). These valves are located in a line connecting the liquid waste disposal system to the upper head injection system (UHI). The purpose of this line is leakage testing of the UHI check valves. The valves in Table 3.4-1 are required to be operable in order to prevent leakage from the reactor coolant system (RCS) into a lower pressure system. The inoperability of these valves creates the potential of an intersystem loss of coolant accident (LOCA).

Justification for Change

By the referenced SER, TVA became aware of the need to add the two UHI valves (FCV-87-7 and FCV-87-8) to the list of pressure isolation valves. In order to add these valves to this list as currently configured, a change to technical specifications is required. Sequoyah has recently received a change to this table for unit 1 technical specifications (Amendment 39 to license No. DPR-77, June 20, 1985).

TVA believes that removing this list from Technical Specifications, and making it a part of plant-controlled instructions, will remove an unnecessary licensing burden from both TVA and NRC.

10CFR50.59 states that a change to a licensed facility or procedures described in the safety analysis may be made without prior Commission approval. However, the change must be evaluated by the criteria of 10CFR50.59 (a)(2)(i),(ii), and (iii) to constitute neither an unreviewed safety question nor a change in technical specifications. Moving the list into plant procedures would allow use of the 10CFR50.59 process in order to make timely changes to the plant. Yet this process would still require an analysis determining the effects of said change upon safe plant operation. Changes to this procedure (other than editorial or clarifying changes) made prior to Commission approval using the criteria of 10CFR50.59(a)(2) will be reported to the Commission at least once annually.

This proposed change will allow changes to the facility and procedures (i.e., adding or deleting valves to or from the list) that do not have an adverse effect on safety without prior Commission approval. Changes which do not satisfactorily meet the criteria of the USQD will require prior Commission approval similar to that received for changes to technical specifications.

TVA believes that this change will fully meet the requirements of safety while reducing the administrative burden of future changes of technical specifications to both TVA and NRC.

The list will be included (as currently configured in Technical Specifications) in the appropriate plant instructions. The two valves referenced previously (FCV-87-7 and FCV-87-8) will be added to this list. These valves are currently being tested by TVA-Sequoyah, as part of our ASME Section XI testing program. With approval of this change, their test interval will be brought into concurrence with the remaining technical specification valves.

A 10CFR50.59 evaluation will be performed addressing the addition of FCV-87-7 and FCV-87-8 to this list. This evaluation will be performed as part of the revision to the appropriate plant instruction adding the list of valves.

References

T. M. Novak's letter to H. G. Parris dated April 5, 1985, "Safety Evaluation Report on Sequoyah Inservice Test Program for Pumps and Valves (IST)" (A02 850415 008)

ENCLOSURE 3

TVA SQN TS 68
DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS
FOR DELETION OF TABLE 3.4-1 FROM THE SEQUOYAH
TECHNICAL SPECIFICATIONS

SIGNIFICANT HAZARDS CONSIDERATIONS

1. Is the probability of an occurrence or the consequences of an accident previously evaluated in the safety analysis report significantly increased?

   No--This proposed change is primarily a change to the method of reporting amendments to the list of pressure isolation valves. This proposed change to technical specifications does not reflect a change to plant design, configuration, or testing requirements. Any changes to this list will be subjected to an unreviewed safety question determination (USQD) per the criterion of 10CFR50.59. Changes potentially adverse to plant safety will be determined as unreviewed safety questions (USQ) and require Commission review and approval prior to implementation. Changes satisfactorily meeting the criterion of the USQD will be made and reported to the Commission in an annual report.

2. Is the possibility for an accident of a new or different type than evaluated previously in the safety analysis report created?

   No--This proposed change does not create a possibility for accidents not previously evaluated in the safety analysis because it does not change the current plant configuration of existing euqipment. The possibility of adding or deleting equipment from this list causing a different type of accident would be identified by performance of a USQD for each specific change.

3. Is the margin of safety significantly reduced?

   No--The margin of safety remains unaffected solely due to this proposed change since no equipment parameter or test described in the Technical Specifications or their bases are added or deleted by this change. Specific changes to this equipment list will be subject to USQD as they are implemented.