Tennessee Valley Authority, Post Office Box 2000, Soddy-Daisy, Tennessee 37384-2000

November 8, 2001

TVA-SQN-TS-01-06 10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

Gentlemen:

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

SEQUOYAH NUCLEAR PLANT (SQN) - UNITS 1 AND 2 - TECHNICAL SPECIFICATION (TS) CHANGE NO. 01-06, "DELETION OF LICENSE CONDITION 2.H, ADMINISTRATIVE CONTROL SECTION 6.6 AND ASSOCIATED LIMITING CONDITIONS FOR OPERATION, AND ADMINISTRATIVE CONTROL SECTION 6.7"

In accordance with the provisions of 10 CFR 50.90, TVA is submitting a request for an amendment to Sequoyah's Licenses DPR-77 and 79 to change the TSs for Units 1 and 2. The proposed change will delete License Condition 2.H, "Reporting to the Commission"; Administrative Control Section 6.6, "Reportable Event Action"; and Administrative Control Section 6.7, "Safety Limit Violation." As Administrative Control Section 6.6 is referenced in several Limiting Conditions for Operation (LCOs) and associated TS Bases, these LCOs and TS Bases will be modified to remove those references.

The proposed change is administrative in nature and will eliminate notification and reporting requirements from the Facility Operating Licenses and TSs, which are adequately governed by the reporting requirements of 10 CFR 50.72 and 10 CFR 50.73. There are no new commitments contained in this letter.

TVA has determined that there are no significant hazards considerations associated with the proposed change and that the change is exempt from environmental review pursuant to

D030

the provisions of 10 CFR 51.22(c)(9). The SQN Plant Operations Review Committee and the SQN Nuclear Safety Review Board have reviewed this proposed change and determined that operation of SQN Units 1 and 2, in accordance with the proposed change, will not endanger the health and safety of the public. Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter to the Tennessee State Department of Public Health.

Enclosure 1 to this letter provides the description and evaluation of the proposed change. This includes TVA's determination that the proposed change does not involve a significant hazards consideration, and is exempt from environmental review. Enclosure 2 contains copies of the appropriate TS pages from Units 1 and 2 marked up to show the proposed change. Enclosure 3 forwards the revised TS pages for Units 1 and 2 which incorporate the proposed change.

The deletion of License Condition 2.H has a precedent with Beaver Valley Power Station Units 1 and 2 (Facility Operating License Numbers DPR-66 and NPF-73) via Amendments 220 and 97, respectively. Additionally, this proposed change is consistent with NUREG-1431.

TVA requests approval for the change as soon as practical and that it be made effective within 45 days of NRC approval. This letter is being sent in accordance with RIS 2001-05. If you have any questions about this change, please telephone me at (423) 843-7170 or J. D. Smith at (423) 843-6672.

Sincerely,

Pedro Salas
Licensing and Industry Affairs Manager

Subscribed and sworn to before me
on this 8th day of November

Peggy M Billingsley
Notary Public

My Commission Expires October 9, 2002

Enclosures

ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT (SQN)
UNITS 1 AND 2
DOCKET NOS. 327 AND 328

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE NO. 01-06
DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

I. DESCRIPTION OF THE PROPOSED CHANGE

TVA proposes a change that will delete License Condition 2.H, "Reporting to the Commission"; Administrative Control Section 6.6, "Reportable Event Action"; Administrative Control Section 6.7, "Safety Limit Violation," and the definition of "REPORTABLE EVENT" from the Sequoyah Units 1 and 2 licenses. License Condition 2.H encompasses the reporting from TVA to the Commission in regards to site and plant characteristics necessary for the issuance of the original license, and Administrative Control 6.6 involves the reporting from TVA to the Commission in regards to plant conditions in the case of a reportable event and the internal review of that event.

Administrative Control 6.6 contains two TSs, 6.6.1.a and 6.6.1.b. TS 6.6.1.a is referenced throughout the SQN TSs under Limiting Conditions for Operation (LCOs), surveillance requirements (SRs), and in the Bases. TS 6.6.1.b describes an internal administrative SQN event review process involving a Plant Operations Review Committee (PORC) review requirement for Licensee Event Reports (LERs) which is proposed to be deleted.

TS Administrative Control 6.7 covers actions to be taken in the event a safety limit is violated. These actions include placing the unit in hot standby within one hour which is redundant to TS Section 2.1, "Safety Limits," and contains reporting requirements that are redundant to 10 CFR 50.72 and 10 CFR 50.73. The proposed change will delete Administrative Control 6.7.

II. REASON FOR THE PROPOSED CHANGE

The proposed change will eliminate notification, reporting, and review requirements from the Facility Operating Licenses and TSs, which are adequately governed by the reporting requirements of 10 CFR 50.72, 10 CFR 50.73, and

the TVA Nuclear Quality Assurance Plan. As such, resources required by both TVA and NRC will be averted.

III. SAFETY ANALYSIS

License Condition 2.H, "Reporting to the Commission," provides for the reporting from TVA to the Commission any violations of the requirements contained in License Conditions 2.C(3) through 2.C(24), 2.E, 2.F, and 2.G for Unit 1 and 2.C(3) through 2.C(16), 2.E, 2.F, and 2.G for Unit 2. Most of these conditions were dated conditions that were satisfied and the reporting on those conditions are no longer applicable. License Condition 2.H requires that TVA report any violations of the following requirements within 24 hours and confirm no later than the first working day following the violation, and then follow by a written report within 14 days. License Condition 2.H can be deleted for the following reasons:

Unit 1 - Condition 2.C(16) and Unit 2 - 2.C(13) - Fire Protection - These conditions are adequately covered by 10 CFR 50.72(b)(3)(v)(A) and the Sequoyah Fire Protection Plan.

Unit 1 - Condition 2.C(22)G(a) - Emergency Preparedness Plan - This condition is a continual process and is adequately covered by 10 CFR 50.73(a)(2)(v)(D) and 10 CFR 50.54(q).

Unit 1 - Condition 2.G - Changes in Effluent Radioactivity Control - This condition is covered adequately by 10 CFR 50.72(b)(3)(v)(C) and 10 CFR 50.73(a)(2)(v)(C).

Units 1 and 2 - Condition 2.E - Physical Protection - This condition is adequately covered by 10 CFR 50.54(p)(1).

Units 1 and 2 - Condition 2.F. - Protection of the Environment condition is adequately covered by 10 CFR 51.20, 10 CFR 51.21, and 10 CFR 51.22.

Administrative Control 6.6, "Reportable Event Action," contains two TSs, 6.6.1.a and 6.6.6.b, both of which may be deleted.

TS 6.6.1.a concerns the reporting of a reportable event to the Commission and is referenced throughout the SQN TSs, specifically:

<u>Units 1 and 2</u>

Reactivity Control Systems - Moderator Temperature Coefficient - LCO 3.1.1.3

Reactor Coolant System - SR 4.4.5.5 and Table 4.4-2 - Steam Generator

Emergency Core Cooling Systems (ECCS) - ECCS Subsystems - LCO 3.5.2

Emergency Core Cooling Systems (ECCS) - ECCS Subsystems - LCO 3.5.3

Containment Systems - Containment Vessel Structural Integrity - SR 4.6.1.6

Containment Systems - Shield Building Structural Integrity - SR 4.6.1.7

Reactor Coolant System - Bases

These references may be deleted for the following reasons:

LCO 3.1.1.3 - LCO Action a.3 states that a report will be made in lieu of a report required by TS 6.6.1.a; therefore, the reference is unnecessary because the reporting requirements of 10 CFR 50.72 and 10 CFR 50.73 will be used as required. Additionally, this change is consistent with NUREG-1431.

SR 4.4.5.5.C and Table 4.4-2 - Steam Generator Tube Inspections - the reporting requirements in the SR and table are adequately covered by the reporting requirements under 10 CFR 50.72 and 10 CFR 50.73 for steam generator tube degradation and structural integrity, as delineated in NUREG-1022, Revision 2.

LCOs 3.5.2 and 3.5.3 - the reporting requirements are adequately covered under 10 CFR 50.72 and 10 CFR 50.73.

SRs 4.6.1.6 and 4.6.1.7 - the SRs are adequately covered by the reporting requirements under 10 CFR 50.72 and 10 CFR 50.73 for degradation of its principal safety barriers.

Since the requirements for event notification and event reporting are delineated in 10 CFR 50.72 and 10 CFR 50.73, TS 6.6.1.a is redundant and should be deleted.

TS 6.6.1.b governs internal SQN reportable event reviews and is being deleted to allow SQN to conform to NUREG-1431. Additionally, PORC and NSRB reviews are delineated in the TVA Nuclear Quality Assurance Plan.

TS 6.7 covers actions to be taken in the event a safety limit is violated. These actions are redundant to TS

Section 2.1, "Safety Limits," and the reporting requirements in 10 CFR 50.72 and 10 CFR 50.73. This change is consistent with NUREG-1431, Revision 2.

Since reporting of reportable events will no longer be in the TS, the definition of "REPORTABLE EVENT" is no longer necessary.

These changes will have no impact on the design, function, or operation of any plant structure, system, or component, either technically or administratively nor will they have a programmatic effect on the TVAN Quality Assurance Program.

IV. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

TVA has concluded that operation of SQN Units 1 and 2, in accordance with the proposed change to the technical specifications (TSs) and operating license, does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c). These changes will delete License Condition 2.H, "Reporting to the Commission," Administrative Control 6.6, "Reportable Event Action," and any references to TS 6.6.1.a, Administrative Control Section 6.7, "Safety Limit Violation," and the definition of "REPORTABLE EVENT" from the TSs for Units 1 and 2.

A. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

These revisions govern the reporting of either site characteristics and past events or of events covered under current NRC regulations and the proposed amendment is administrative in nature. Therefore, it does not increase the probability or consequences of any accident previously evaluated because it does not affect the state of the plant in any physical manner.

B. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment is strictly administrative and does not affect plant equipment or operational procedures. Therefore, it will not create any new or different accidents.

C. **<u>The proposed amendment does not involve a significant reduction in a margin of safety.</u>**

The proposed amendment affects the reporting to the Commission. As such, it does not affect personnel, public, or plant safety. Since the amendment will not affect the plant in a physical manner nor will it affect personnel, public, or plant safety, it will therefore not reduce the margin of safety.

V. ENVIRONMENTAL IMPACT CONSIDERATION

The proposed change does not involve a significant hazards consideration, a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite, or a significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
SEQUOYAH PLANT (SQN)
UNITS 1 AND 2

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE 01-06
MARKED PAGES

I. AFFECTED PAGE LIST

<u>Unit 1</u>
OL 13
Index XVII
1-5
3/4 1-4
3/4 4-10
3/4 4-12
3/4 5-4
3/4 5-8
3/4 6-11
3/4 6-12
B 3/4 4-4a
6-6

<u>Unit 2</u>
OL 12
Index XVII
1-6
3/4 1-4
3/4 4-14b
3/4 4-16
3/4 5-4
3/4 5-8
3/4 6-11
3/4 6-12
B 3/4 4-3b
6-6

II. MARKED PAGES

See attached.

F. This license is subject to the following additional condition for the protection of the environment:

Before engaging in additional construction or operational activities which may result in an environmental impact that was not evaluated by the Commission, Tennessee Valley Authority will prepare and record an environmental evaluation of such activity. When the evaluation indicates that such activity may result in a significant adverse environmental impact that was not evaluated, or that is significantly greater than that evaluated in the Final Environmental Statement prepared by the Tennessee Valley Authority and the Environmental Impact Appraisal prepared by the Commission in May 1979, the Tennessee Valley Authority shall provide a written evaluation of such activities and obtain prior approval from the Director, Office of Nuclear Reactor Regulation.

G. If TVA plans to remove or to make significant changes in the normal operation of equipment that controls the amount of radioactivity in effluents from the Sequoyah Nuclear Plants, the Commission shall be notified in writing regardless of whether the change affects the amount of radioactivity in the effluents.

H. ~~TVA shall report any violations of the requirements contained in Sections 2.C(3) through 2.C(24), 2.E, 2.F, and 2.G of this license within 24 hours by telephone and confirmed by telegram, mailgram, or facsimile transmission to the Director of the Regional Office, or his designate, no later than the first working day following the violation with a written followup report within 14 days.~~ Deleted

I. TVA shall immediately notify the Commission of any accident at this facility which could result in an unplanned release of quantities of fission products in excess of allowable limits for normal operation established by the Commission.

J. TVA shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

K. This amended license is effective as of the date of issuance and shall expire September 17, 2020.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Attachment:
Appendices A and B Technical Specifications

Date of Issuance:
September 17, 1980

December 29, 1988
Amendment No. 93

ADMINISTRATIVE CONTROLS

PRESSURE BOUNDARY LEAKAGE

1.22 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

1.23 DELETED

PURGE - PURGING

1.24 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.25 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.

RATED THERMAL POWER (RTP)

1.26 RATED THERMAL POWER (RTP) shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt.

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

1.27 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its (RTS) trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by NRC.

REPORTABLE EVENT

1.28 ~~A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.~~ DELETED

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the limits specified in the COLR. The maximum upper limit shall be less than 0 delta k/k/°F.

APPLICABILITY: Beginning of cycle life (BOL) limit - MODES 1 and 2* only#
End of life cycle (EOL) limit - MODES 1, 2 and 3 only#

ACTION:

a. With the MTC more positive than the BOL limit specified in the COLR operation in MODES 1 and 2 may proceed provided:

   1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than the BOL limit specified in the COLR within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.

   2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

   ~~3. In lieu of any other report required by Specification 6.6.1, a Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.~~

b. With the MTC more negative than the EOL limit specified in the COLR, be in HOT SHUTDOWN within 12 hours.

*With K_{eff} greater than or equal to 1.0

#See Special Test Exception 3.10.3

SURVEILLANCE REQUIREMENTS (Continued)

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following completion of the inspection. This Special Report shall include:

1. Number and extent of tubes inspected.
2. Location and percent of wall-thickness penetration for each indication of an imperfection.
3. Identification of tubes plugged.

c. ~~Results of steam generator tube inspections which fall into Category C-3 shall be reported pursuant to Specification 6.6.1 prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.~~ Deleted

d. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the staff prior to returning the steam generators to service should any of the following conditions arise:

1. If estimated leakage based on the projected end-of-cycle (or if not practical using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (determined from the licensing basis dose calculation for the postulated main steam line break) for the next operating cycle.

2. If circumferential crack-like indications are detected at the tube support plate intersections.

3. If indications are identified that extend beyond the confines of the tube support plate.

4. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

5. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1×10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G. plug defective tubes and inspect 2S tubes in each other S.G. ~~Prompt notification to NRC pursuant to Specification 6.6.1~~	All other S.G are C-1	None	N/A	N/A
			Some S/Gs C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes ~~Prompt notification to NRC pursuant to Specification 6.6.1.~~	N/A	N/A

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} Greater Than or Equal to 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE safety injection pump,

c. One OPERABLE residual heat removal heat exchanger,

d. One OPERABLE residual heat removal pump, and

e. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a safety injection signal and automatically transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

~~a.~~ With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

~~b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a REPORTABLE EVENT shall be prepared and submitted to the Commission pursuant to Specification 6.6.1. This report shall include a description of the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this report whenever its value exceeds 0.70.~~

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} Less Than 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE residual heat removal heat exchanger,

c. One OPERABLE residual heat removal pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank upon being manually realigned and automatically transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the residual heat removal heat exchanger or residual heat removal pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

~~c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a REPORTABLE EVENT shall be prepared and submitted to the Commission pursuant to Specification 6.6.1. This report shall include a description of the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this report whenever its value exceeds 0.70.~~

CONTAINMENT SYSTEMS

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the containment vessel not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The structural integrity of the containment vessel shall be determined during shutdown by a visual inspection of the exposed accessible interior and exterior surfaces of the vessel. This inspection shall be performed in accordance with the Containment Leakage Rate Test Program to verify no apparent changes in appearance of the surfaces or other abnormal degradation. ~~Any abnormal degradation of the containment vessel detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.6.1.~~

CONTAINMENT SYSTEMS

SHIELD BUILDING STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.7 The structural integrity of the shield building shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.7.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the shield building not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The structural integrity of the shield building shall be determined during the shutdown for each Type A containment leakage rate test (Specification 4.6.1.1.c) by a visual inspection of the exposed accessible interior and exterior surfaces of the shield building and verifying no apparent changes in appearance of the concrete surfaces or other abnormal degradation. ~~Any abnormal degradation of the shield building detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.6.1.~~

Freespan Indication Repair Limits

The tube will be repaired if the crack length outside the dented TSP is ≥40% maximum depth.

Crack Length Limit for ≥40% Maximum Depth

The crack length limit for ≥40% maximum depth indications is defined as 0.375 inch from the centerline of the TSP. This limit defines the edges of the TSP thickness of 0.75 inch for Model 51 S/Gs. It is acceptable for the crack to extend to both edges of the TSP as long as the maximum depth of the crack outside the TSP is <40% maximum depth and the requirements for EOC conditions are acceptable.

Operational Assessment Repair Bases

If the indication satisfies the above maximum depth and length requirements, the repair bases is then obtained by projecting the crack profile to the end of the next operating cycle and determining the burst pressure and leakage for the projected profile. The burst pressure and leakage is compared to the requirements in WCAP 15128, Revision 2, dated February 2000 as supplemented by TVA's letter to NRC dated March 2, 2000. Separate analyses are required for the total crack length and the length outside the TSP due to differences in requirements. If the projected EOC requirements are satisfied, the tube will be left in service.

The results of the condition monitoring and operational assessments will be reported to the NRC within 120 days following completion of the inspection.

~~Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.6.1 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.~~

6.4 TRAINING

6.4.1 DELETED

6.5 REVIEW AND AUDIT

6.5.0 DELETED

6.5.1 PLANT OPERATIONS REVIEW COMMITTEE (PORC) (DELETED)

6.5.1A TECHNICAL REVIEW AND CONTROL (DELETED)

6.5.2 NUCLEAR SAFETY REVIEW BOARD (NSRB) (DELETED)

6.5.3 THIS SPECIFICATION IS DELETED

6.6 REPORTABLE EVENT ACTION (DELETED)

~~6.6.1 The following actions shall be taken for REPORTABLE EVENTS:~~

~~a.~~ ~~The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and~~

~~b.~~ ~~Each REPORTABLE EVENT shall be reviewed by the PORC and the results of this review shall be submitted to the NSRB and the Site Vice President.~~

6.7 SAFETY LIMIT VIOLATION (DELETED)

~~6.7.1 The following actions shall be taken in the event a Safety Limit is violated:~~

~~a.~~ ~~The unit shall be placed in at least HOT STANDBY within one hour.~~

~~b.~~ ~~The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Site Vice President and the NSRB shall be notified within 24 hours.~~

~~c.~~ ~~A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.~~

~~d.~~ ~~The Safety Limit Violation Report shall be submitted to the Commission, the NSRB and the Site Vice President within 14 days of the violation.~~

6.8 PROCEDURES & PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

F. Reactor Safety Methodology Applications Programs (Section 24.0)

TVA will provide a report prepared by the Kaman Sciences Corporation (KSC) on a full scale nuclear safety and availability analysis within six months from the date of the KSC report.

G. This amended license is subject to the following additional condition for the protection of the environment:

Before engaging in additional construction or operational activities which may result in an environmental impact that was not evaluated by the Commission, Tennessee Valley Authority will prepare and record an environmental evaluation of such activity. When the evaluation indicates that such activity may result in a significant adverse environmental impact that was not evaluated, or that is significantly greater than that evaluated in the Final Environmental Statement prepared by the Tennessee Valley Authority and the Environmental Impact Appraisal prepared by the Commission in May 1979, the Tennessee Valley Authority shall provide a written evaluation of such activities and obtain prior approval from the Director, Office of Nuclear Reactor Regulation.

H. ~~TVA shall report any violations of the requirements contained in Sections 2.C(3) through 2.C(16), 2.E, 2.F, and 2.G of this license within 24 hours by telephone and confirmed by telegram, mailgram, or facsimile transmission to the Director of the Regional Office, or his designee, no later than the first working day following the violation with a written followup report within 14 days.~~ Deleted

I. TVA shall immediately notify the Commission of any accident at this facility which could result in an unplanned release of quantities of fission products in excess of allowable limits for normal operation established by the Commission.

J. TVA shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

INDEX

ADMINISTRATIVE CONTROLS

RATED THERMAL POWER (RTP)

1.26 RATED THERMAL POWER (RTP) shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt.

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

1.27 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its (RTS) trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by NRC.

REPORTABLE EVENT

1.28 ~~A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.~~ DELETED

SHIELD BUILDING INTEGRITY

1.29 SHIELD BUILDING INTEGRITY shall exist when:

a. The door in each access opening is closed except when the access opening is being used for normal transit entry and exit.

b. The emergency gas treatment system is OPERABLE.

c. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

SHUTDOWN MARGIN

1.30 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.31 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee (see figure 5.1-1).

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the limits specified in the COLR. The maximum upper limit shall be less than 0 delta k/k/°F.

APPLICABILITY: Beginning of Cycle life (BOL) Limit - Modes 1 and 2* only#
End of Cycle Life (EOL) Limit - Modes 1, 2, and 3 only#

ACTION:

a. With the MTC more positive than the BOL limit specified in the COLR operation in Modes 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than the BOL limit specified in the COLR within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

~~3. In lieu of any other report required by Specification 6.6.1, a Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.~~

b. With the MTC more negative than the EOL limit specified in the COLR be in HOT SHUTDOWN within 12 hours.

* With k_{eff} greater than or equal to 1.0

\# See Special Test Exception 3.10.3

4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. ~~Results of steam generator tube inspections which fall into Category C-3 shall be reported pursuant to Specification 6.6.1 prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.~~ Deleted

d. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the staff prior to returning the steam generators to service should any of the following conditions arise:

1. If estimated leakage based on the projected end-of-cycle (or if not practical using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (determined from the licensing basis dose calculation for the postulated main steam line break) for the next operating cycle.

2. If circumferential crack-like indications are detected at the tube support plate intersections.

3. If indications are identified that extend beyond the confines of the tube support plate.

4. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

5. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1 X 10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G. plug defective tubes and inspect 2S tubes in each other S.G. ~~Prompt notification to NRC pursuant to Specification 6.6.1~~	All other S.G are C-1	None	N/A	N/A
			Some S/Gs C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S/G is C-3	Inspect all tubes in each S.G. and plug defective tubes. ~~Prompt notification to NRC pursuant to Specification 6.6.1.~~	N/A	N/A

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} Greater Than or Equal to 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent emergency core cooling system (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE safety injection pump,

c. One OPERABLE residual heat removal heat exchanger,

d. One OPERABLE residual heat removal pump, and

e. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a safety injection signal and automatically transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

~~a.~~ With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

~~b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a REPORTABLE EVENT shall be prepared and submitted to the Commission pursuant to Specification 6.6.1. This report shall include a description of the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.~~

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE residual heat removal heat exchanger,

c. One OPERABLE residual heat removal pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank upon being manually realigned and automatically transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the residual heat removal heat exchanger or residual heat removal pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

~~c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a REPORTABLE EVENT shall be prepared and submitted to the Commission pursuant to Specification 6.6.1. This report shall include a description of the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.~~

CONTAINMENT SYSTEMS

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the containment vessel not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The structural integrity of the containment vessel shall be determined during shutdown by a visual inspection of the exposed accessible interior and exterior surfaces of the vessel. This inspection shall be performed in accordance with the Containment Leakage Rate Test Program to verify no apparent changes in appearance of the surfaces or other abnormal degradation. ~~Any abnormal degradation of the containment vessel detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.6.1.~~

CONTAINMENT SYSTEMS

SHIELD BUILDING STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.7 The structural integrity of the shield building shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.7.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION.

With the structural integrity of the shield building not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The structural integrity of the shield building shall be determined during the shutdown for each Type A containment leakage rate test (Specification 4.6.1.1.c) by a visual inspection of the exposed accessible interior and exterior surfaces of the shield building and verifying no apparent changes in appearance of the concrete surfaces or other abnormal degradation. ~~Any abnormal degradation of the shield building detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.6.1.~~

results in the lowest burst pressure and the longest length that would tear through-wall at steam-line break conditions. The repair bases for PWSCC at dented TSP intersections is obtained by projecting the crack profile to the end of the next operating cycle and determining if the projected profile meets the requirements of WCAP 15128, Revision 2, dated February 2000 as supplemented by TVA's letter to NRC dated March 2, 2000. The following provides the limits and bases for repair established in the WCAP analyses:

Freespan Indication Repair Limits

The tube will be repaired if the crack length outside the dented TSP is ≥40% maximum depth.

Crack Length Limit for ≥40% Maximum Depth

The crack length limit for ≥40% maximum depth indications is defined as 0.375 inch from the centerline of the TSP. This limit defines the edges of the TSP thickness of 0.75 inch for Model 51 S/Gs. It is acceptable for the crack to extend to both edges of the TSP as long as the maximum depth of the crack outside the TSP is <40% maximum depth and the requirements for EOC conditions are acceptable.

Operational Assessment Repair Bases

If the indication satisfies the above maximum depth and length requirements, the repair bases is then obtained by projecting the crack profile to the end of the next operating cycle and determining the burst pressure and leakage for the projected profile. The burst pressure and leakage is compared to the requirements in WCAP 15128, Revision 2, dated February 2000 as supplemented by TVA's letter to NRC dated March 2, 2000. Separate analyses are required for the total crack length and the length outside the TSP due to differences in requirements. If the projected EOC requirements are satisfied, the tube will be left in service.

The results of the condition monitoring and operational assessments will be reported to the NRC within 120 days following completion of the inspection.

~~Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.6.1 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.~~

6.6 REPORTABLE EVENT ACTION (DELETED)

~~6.6.1 The following actions shall be taken for REPORTABLE EVENTS:~~

~~a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and~~

~~b. Each REPORTABLE EVENT shall be reviewed by the PORC and the results of this review shall be submitted to the NSRB and the Site Vice President.~~

6.7 SAFETY LIMIT VIOLATION (DELETED)

~~6.7.1 The following actions shall be taken in the event a Safety Limit is violated:~~

~~a. The unit shall be placed in at least HOT STANDBY within one hour.~~

~~b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Site Vice President and the NSRB shall be notified within 24 hours.~~

~~c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.~~

~~d. The Safety Limit Violation Report shall be submitted to the Commission, the NSRB and the Site Vice President within l4 days of the violation.~~

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. DELETED

e. DELETED

f. Fire Protection Program implementation.

g. DELETED

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
SEQUOYAH PLANT (SQN)
UNITS 1 AND 2

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE 01-06
REVISED PAGES

I. AFFECTED PAGE LIST

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II. REVISED PAGES

See attached.

F. This license is subject to the following additional condition for the protection of the environment:

Before engaging in additional construction or operational activities which may result in an environmental impact that was not evaluated by the Commission, Tennessee Valley Authority will prepare and record an environmental evaluation of such activity. When the evaluation indicates that such activity may result in a significant adverse environmental impact that was not evaluated, or that is significantly greater than that evaluated in the Final Environmental Statement prepared by the Tennessee Valley Authority and the Environmental Impact Appraisal prepared by the Commission in May 1979, the Tennessee Valley Authority shall provide a written evaluation of such activities and obtain prior approval from the Director, Office of Nuclear Reactor Regulation.

G. If TVA plans to remove or to make significant changes in the normal operation of equipment that controls the amount of radioactivity in effluents from the Sequoyah Nuclear Plants, the Commission shall be notified in writing regardless of whether the change affects the amount of radioactivity in the effluents.

H. Deleted.

I. TVA shall immediately notify the Commission of any accident at this facility which could result in an unplanned release of quantities of fission products in excess of allowable limits for normal operation established by the Commission.

J. TVA shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

K. This amended license is effective as of the date of issuance and shall expire September 17, 2020.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Attachment:
Appendices A and B Technical Specifications

Date of Issuance:
September 17, 1980

Amendment No. 93,

PRESSURE BOUNDARY LEAKAGE

1.22 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

1.23 DELETED

PURGE - PURGING

1.24 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.25 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.

RATED THERMAL POWER (RTP)

1.26 RATED THERMAL POWER (RTP) shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt.

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

1.27 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its (RTS) trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by NRC.

REPORTABLE EVENT

1.28 DELETED

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the limits specified in the COLR. The maximum upper limit shall be less than 0 delta k/k/°F.

APPLICABILITY: Beginning of cycle life (BOL) limit - MODES 1 and 2* only#
End of life cycle (EOL) limit - MODES 1, 2 and 3 only#

ACTION:

a. With the MTC more positive than the BOL limit specified in the COLR operation in MODES 1 and 2 may proceed provided:

   1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than the BOL limit specified in the COLR within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.

   2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the EOL limit specified in the COLR, be in HOT SHUTDOWN within 12 hours.

*With K_{eff} greater than or equal to 1.0

#See Special Test Exception 3.10.3

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following completion of the inspection. This Special Report shall include:

1. Number and extent of tubes inspected.
2. Location and percent of wall-thickness penetration for each indication of an imperfection.
3. Identification of tubes plugged.

c. Deleted.

d. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the staff prior to returning the steam generators to service should any of the following conditions arise:

1. If estimated leakage based on the projected end-of-cycle (or if not practical using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (determined from the licensing basis dose calculation for the postulated main steam line break) for the next operating cycle.

2. If circumferential crack-like indications are detected at the tube support plate intersections.

3. If indications are identified that extend beyond the confines of the tube support plate.

4. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

5. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1×10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

<table>
<tr><th colspan="3">1ST SAMPLE INSPECTION</th><th colspan="2">2ND SAMPLE INSPECTION</th><th colspan="2">3RD SAMPLE INSPECTION</th></tr>
<tr><th>Sample Size</th><th>Result</th><th>Action Required</th><th>Result</th><th>Action Required</th><th>Result</th><th>Action Required</th></tr>
<tr><td rowspan="9">A minimum of S Tubes per S.G.</td><td>C-1</td><td>None</td><td>N/A</td><td>N/A</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="5">C-2</td><td rowspan="5">Plug defective tubes and inspect additional 2S tubes in this S.G.</td><td>C-1</td><td>None</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="3">C-2</td><td rowspan="3">Plug defective tubes and inspect additional 4S tubes in this S.G.</td><td>C-1</td><td>None</td></tr>
<tr><td>C-2</td><td>Plug defective tubes</td></tr>
<tr><td>C-3</td><td>Perform action for C-3 result of first sample</td></tr>
<tr><td>C-3</td><td>Perform action for C-3 result of first sample</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="3">C-3</td><td rowspan="3">Inspect all tubes in this S.G. plug defective tubes and inspect 2S tubes in each other S.G.</td><td>All other S.G are C-1</td><td>None</td><td>N/A</td><td>N/A</td></tr>
<tr><td>Some S/Gs C-2 but no additional S.G. are C-3</td><td>Perform action for C-2 result of second sample</td><td>N/A</td><td>N/A</td></tr>
<tr><td>Additional S.G. is C-3</td><td>Inspect all tubes in each S.G. and plug defective tubes.</td><td>N/A</td><td>N/A</td></tr>
</table>

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} Greater Than or Equal to 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE safety injection pump,

c. One OPERABLE residual heat removal heat exchanger,

d. One OPERABLE residual heat removal pump, and

e. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a safety injection signal and automatically transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} Less Than 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE residual heat removal heat exchanger,

c. One OPERABLE residual heat removal pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank upon being manually realigned and automatically transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the residual heat removal heat exchanger or residual heat removal pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the containment vessel not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The structural integrity of the containment vessel shall be determined during shutdown by a visual inspection of the exposed accessible interior and exterior surfaces of the vessel. This inspection shall be performed in accordance with the Containment Leakage Rate Test Program to verify no apparent changes in appearance of the surfaces or other abnormal degradation.

SHIELD BUILDING STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.7 The structural integrity of the shield building shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.7.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the shield building not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The structural integrity of the shield building shall be determined during the shutdown for each Type A containment leakage rate test (Specification 4.6.1.1.c) by a visual inspection of the exposed accessible interior and exterior surfaces of the shield building and verifying no apparent changes in appearance of the concrete surfaces or other abnormal degradation.

Freespan Indication Repair Limits

The tube will be repaired if the crack length outside the dented TSP is ≥40% maximum depth.

Crack Length Limit for ≥40% Maximum Depth

The crack length limit for ≥40% maximum depth indications is defined as 0.375 inch from the centerline of the TSP. This limit defines the edges of the TSP thickness of 0.75 inch for Model 51 S/Gs. It is acceptable for the crack to extend to both edges of the TSP as long as the maximum depth of the crack outside the TSP is <40% maximum depth and the requirements for EOC conditions are acceptable.

Operational Assessment Repair Bases

If the indication satisfies the above maximum depth and length requirements, the repair bases is then obtained by projecting the crack profile to the end of the next operating cycle and determining the burst pressure and leakage for the projected profile. The burst pressure and leakage is compared to the requirements in WCAP 15128, Revision 2, dated February 2000 as supplemented by TVA's letter to NRC dated March 2, 2000. Separate analyses are required for the total crack length and the length outside the TSP due to differences in requirements. If the projected EOC requirements are satisfied, the tube will be left in service.

The results of the condition monitoring and operational assessments will be reported to the NRC within 120 days following completion of the inspection.

6.4 TRAINING

6.4.1 DELETED

6.5 REVIEW AND AUDIT

6.5.0 DELETED

6.5.1 PLANT OPERATIONS REVIEW COMMITTEE (PORC) (DELETED)

6.5.1A TECHNICAL REVIEW AND CONTROL (DELETED)

6.5.2 NUCLEAR SAFETY REVIEW BOARD (NSRB) (DELETED)

6.5.3 THIS SPECIFICATION IS DELETED

6.6 REPORTABLE EVENT ACTION (DELETED)

6.7 SAFETY LIMIT VIOLATION (DELETED)

6.8 PROCEDURES & PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

F. Reactor Safety Methodology Applications Programs (Section 24.0)

TVA will provide a report prepared by the Kaman Sciences Corporation (KSC) on a full scale nuclear safety and availability analysis within six months from the date of the KSC report.

G. This amended license is subject to the following additional condition for the protection of the environment:

Before engaging in additional construction or operational activities which may result in an environmental impact that was not evaluated by the Commission, Tennessee Valley Authority will prepare and record an environmental evaluation of such activity. When the evaluation indicates that such activity may result in a significant adverse environmental impact that was not evaluated, or that is significantly greater than that evaluated in the Final Environmental Statement prepared by the Tennessee Valley Authority and the Environmental Impact Appraisal prepared by the Commission in May 1979, the Tennessee Valley Authority shall provide a written evaluation of such activities and obtain prior approval from the Director, Office of Nuclear Reactor Regulation.

H. Deleted

I. TVA shall immediately notify the Commission of any accident at this facility which could result in an unplanned release of quantities of fission products in excess of allowable limits for normal operation established by the Commission.

J. TVA shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

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ADMINISTRATIVE CONTROLS

RATED THERMAL POWER (RTP)

1.26 RATED THERMAL POWER (RTP) shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt.

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

1.27 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its (RTS) trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by NRC.

REPORTABLE EVENT

1.28 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

SHIELD BUILDING INTEGRITY

1.29 SHIELD BUILDING INTEGRITY shall exist when:

a. The door in each access opening is closed except when the access opening is being used for normal transit entry and exit.

b. The emergency gas treatment system is OPERABLE.

c. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

SHUTDOWN MARGIN

1.30 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.31 DELETED

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the limits specified in the COLR. The maximum upper limit shall be less than 0 delta k/k/°F.

APPLICABILITY: Beginning of Cycle life (BOL) Limit - Modes 1 and 2* only#
End of Cycle Life (EOL) Limit - Modes 1, 2, and 3 only#

ACTION:

a. With the MTC more positive than the BOL limit specified in the COLR operation in Modes 1 and 2 may proceed provided:

   1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than the BOL limit specified in the COLR within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.

   2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the EOL limit specified in the COLR be in HOT SHUTDOWN within 12 hours.

* With k_{eff} greater than or equal to 1.0

\# See Special Test Exception 3.10.3

4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Deleted.

d. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the staff prior to returning the steam generators to service should any of the following conditions arise:

1. If estimated leakage based on the projected end-of-cycle (or if not practical using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (determined from the licensing basis dose calculation for the postulated main steam line break) for the next operating cycle.

2. If circumferential crack-like indications are detected at the tube support plate intersections.

3. If indications are identified that extend beyond the confines of the tube support plate.

4. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

5. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1 X 10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G. plug defective tubes and inspect 2S tubes in each other S.G.	All other S.G are C-1	None	N/A	N/A
			Some S/Gs C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S/G is C-3	Inspect all tubes in each S.G. and plug defective tubes.	N/A	N/A

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} Greater Than or Equal to 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent emergency core cooling system (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE safety injection pump,

c. One OPERABLE residual heat removal heat exchanger,

d. One OPERABLE residual heat removal pump, and

e. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a safety injection signal and automatically transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE residual heat removal heat exchanger,

c. One OPERABLE residual heat removal pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank upon being manually realigned and automatically transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the residual heat removal heat exchanger or residual heat removal pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the containment vessel not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The structural integrity of the containment vessel shall be determined during shutdown by a visual inspection of the exposed accessible interior and exterior surfaces of the vessel. This inspection shall be performed in accordance with the Containment Leakage Rate Test Program to verify no apparent changes in appearance of the surfaces or other abnormal degradation.

CONTAINMENT SYSTEMS

SHIELD BUILDING STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.7 The structural integrity of the shield building shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.7.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION.

With the structural integrity of the shield building not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The structural integrity of the shield building shall be determined during the shutdown for each Type A containment leakage rate test (Specification 4.6.1.1.c) by a visual inspection of the exposed accessible interior and exterior surfaces of the shield building and verifying no apparent changes in appearance of the concrete surfaces or other abnormal degradation.

results in the lowest burst pressure and the longest length that would tear through-wall at steam-line break conditions. The repair bases for PWSCC at dented TSP intersections is obtained by projecting the crack profile to the end of the next operating cycle and determining if the projected profile meets the requirements of WCAP 15128, Revision 2, dated February 2000 as supplemented by TVA's letter to NRC dated March 2, 2000. The following provides the limits and bases for repair established in the WCAP analyses:

Freespan Indication Repair Limits

The tube will be repaired if the crack length outside the dented TSP is ≥40% maximum depth.

Crack Length Limit for ≥40% Maximum Depth

The crack length limit for ≥40% maximum depth indications is defined as 0.375 inch from the centerline of the TSP. This limit defines the edges of the TSP thickness of 0.75 inch for Model 51 S/Gs. It is acceptable for the crack to extend to both edges of the TSP as long as the maximum depth of the crack outside the TSP is <40% maximum depth and the requirements for EOC conditions are acceptable.

Operational Assessment Repair Bases

If the indication satisfies the above maximum depth and length requirements, the repair bases is then obtained by projecting the crack profile to the end of the next operating cycle and determining the burst pressure and leakage for the projected profile. The burst pressure and leakage is compared to the requirements in WCAP 15128, Revision 2, dated February 2000 as supplemented by TVA's letter to NRC dated March 2, 2000. Separate analyses are required for the total crack length and the length outside the TSP due to differences in requirements. If the projected EOC requirements are satisfied, the tube will be left in service.

The results of the condition monitoring and operational assessments will be reported to the NRC within 120 days following completion of the inspection.

6.6 REPORTABLE EVENT ACTION (DELETED)

6.7 SAFETY LIMIT VIOLATION (DELETED)

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. DELETED

e. DELETED

f. Fire Protection Program implementation.

g. DELETED