May 24, 2002

Mr. J. A. Scalice
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS REGARDING TECHNICAL SPECIFICATION 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" (TAC NOS. MB3879 AND MB3880)

Dear Mr. Scalice:

The Commission has issued the enclosed Amendment No. 276 to Facility Operating License No. DPR-77 and Amendment No. 267 to Facility Operating License No. DPR-79 for the Sequoyah Nuclear Plant, Units 1 and 2, respectively. These amendments are in response to your application dated November 8, 2001, as amended by your letter dated April 8, 2002. The amendments delete License Condition 2.H, "Reporting to the Commission," from the Operating Licenses. In addition, the amendments modify the Technical Specifications (TS) by deleting 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 are modified to remove those references.

The staff did not, however, approve the requested change to TS 4.4.5.5.c involving reporting of Category C-3 steam generator tube inspection results for reasons discussed in the enclosed Safety Evaluation (Enclosure 3).

The staff's Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice. Enclosure 4 is a *Federal Register* Notice of Partial Denial of Amendment.

Sincerely,

/RA/
Ronald W. Hernan, Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 50-328

Enclosures:
1. Amendment No. 276 to License No. DPR-77
2. Amendment No. 267 to License No. DPR-79
3. Safety Evaluation
4. *Federal Register* Notice

cc w/enclosures: See next page

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The staff did not, however, approve the requested change to TS 4.4.5.5.c involving reporting of Category C-3 steam generator tube inspection results for reasons discussed in the enclosed Safety Evaluation (Enclosure 3).

The staff's Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice. Enclosure 4 is a *Federal Register* Notice of Partial Denial of Amendment.

Sincerely,
/RA/
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Docket Nos. 50-327 and 50-328

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1. Amendment No. 276 to License No. DPR-77
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3. Safety Evaluation
4. *Federal Register* Notice

cc w/enclosures: See next page

DOCUMENT NAME: C:\ORPCheckout\FileNET\ML021500335.wpd
ADAMS Accession No. *SEE PREVIOUS CONCURRENCE

OFFICE	PDII-2/PM	PDII-2/LA	EMCB/SC	RORP/TSS/SC	OGC	PDII-2/SC
NAME	RHernan	BClayton	LLund*	RDennig*	AHodgdon*	TKoshy
DATE	5/21/02	5/21/02	4/10/02	4/12/02	5/20/02	5/23/02

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Sequoyah Nuclear Plant, Units 1 and 2 — Amendment No. 276 to Facility Operating License No. DPR-77 and Amendment No. 267 to Facility Operating License No. DPR-79

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TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 276
License No. DPR-77

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated November 8, 2001, as amended on April 8, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Operating License and Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-77 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 276, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented no later than 45 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Thomas Koshy, Acting Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Operating License and Technical Specifications

Date of Issuance: May 24, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 276

FACILITY OPERATING LICENSE NO. DPR-77

DOCKET NO. 50-327

Remove pages 12a and 13 of Operating License DPR-77 and replace them with the attached page 13.

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain a vertical line(s) indicating the area of change.

REMOVE	INSERT
Index Page XVII	Index Page XVII
1-5	1-5
3/4 1-4	3/4 1-4
3/4 4-10	3/4 4-10
3/4 4-12	3/4 4-12
3/4 5-4	3/4 5-4
3/4 5-8	3/4 5-8
3/4 6-11	3/4 6-11
3/4 6-12	3/4 6-12
B 3/4 4-4a	B 3/4 4-4a
6-6	6-6

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 267
License No. DPR-79

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated November 8, 2001, as amended on April 8, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Operating License and Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-79 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 267, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented no later than 45 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Thomas Koshy, Acting Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Operating License and Technical Specifications

Date of Issuance: May 24, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 267

FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

Replace page 12 of Operating License DPR-79 with the attached page 12.

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain a vertical line(s) indicating the area of change.

REMOVE	INSERT
Index Page XVII	Index Page XVII
1-6	1-6
3/4 1-4	3/4 1-4
3/4 4-14b	3/4 4-14b
3/4 4-16	3/4 4-16
3/4 5-4	3/4 5-4
3/4 5-8	3/4 5-8
3/4 6-11	3/4 6-11
3/4 6-12	3/4 6-12
B 3/4 4-3b	B 3/4 4-3b
6-6	6-6

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 276 TO FACILITY OPERATING LICENSE NO. DPR-77

AND AMENDMENT NO. 267 TO FACILITY OPERATING LICENSE NO. DPR-79

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

1.0 INTRODUCTION

By application dated November 8, 2001, as amended on April 8, 2002, the Tennessee Valley Authority (TVA) proposed to the U.S. Nuclear Regulatory Commission (NRC), an amendment to the Technical Specifications (TSs) for Sequoyah Nuclear Plant (SQN), Units 1 and 2. The requested changes would revise the Facility Operating Licenses by deleting License Condition 2.H, “Reporting to the Commission.” In addition, Administrative Control Section 6.6, “Reportable Event Action,” and Administrative Control Section 6.7, “Safety Limit Violation,” are deleted from the TSs. 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. TVA also requested changes to TS 4.4.5.5.C and Table 4.4-2 involving reporting of Category C-3 steam generator tube inspection results.

2.0 BACKGROUND

The proposed changes would eliminate notification, reporting, and review requirements that are adequately governed by the reporting requirements of Title 10, *Code of Federal Regulations* (10 CFR), Sections 50.72 and 50.73, and the TVA Nuclear (TVAN) Quality Assurance Plan from the Facility Operating Licenses and TSs. By eliminating these duplications, resource requirements required by both TVA and NRC will be averted.

3.0 EVALUATION

The proposed revision removes specifications that are not required to be contained in the TSs in accordance with the criteria in 10 CFR 50.36. The inclusion of these specifications in the TSs places an unnecessary potential burden on TVA and the NRC for processing licensing amendments. The proposed revision will reduce TVA and NRC activities for functions that were not intended to be controlled as TS requirements while maintaining an appropriate level of requirement control and an improved level of consistency with NUREG-1431, “Improved Standard Technical Specifications for Westinghouse-Designed Plants.” The NRC staff criteria regarding TSs required by 10 CFR 50.36 is as follows:

1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

ENCLOSURE 3

2. A process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design-basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

TVA has proposed that the following reporting requirements in the SQN TSs do not meet the criteria listed above and should, therefore, be removed from the TSs.

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

Condition 2.C(22)G(a) (Unit 1) - “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).

Condition 2.G (Unit 1) - “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).

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

Condition 2.F (Units 1 and 2) - “Protection of the Environment”: This 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”: This Section contains two TSs, 6.6.1.a and 6.6.6.b, both of which may be deleted for the following reasons:

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

Reactivity Control Systems - Moderator Temperature Coefficient - LCO 3.1.1.3
Emergency Core Cooling Systems (ECCS) - ECCS Subsystems - LCO 3.5.2
Emergency Core Cooling Systems - 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

TVA proposed, and the staff agrees, that these references may be deleted for the following reasons:

LCO 3.1.1.3, Action Statement 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.

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 a nuclear power plant's principal safety barriers.

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

TS 6.6.1.b governs internal SQN reportable event reviews and is being deleted to allow SQN to conform to NUREG-1431. Additionally, the Plant Operations Review Committee and Nuclear Safety Review Board reviews are delineated in the TVAN 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.

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

License Condition 2.H, "Reporting to the Commission," provides for the reporting by TVA to the Commission of 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 have been satisfied and whose reporting is no longer applicable. License Condition 2.H requires that TVA report any violations of the listed 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. Based on the discussion above, TVA stated that License Condition 2.H can be deleted.

The NRC staff agrees that 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. Therefore, the NRC staff finds the changes to be acceptable.

However, TVA also included the following request and justification that the NRC staff finds unacceptable:

> 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.

TVA is required by SR 4.4.5.5.C and Table 4.4-2 to notify the NRC, pursuant to 10 CFR 50.73 and prior to resumption of plant operation, results of steam generator tube inspections that fall into Category 3. TVA's justification was that 10 CFR 50.72 and 10 CFR 50.73, as clarified by NUREG-1022, Revision 2, would provide adequate reporting requirements. In fact, a Category 3 inspection result would not necessarily require NRC notification under these regulations, unless the burst pressure margin or leakage rate limits were violated. The NRC position regarding notification of Category 3 inspection results has not changed. Therefore, the NRC staff rejects TVA's request and justification to delete TS 4.4.5.5.c. This decision was conveyed to TVA on April 5, 2002, via a conference call. As a result of the call, TVA submitted revisions to TS pages 3/4 4-10 (Unit 1) and 3/4 4-14b (Unit 2) on April 8, 2002. These changes incorporate language from TS 6.6.1, which will be deleted by this amendment. The staff considers removal of the notification actions in Table 4.4-2 acceptable because they are redundant to TS 4.4.5.5.c. Likewise, removal of the Bases discussion regarding TS 4.4.5.5.c is acceptable because it adds little to enhance understanding of the notification requirement.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Tennessee State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

These amendments would involve changes to reporting requirements. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (67 FR 5339). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Ronald W. Hernan, NRR

Date: May 24, 2002

7590-01-P

UNITED STATES NUCLEAR REGULATORY COMMISSION

TENNESSEE VALLEY AUTHORITY

DOCKET NOS. 50-327 AND 50-328

NOTICE OF PARTIAL DENIAL OF AMENDMENT TO FACILITY OPERATING LICENSE AND OPPORTUNITY FOR HEARING

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has denied a portion of a request by the Tennessee Valley Authority (the licensee) for an amendment to Facility Operating License Nos. DPR-77 and DPR-79, issued to the licensee for operation of the Sequoyah Nuclear Plant, Unit Nos. 1 and 2, located in Hamilton County, Tennessee.

Notice of Consideration of Issuance of this amendment was published in the *Federal Register* on February 5, 2002 (67 FR 5339).

The purpose of the licensee's amendment request was to revise the Technical Specifications (TS) by changing TS 4.4.5.5.C and Table 4.4-2, which involve reporting Category C-3 steam generator tube inspection results to the NRC. The request also involved eliminating several other reporting requirements

The NRC staff has concluded that the licensee's request regarding steam generator Category C-3 condition reporting cannot be granted. The licensee was notified of the Commission's denial of the proposed change by a letter dated May 24, 2002.

By July 5, 2002, the licensee may demand a hearing with respect to the denial described above. Any person whose interest may be affected by this proceeding may file a written petition for leave to intervene.

A request for hearing or petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001 Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room (PDR), located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland, by the above date. A copy of any petitions should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902., attorney for the licensee.

For further details with respect to this action, see (1) the application for amendment dated February 5, 2002, and (2) the Commission's letter to the licensee dated May 24, 2002.

Documents may be examined, and/or copied for a fee, at the NRC's PDR, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland, and will be accessible electronically through the Agencywide Documents Access and Management System's Public Electronic Reading Room link at the NRC Web site http://www.nrc.gov/reading-rm/adams.html. Persons who do not have access to ADAMS or who encounter problems in accessing documents located in ADAMS, should contact the NRC PDR Reference staff by telephone at 1-800-397-4209, 301-415-4737, or by e-mail to pdr@nrc.gov.

Dated at Rockville, Maryland, this 24th day of May 2002.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Herbert N. Berkow, Director
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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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

Original signed by

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Attachment:
Appendices A and B Technical Specifications

Date of Issuance:
September 17, 1980

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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 3455 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. Results of steam generator tube inspections which fall into Category C-3 shall be reported as a degraded condition pursuant to 10 CFR 50.73 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.

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

1^{ST} SAMPLE INSPECTION			**2^{ND} SAMPLE INSPECTION**		**3^{RD} 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 (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:

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.

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.

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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RATED THERMAL POWER (RTP)

1.26 RATED THERMAL POWER (RTP) shall be a total reactor core heat transfer rate to the reactor coolant of 3455 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

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).

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. Results of steam generator tube inspections which fall into Category C-3 shall be reported as a degraded condition pursuant to 10 CFR 50.73 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.

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.	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.

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