June 13, 1995

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

SUBJECT: ISSUANCE OF AMENDMENTS - SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. M91977 AND M91978) (TS 95-05)

Dear Mr. Kingsley:

The Commission has issued the enclosed Amendment No. 203 to Facility Operating License No. DPR-77 and Amendment No. 193 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 April 6, 1995.

The amendments revise the technical specifications by deleting Tables 3.6-1, 3.6-2, and 3.8-2 and references to them, incorporating related guidance and justification, and modifying the specification related to electrical equipment protective devices.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly <u>Federal Register</u> notice.

Sincerely,

ORIGINAL SIGNED BY:

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

Docket Nos. 50-327 and 50-328

Enclosures:
1. Amendment No. 203 to License No. DPR-77
2. Amendment No. 193 to License No. DPR-79
3. Safety Evaluation

cc w/enclosures: See next page

DOCUMENT NAME: G:\SQN\91977.AME

OGC
5/5/95

To receive a copy of this document, indicate in the box:
"C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure
"N" = No copy

OFFICE	PDII-3/LA		PDII-3/PM	E	EELB	E	OTSB 95-090	E	PDII-3/D	C
NAME	BClayton		DLaBarge		CBerlinger		CGrimes		FHebdon	
DATE	4/20/95		4/20/95		4/25/95		5/2/95		6/13/95	

OFFICIAL RECORD COPY

9506200259 950613
PDR ADOCK 05000327
P PDR

DF01

AMENDMENT NO. 203 FOR SEQUOYAH UNIT NO. 1 - DOCKET NO. 50-327 and
AMENDMENT NO. 193 FOR SEQUOYAH UNIT NO. 2 - DOCKET NO. 50-328
DATED: June 13, 1995

DISTRIBUTION W/ENCLOSURE

Docket Files	
PUBLIC	
SQN Reading File	
S. Varga	0-14-E-4
J. Zwolinski	
G. Hill	T-5-C-3(2 per docket)
ACRS(4)	
OPA	0-2-G-5
OC/LFDCB	T9-E10
E. Merschoff	RII
M. Lesser	RII

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 203
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 April 6, 1995, 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 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. 203, 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 within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: June 13, 1995

Mr. Oliver D. Kingsley, Jr.
Tennessee Valley Authority

SEQUOYAH NUCLEAR PLANT

cc:
Mr. O. J. Zeringue, Sr. Vice President
Nuclear Operations
Tennessee Valley Authority
3B Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Dr. Mark O. Medford, Vice President
Engineering & Technical Services
Tennessee Valley Authority
3B Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. D. E. Nunn, Vice President
New Plant Completion
Tennessee Valley Authority
3B Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Site Vice President
Sequoyah Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Soddy Daisy, TN 37379

General Counsel
Tennessee Valley Authority
ET 11H
400 West Summit Hill Drive
Knoxville, TN 37902

Mr. P. P. Carier, Manager
Corporate Licensing
Tennessee Valley Authority
4G Blue Ridge
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Ralph H. Shell
Site Licensing Manager
Sequoyah Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Soddy Daisy, TN 37379

TVA Representative
Tennessee Valley Authority
11921 Rockville Pike
Suite 402
Rockville, MD 20852

Regional Administrator
U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW., Suite 2900
Atlanta, GA 30323

Mr. William E. Holland
Senior Resident Inspector
Sequoyah Nuclear Plant
U.S. Nuclear Regulatory Commission
2600 Igou Ferry Road
Soddy Daisy, TN 37379

Mr. Michael H. Mobley, Director
Division of Radiological Health
3rd Floor, L and C Annex
401 Church Street
Nashville, TN 37243-1532

County Judge
Hamilton County Courthouse
Chattanooga, TN 37402-2801

ATTACHMENT TO LICENSE AMENDMENT NO. 203

FACILITY OPERATING LICENSE NO. DPR-77

DOCKET NO. 50-327

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
1-2	1-2
3/4 6-1	3/4 6-1
3/4 6-2	3/4 6-2
3/4 6-4	3/4 6-4
3/4 6-5	---
3/4 6-6	---
3/4 6-6a	---
3/4 6-17	3/4 6-17
3/4 6-18	3/4 6-18
3/4 6-19	3/4 6-19
3/4 6-20	---
3/4 6-21	---
3/4 6-22	---
3/4 6-23	---
3/4 8-15	3/4 8-15
3/4 8-16	3/4 8-16
3/4 8-17	3/4 8-17
3/4 8-18	3/4 8-18
3/4 8-19	---
B3/4 6-2	B3/4 6-2
B3/4 6-3	B3/4 6-3
---	B3/4 6-3a

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

c. Digital channels - the injection of a simulated signal into the channel as close to the sensor input to the process racks as practicable to verify OPERABILITY including alarm and/or trip functions.

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:
   1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or
   2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.

b. All equipment hatches are closed and sealed.

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are within the limits of Specification 4.6.1.1.c,

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

f. Secondary containment bypass leakage is within the limits of Specification 3.6.1.2.

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.9 CORE ALTERATIONS shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMIT REPORT

1.10 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.14. Unit operation within these operating limits is addressed in individual specifications.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

c. Perform required visual examinations and leakage rate testing at P_a in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions. The maximum allowable leakage rate, L_a, is 0.25% of containment air weight per day at the calculated peak containment pressure P_a, 12 psig.

*Except valves, blind flanges, and deactivated automatic valves which are located inside the annulus or containment or the main steam valve vaults and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

CONTAINMENT SYSTEMS

SECONDARY CONTAINMENT BYPASS LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Secondary Containment bypass leakage rates shall be limited to a combined bypass leakage rate of less than or equal to 0.25 L_a for all penetrations that are secondary containment BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the combined bypass leakage rate exceeding 0.25 L_a for BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING, restore the combined bypass leakage rate from BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING to less than or equal to 0.25 L_a within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Pages 3/4 6-4 through 3/4 6-6a intentionally deleted

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 Each containment isolation valve shall be OPERABLE.*

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one or more of the isolation valve(s), except containment vacuum relief isolation valve(s), inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:

   1. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

   2. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

   3. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange; or

   4. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one or more containment vacuum relief isolation valve(s) inoperable, the valve(s) must be returned to OPERABLE status within 72 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 do not apply.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Deleted

*Penetration flow path(s) may be unisolated intermittently under administrative controls.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.2 Each automatic containment isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Phase A containment isolation test signal, each Phase A isolation valve actuates to its isolation position.

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

c. Verifying that on a Containment Ventilation isolation test signal, each Containment Ventilation Isolation valve actuates to its isolation position.

d. Verifying that on a high containment pressure isolation test signal, each Containment Vacuum Relief Valve actuates to its isolation position.

e. Verifying that on a Safety Injection test signal that the Normal Charging Isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each power operated or automatic containment isolation valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

Pages 3/4 6-19 through 3/4 6-23 intentionally deleted

ELECTRICAL POWER SYSTEMS

3/4.8.3 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.3.1 Primary and Backup containment penetration conductor overcurrent protective devices associated with each containment electrical penetration circuit shall be OPERABLE. The scope of these protective devices excludes those circuits for which credible fault currents would not exceed the electrical penetration design rating.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the containment penetration conductor overcurrent protective devices inoperable:

a. Restore the protective device(s) to OPERABLE status or de-energize the circuit(s) by tripping the associated backup circuit breaker within 72 hours and verify the backup circuit breaker to be tripped at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, or

b. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 All containment penetration conductor overcurrent protective devices shall be demonstrated OPERABLE:

a. At least once per 18 months:

   1. For at least one 6.9 kV reactor coolant pump circuit, such that all reactor coolant pump circuits are demonstrated OPERABLE at least once per 72 months, by performance of:

      (a) A CHANNEL CALIBRATION of the associated protective relays specified in appropriate plant instructions, and

      (b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed.

SURVEILLANCE REQUIREMENTS (Continued)

(c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 1 of the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. The functional test shall consist of injecting a current input at the specified setpoint to each selected circuit breaker and verifying that each circuit breaker functions as designed. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

3. By selecting and verifying a representative sample of each type of fuse on a rotating basis. Verification will be accomplished as described by SR 4.8.3.1.a.3.a. Each representative sample of fuses shall include at least 10% of all fuses of that type. Fuses found inoperable during verification shall be replaced with OPERABLE fuses prior to resuming operation. For each fuse found inoperable during verification, an additional representative sample of at least 10% of all fuses of that type shall be functionally tested until no more failures are found or all fuses of that type have been functionally tested.

(a) A fuse verification and maintenance program will be maintained to ensure that:

1. The proper size and type of fuse is installed,

2. The fuse shows no sign of deterioration, and

3. The fuse connections are tight and clean.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with manufacturer's recommendations.

ELECTRICAL POWER SYSTEMS

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

LIMITING CONDITION FOR OPERATION

3.8.3.2 The thermal overload protection devices, integral with the motor starter, of each valve used in safety systems shall be OPERABLE.

APPLICABILITY: Whenever the motor operated valve is required to be OPERABLE.

ACTION:

With one or more of the thermal overload protection devices inoperable, declare the affected valve(s) inoperable and apply the ACTION Statement to the affected valve(s).

SURVEILLANCE REQUIREMENTS

4.8.3.2 The above required thermal overload protection devices shall be demonstrated OPERABLE:

a. At least once per 18 months by the performance of a CHANNEL CALIBRATION of a representative sample of at least 25% of all thermal overload devices which are not bypassed, such that each non-bypassed device is calibrated at least once per 6 years.

b. At least once per 18 months, by the performance of a CHANNEL FUNCTIONAL TEST of the bypass circuitry for those thermal overload devices which are normally in force during plant operation and bypassed under accident conditions.

Pages 3/4 8-18 through 3/4 8-19 intentionally deleted

leakage paths to the auxiliary building is provided in plant procedures. Restricting the leakage through the bypass leakage paths to 0.25 L_a provides assurance that the leakage fraction assumptions used in the evaluation of site boundary radiation doses remain valid.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provide assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the annulus atmosphere of 0.5 psig and 2) the containment peak pressure does not exceed the maximum allowable internal pressure of 12 psig during LOCA conditions.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that 1) the containment air mass is limited to an initial mass sufficiently low to prevent exceeding the maximum allowable internal pressure during LOCA conditions and 2) the ambient air temperature does not exceed that temperature allowable for the continuous duty rating specified for equipment and instrumentation located within containment.

The containment pressure transient is sensitive to the initially contained air mass during a LOCA. The contained air mass increases with decreasing temperature. The lower temperature limits of 100°F for the lower compartment, 85°F for the upper compartment, and 60°F when less than or equal to 5% of RATED THERMAL POWER will limit the peak pressure to an acceptable value. The upper temperature limit influences the peak accident temperature slightly during a LOCA; however, this limit is based primarily upon equipment protection and anticipated operating conditions. Both the upper and lower temperature limits are consistent with the parameters used in the accident analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 12 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.8 EMERGENCY GAS TREATMENT SYSTEM (EGTS)

The OPERABILITY of the EGTS cleanup subsystem ensures that during LOCA conditions, containment vessel leakage into the annulus will be filtered through the HEPA filters and charcoal adsorber trains prior to discharge to the atmosphere. This requirement is necessary to meet the assumptions used in the accident analyses and limit the site boundary radiation doses to within the limits of 10 CFR 100 during LOCA conditions. Cumulative operation of the system with the heaters on for 10 hours over a 31 day period is sufficient to reduce the buildup of moisture on the absorbers and HEPA filters. ANSI N510-1975 will be used as a procedural guide for surveillance testing.

3/4.6.1.9 CONTAINMENT VENTILATION SYSTEM

Use of the containment purge lines is restricted to only one pair (one supply line and one exhaust line) of purge system lines at a time to ensure that the site boundary dose guidelines of 10 CFR Part 100 would not be exceeded in the event of a loss of coolant accident during purging operations. The analysis of this accident assumed purging through the largest pair of lines (a 24 inch inlet line and a 24 inch outlet line), a pre-existing iodine spike in the reactor coolant and four second valve closure times.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SUBSYSTEMS

The OPERABILITY of the containment spray subsystems ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the accident analyses.

3/4.6.2.2 CONTAINMENT COOLING FANS

The OPERABILITY of the lower containment vent coolers ensures that adequate heat removal capacity is available to provide long-term cooling following a non-LOCA event. Postaccident use of these coolers ensures containment temperatures remain within environmental qualification limits for all safety-related equipment required to remain functional.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The operability of containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a loss of coolant accident.

Additional valves have been identified as barrier valves, which in addition to the containment isolation valves discussed above, are a part of the accident monitoring instrumentation in Technical Specification 3/4.3.3.7 and are designated as Category 1 in accordance with Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1980.

CONTAINMENT SYSTEMS

3/4.6.3 CONTAINMENT ISOLATION VALVES (Continued)

BASES

The opening of penetration flow path(s) on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing the operator to close these valves in an accident situation, and (3) assuring that the environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment. For valves with controls located in the control room, these conditions can be satisfied by including a specific reference to closing the particular valves in the emergency procedures, since communication and environmental factors are not affected because of the location of the valve controls.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 193
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 April 6, 1995, 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 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. 193 , 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 within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: June 13, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 193

FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
1-2	1-2
3/4 6-1	3/4 6-1
3/4 6-2	3/4 6-2
3/4 6-4	3/4 6-4
3/4 6-5	---
3/4 6-6	---
3/4 6-6a	---
3/4 6-17	3/4 6-17
3/4 6-18	3/4 6-18
3/4 6-19	3/4 6-19
3/4 6-20	---
3/4 6-21	---
3/4 6-22	---
3/4 6-23	---
3/4 8-16	3/4 8-16
3/4 8-17	3/4 8-17
3/4 8-18	3/4 8-18
3/4 8-19	3/4 8-19
3/4 8-20	---
B3/4 6-2	B3/4 6-2
B3/4 6-3	B3/4 6-3
---	B3/4 6-3a

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

c. Digital channels - the injection of a simulated signal into the channel as close to the sensor input to the process racks as practicable to verify OPERABILITY including alarm and/or trip functions.

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:
   1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or
   2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.

b. All equipment hatches are closed and sealed.

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are within the limits of Specification 4.6.1.1.c,

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

f. Secondary containment bypass leakage is within the limits of Specification 3.6.1.2.

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.9 CORE ALTERATIONS shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMIT REPORT

1.10 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.14. Unit operation within these operating limits is addressed in individual specifications.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

c. Perform required visual examinations and leakage rate testing at P_a in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions. The maximum allowable leakage rate, L_a, is 0.25% of containment air weight per day at the calculated peak containment pressure P_a, 12 psig.

---

*Except valves, blind flanges, and deactivated automatic valves which are located inside the annulus or containment or the main steam valve vaults and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

CONTAINMENT SYSTEMS

SECONDARY CONTAINMENT BYPASS LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Secondary Containment bypass leakage rates shall be limited to a combined bypass leakage rate of less than or equal to 0.25 L_a for all penetrations that are secondary containment BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the combined bypass leakage rate exceeding 0.25 L_a for BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING, restore the combined bypass leakage rate from BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING to less than or equal to 0.25 L_a within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Pages 3/4 6-4 through 3/4 6-6a intentionally deleted

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 Each containment isolation valve shall be OPERABLE.*

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one or more of the isolation valve(s), except containment vacuum relief isolation valve(s), inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:

   1. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

   2. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

   3. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange; or

   4. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one or more containment vacuum relief isolation valve(s) inoperable, the valve(s) must be returned to OPERABLE status within 72 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 do not apply.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Deleted

*Penetration flow path(s) may be unisolated intermittently under administrative controls.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.2 Each automatic containment isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Phase A containment isolation test signal, each Phase A isolation valve actuates to its isolation position.

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

c. Verifying that on a Containment Ventilation isolation test signal, each Containment Ventilation Isolation valve actuates to its isolation position.

d. Verifying that on a high containment pressure isolation test signal, each containment vacuum relief valve actuates to its isolation position.

e. Verifying that on a Safety Injection test signal that the Normal Charging Isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each power operated or automatic containment isolation valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

Pages 3/4 6-19 through 3/4 6-23 intentionally deleted

ELECTRICAL POWER SYSTEMS

3/4.8.3 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.3.1 Primary and Backup containment penetration conductor overcurrent protective devices associated with each containment electrical penetration circuit shall be OPERABLE. The scope of these protective devices excludes those circuits for which credible fault currents would not exceed the electrical penetration design rating.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the containment penetration conductor overcurrent protective devices inoperable:

a. Restore the protective device(s) to OPERABLE status or de-energize the circuit(s) by tripping the associated backup circuit breaker within 72 hours and verify the backup circuit breaker to be tripped at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, or

b. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 All containment penetration conductor overcurrent protective devices shall be demonstrated OPERABLE:

a. At least once per 18 months:

   1. For at least one 6.9 kV reactor coolant pump circuit, such that all reactor coolant pump circuits are demonstrated OPERABLE at least once per 72 months, by performance of:

      (a) A CHANNEL CALIBRATION of the associated protective relays specified in appropriate plant instructions, and

      (b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed.

SURVEILLANCE REQUIREMENTS (Continued)

(c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 1 of the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. The functional test shall consist of injecting a current input at the specified setpoint to each selected circuit breaker and verifying that each circuit breaker functions as designed. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

3. By selecting and verifying a representative sample of each type of fuse on a rotating basis. Verification will be accomplished as described by SR 4.8.3.1.a.3.a. Each representative sample of fuses shall include at least 10% of all fuses of that type. Fuses found inoperable during verification shall be replaced with OPERABLE fuses prior to resuming operation. For each fuse found inoperable during verification, an additional representative sample of at least 10% of all fuses of that type shall be functionally tested until no more failures are found or all fuses of that type have been functionally tested.

(a) A fuse verification and maintenance program will be maintained to ensure that:

1. The proper size and type of fuse is installed,

2. The fuse shows no sign of deterioration, and

3. The fuse connections are tight and clean.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with manufacturer's recommendations.

ELECTRICAL POWER SYSTEMS

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

LIMITING CONDITION FOR OPERATION

3.8.3.2 The thermal overload protection devices, integral with the motor starter, of each valve used in safety systems shall be OPERABLE.

APPLICABILITY: Whenever the motor operated valve is required to be OPERABLE.

ACTION:

With one or more of the thermal overload protection devices inoperable, declare the affected valve(s) inoperable and apply the ACTION Statement to the affected valve(s).

SURVEILLANCE REQUIREMENTS

4.8.3.2 The above required thermal overload protection devices shall be demonstrated OPERABLE:

a. At least once per 18 months by the performance of a CHANNEL CALIBRATION of a representative sample of at least 25% of all thermal overload devices which are not bypassed, such that each non-bypassed device is calibrated at least once per 6 years.

b. At least once per 18 months, by the performance of a CHANNEL FUNCTIONAL TEST of the bypass circuitry for those thermal overload devices which are normally in force during plant operation and bypassed under accident conditions.

Pages 3/4 8-19 through 3/4 8-20 intentionally deleted

leakage paths to the auxiliary building is provided in plant procedures. Restricting the leakage through the bypass leakage paths to 0.25 L_a provides assurance that the leakage fraction assumptions used in the evaluation of site boundary radiation doses remain valid.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provide assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the annulus atmosphere of 0.5 psig and 2) the containment peak pressure does not exceed the maximum allowable internal pressure of 12 psig during LOCA conditions.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that 1) the containment air mass is limited to an initial mass sufficiently low to prevent exceeding the maximum allowable internal pressure during LOCA conditions and 2) the ambient air temperature does not exceed that temperature allowable for the continuous duty rating specified for equipment and instrumentation located within containment.

The containment pressure transient is sensitive to the initially contained air mass during a LOCA. The contained air mass increases with decreasing temperature. The lower temperature limits of 100°F for the lower compartment, 85°F for the upper compartment, and 60°F when less than or equal to 5% of RATED THERMAL POWER will limit the peak pressure to an acceptable value. The upper temperature limit influences the peak accident temperature slightly during a LOCA; however, this limit is based primarily upon equipment protection and anticipated operating conditions. Both the upper and lower temperature limits are consistent with the parameters used in the accident analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 12 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

CONTAINMENT SYSTEMS

BASES

3/4.6.1.8 EMERGENCY GAS TREATMENT SYSTEM (EGTS)

The OPERABILITY of the EGTS cleanup subsystem ensures that during LOCA conditions, containment vessel leakage into the annulus will be filtered through the HEPA filters and charcoal adsorber trains prior to discharge to the atmosphere. This requirement is necessary to meet the assumptions used in the accident analyses and limit the site boundary radiation doses to within the limits of 10 CFR 100 during LOCA conditions. Cumulative operation of the system with the heaters on for 10 hours over a 31 day period is sufficient to reduce the buildup of moisture on the absorbers and HEPA filters. ANSI N510-1975 will be used as a procedural guide for surveillance testing.

3/4.6.1.9 CONTAINMENT VENTILATION SYSTEM

Use of the containment purge lines is restricted to only one pair (one supply line and one exhaust line) of purge system lines at a time to ensure that the site boundary dose guidelines of 10 CFR Part 100 would not be exceeded in the event of a loss of coolant accident during purging operations. The analysis of this accident assumed purging through the largest pair of lines (a 24 inch inlet line and a 24 inch outlet line), a pre-existing iodine spike in the reactor coolant and four second valve closure times.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SUBSYSTEMS

The OPERABILITY of the containment spray subsystems ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the accident analyses.

3/4.6.2.2 CONTAINMENT COOLING FANS

The OPERABILITY of the lower containment vent coolers ensures that adequate heat removal capacity is available to provide long-term cooling following a non-LOCA event. Postaccident use of these coolers ensures containment temperatures remain within environmental qualification limits for all safety-related equipment required to remain functional.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The operability of containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a loss of coolant accident.

Additional valves have been identified as barrier valves, which in addition to the containment isolation valves discussed above, are a part of the accident monitoring instrumentation in Technical Specification 3/4.3.3.7 and are designated as Category 1 in accordance with Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1980.

3/4.6.3 CONTAINMENT ISOLATION VALVES (Continued)

The opening of penetration flow path(s) on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing the operator to close these valves in an accident situation, and (3) assuring that the environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment. For valves with controls located in the control room, these conditions can be satisfied by including a specific reference to closing the particular valves in the emergency procedures, since communication and environmental factors are not affected because of the location of the valve controls.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 203 TO FACILITY OPERATING LICENSE NO. DPR-77

AND AMENDMENT NO. 193 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 April 6, 1995, the Tennessee Valley Authority (the licensee) proposed an amendment to the Technical Specifications (TS) for Sequoyah Nuclear Plant (SQN) Units 1 and 2. The requested changes would delete Tables 3.6-1, 3.6-2, and 3.8-2 and references to them, incorporate related guidance and justification, and modify the specification related to electrical equipment protective devices. The information and controls provided by the tables and the specifications that reference them would be relocated to administratively controlled procedures in accordance with Generic Letter (GL) 91-08.

2.0 BACKGROUND

Section 182a of the Atomic Energy Act (the "Act") requires applicants for nuclear power plant operating licenses to state TS to be included as part of the license. The Commission's regulatory requirements related to the content of TS are set forth in 10 CFR 50.36. That regulation requires that the TS include items in five specific categories, including (1) safety limits, limiting safety system settings and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. However, the regulation does not specify the particular requirements to be included in a plant's TS.

The Commission has provided guidance for the contents of TS in its "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" ("Final Policy Statement"), 58 Fed. Reg. 39132 (July 22, 1993), in which the Commission indicated that compliance with the Final Policy Statement satisfies §182a of the Act. In particular, the Commission indicated that certain items could be relocated from the TS to licensee-controlled documents, consistent with the standard enunciated in *Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). In that case, the Atomic Safety and Licensing Appeal Board indicated that "technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety."

Consistent with this approach, the Final Policy Statement identified four criteria to be used in determining whether a particular matter is required to be included in the TS, 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; (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; and (4) a structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.[1] As a result, existing TS requirements which fall within or satisfy any of the criteria in the Final Policy Statement must be retained in the TS, while those TS requirements which do not fall within or satisfy these criteria may be relocated to other, licensee-controlled documents.

3.0 EVALUATION

The licensee has proposed the following TS changes:

(1) Replace the reference to Table 3.6-2 from Definition 1.7.a.2 for Containment Integrity and delete the reference to Table 3.6-1 from Technical Specification 3.6.1.2.

The staff concludes that these proposed changes are administrative or editorial in nature since they reflect the TS changes evaluated below. In addition, they provide clarification to the TS and represent no technical change to the current requirements and are consistent with GL 91-08. Therefore, they are acceptable.

(2) Replace the reference to Table 3.6-2 from Surveillance Requirement (SR) 4.6.1.1 with phrases that will allow the valves to be opened under administrative control.

As pointed out in GL 91-08, the design of the applicable penetrations includes positive control features to ensure that they are maintained closed. Therefore, in the absence of this provision, the opening of these locked or sealed closed valves would be contrary to the operability requirements for these valves that are currently listed in

---

[1]The Commission recently promulgated a proposed change to 10 CFR 50.36, pursuant to which the rule would be amended to codify and incorporate these criteria (59 FR 48180). The Commission's Final Policy Statement specified that only limiting conditions for Reactor Core Isolation Cooling, Isolation Condenser, Residual Heat Removal, Standby Liquid Control, and Recirculation Pump Trip, meet the guidance for inclusion in the TS under Criterion 4 (58 FR 39137). The Commission has solicited public comments on the scope of Criterion 4, in the pending rulemaking.

the TS table of containment isolation valves. With the removal of the TS list of valves, the operability requirements will apply to all containment isolation valves that have the locked or sealed closed feature. Therefore, this change is consistent with the guidance of GL 91-08, does not change the intent of the current TS and is acceptable.

(3) Delete Table 3.6-1, "Bypass Leakage Paths to the Auxiliary Building -- Secondary Containment Bypass Leakage Paths." A phrase would be added to indicate that the pages are intentionally deleted.

As pointed out in GL 91-08, the Final Safety Analysis Report defines the penetrations that are secondary containment bypass leakage paths. This definition is adequate such that the TS do not require further clarification and removal of the TS list is satisfactory.

(4) Revise Specification 3.6.3 to delete the reference to Table 3.6-2, add a phrase to take exception to the containment vacuum isolation valves, and add an action statement to indicate that Specification 3.0.4 does not apply to this specification. In addition, TS 3.6.3 would be changed to indicate that each containment isolation valve shall be operable and add a reference to a new footnote that indicates that the penetration flow path(s) may be unisolated intermittently under administrative controls.

These changes are consistent with GL 91-08 and are, therefore, acceptable.

(5) Delete SR 4.6.3.1 that requires the isolation valves in Table 3.6-2 to be demonstrated operable prior to returning the valve to service after maintenance, repair or replacement.

This proposed change, while not addressed in GL 91-08, is consistent with NUREG-1431. In addition, post maintenance testing is performed to ensure that the equipment meets all surveillance requirements prior to restoring equipment to an operable status. Therefore, the requirement to perform such tests is implicit in the definition and determination of operability. Therefore, it does not need to be an explicit TS requirement, and its removal is acceptable.

(6) Delete references to Table 3.6-2 in Specifications 4.6.3.2 and 4.6.3.3 and add additional wording to indicate that the specifications apply to automatic containment isolation valves.

The staff concludes that these proposed changes are administrative or editorial in nature since they reflect the TS changes evaluated below. In addition, they provide clarification to the TS and represent no technical change to the current requirements. Therefore, they are acceptable.

(7) Delete Table 3.6-2, "Containment Isolation Valves" and add a note to the page indicating that the information has been intentionally deleted. In addition, the two footnotes contained in the table pertaining to TS

3.0.4 would be deleted.

These proposed changes are consistent with the guidance given in GL 91-08. The necessary requirements covered by the footnotes has been incorporated into the proposed change to TS 3.6.3.c. Therefore, these changes are acceptable.

(8) Revise Specification 3.8.3.1 to specify that the Limiting Condition for Operation applies to primary and backup containment penetration conductor overcurrent protective devices associated with each containment electrical penetration shall be operable, add a phrase to indicate that the scope of these protective devices excludes those circuits for which credible fault currents would not exceed the electrical penetration design rating, and delete the phrase that references appropriate plant instructions in the action statement.

These proposed changes incorporate the guidance of GL 91-08 and are, therefore, acceptable.

(9) Delete the phrase from SR 4.8.3.1.a.3 that indicates that a complete listing of all fuses to be verified in accordance with the requirement will be maintained in appropriate plant instructions.

The need to maintain a list of fuses is inherent in the TS requirement to perform the tests. Therefore, deleting the requirement to maintain such a list is within the intent of GL 91-08 and is acceptable.

(10) Replace the phrase "appropriate plant instructions based on" with "procedures prepared in conjunction with" in SR 4.8.3.1.b. This will clarify that the procedures used to perform the 60-month circuit breaker inspection and preventive maintenance are prepared in accordance with manufacturer's recommendations.

This proposed change is clerical in nature and does not change the TS requirements. Therefore, it is acceptable.

(11) Delete Table 3.8-2, "Motor Operated Valves Thermal Overload Protection," and replace it with a note that indicates that the pages are intentionally blank. Replace the reference to Table 3.8-2 in Specification 3.8.3.2 with a phrase that indicates that the Requirement is applicable to valves used in safety systems.

These changes are consistent with the guidance of GL 91-08 and do not change the intent of the current TS. They are, therefore, acceptable.

(12) Incorporate appropriate changes to the Bases to reflect these changes.

The proposed change would add information that is consistent with GL 91-08, plus additional clarifying information that recognizes use of controls that are located in the control room. The change is appropriate and, therefore, acceptable.

4.0 SUMMARY

The staff reviewed the proposed changes and determined that the removal of these tables and the related requirements do not eliminate the requirements for the licensee to ensure that the system, structure, or component is capable of performing its safety function. Although these tables are removed from the TS and incorporated into the Sequoyah administratively controlled documents, since they are controlled documents described in the Final Safety Analysis Report, the licensee must evaluate any changes that affect these components and procedures in accordance with 10 CFR 50.59. Should the licensee's determination conclude that an unreviewed safety question is involved, due to either (1) an increase in the probability or consequence of accidents or malfunctions of equipment important to safety, (2) the creation of a possibility for an accident or malfunction of a different type than any evaluated previously, or (3) a reduction in the margin of safety, NRC approval and a license amendment would be required prior to implementation of the change. NRC inspection and enforcement programs also enable the staff to monitor facility changes and licensee adherence to Updated Final Safety Analysis commitments and to take any remedial action that may be appropriate.

Based on this review, the staff concluded that 10 CFR 50.36 does not require these tables to be retained in the TS. Requirements related to operability, applicability, and surveillance requirements, including performance of testing to ensure operability, are retained due to their importance in mitigating the consequences of an accident. However, the staff determined that the inclusion of these tables is an operational detail related to the licensee's safety analysis, which are adequately controlled by the requirements of 10 CFR 50.59. Therefore, the continued processing of license amendments related to revisions of the affected tables, where the revisions to those requirements do not involve an unreviewed safety question under 10 CFR 50.59, would afford no significant benefit with regard to protecting the public health and safety.

The staff has concluded, therefore, that removal of these tables and references to them is acceptable because (1) their inclusion in the TS is not specifically required by 10 CFR 50.36 or other regulations, (2) the tables have been incorporated into the Sequoyah administratively controlled document, and (3) changes that are deemed to involve an unreviewed safety question will require prior NRC approval in accordance with 10 CFR 50.59(c).

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

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. 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 (60 FR 24919). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). 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: David E. LaBarge

Dated: June 13, 1995