Docket Nos. 50-327
and 50-328

February 10, 1994

Tennessee Valley Authority
ATTN: Dr. Mark O. Medford, Vice President
Technical Support
3B Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Dr. Medford:

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M85974 AND M85975) (TS 93-02)

The Commission has issued the enclosed Amendment No. 176 to Facility Operating License No. DPR-77 and Amendment No. 167 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 March 10, 1993, which was amended by letter dated January 31, 1994.

The amendments incorporate a reference to the test requirements of 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," and remove the corresponding detailed test requirements from the Technical Specifications.

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

Sincerely,

/S/

David E. LaBarge, Sr. Project Manager
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 176 to
   License No. DPR-77
2. Amendment No. 167 to
   License No. DPR-79
3. Safety Evaluation

cc w/enclosures:
See next page

*See previous concurrence

NAME:	PDII-4/LA	PDII-4/PM	OTSB*	OGC	PDII-4/D
OFFICE:	BClayton	DLaBarge	CGrimes	EHolen	FHebdon
DATE:	2/1/93	2/1/93	/ /93	2/8/94	2/10/93

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AMENDMENT NO. 176 FOR SEQUOYAH UNIT NO. 1 - DOCKET NO. 50-327 and
AMENDMENT NO. 167 FOR SEQUOYAH UNIT NO. 2 - DOCKET NO. 50-328
DATED: February 10, 1994

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cc: Plant Service List

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ATTN: Dr. Mark O. Medford

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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. 176
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 March 10, 1993, which was amended by letter dated January 31, 1994, 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. 176, 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 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Mohan C. Thadani / for

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: February 10, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 176

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
VII	VII
1-2	1-2
3/4 6-1	3/4 6-1
3/4 6-2	3/4 6-2
3/4 6-3	3/4 6-3
3/4 6-4	3/4 6-4
3/4 6-8	3/4 6-8
3/4 6-11	3/4 6-11
3/4 6-12	3/4 6-12
3/4 6-15	3/4 6-15
3/4 6-23	3/4 6-23
B3/4 6-1	B3/4 6-1
B3/4 6-2	B3/4 6-2
----	B3/4 6-2a

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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 as provided in Table 3.6-2 of 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 ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative 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 as provided in Table 3.6-2 of 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 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.

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 identified in Table 3.6-1 as 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.

SECONDARY CONTAINMENT BYPASS LEAKAGE

SURVEILLANCE REQUIREMENTS

4.6.1.2 The secondary containment bypass leakage rates shall be demonstrated:

a. The combined bypass leakage rate to the auxiliary building shall be determined to be less than or equal to 0.25 L_a by applicable Type B and C tests at least once per 24 months except for penetrations which are not individually testable; penetrations not individually testable shall be determined to have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a (12 psig) during each Type A test.*

b. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a (13.2 psig) and the seal system capacity is adequate to maintain system pressure (or fluid head for the containment spray system and RHR spray system valves at penetrations 48A, 48B, 49A and 49B) for at least 30 days.

c. The provisions of Specification 4.0.2 are not applicable.

*Results shall be evaluated against the acceptance criteria of Specification 4.6.1.1.c in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.

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SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. After each opening, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage less than or equal to 0.01 L_a as determined by precision flow measurements when measured for at least two minutes with the volume between the door seals at a pressure greater than or equal to 6 psig,

b. By conducting an overall air lock leakage test at not less than P_a (12 psig) and by verifying the overall air lock leakage rate is within the limit of Specification 3.6.1.3.b and the results evaluated in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions:#

1. At least once per six months, and

2. Prior to establishing CONTAINMENT INTEGRITY if opened when CONTAINMENT INTEGRITY was not required when maintenance has been performed on the air lock that could affect the air lock sealing capability.*

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

#The provisions of Specification 4.0.2 are not applicable.

*Exemption to Appendix "J" of 10 CFR 50.

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 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 vessel. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance of the surfaces or other abnormal degradation. Any abnormal degradation of the containment vessel detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.6.1.

CONTAINMENT SYSTEMS

SHIELD BUILDING STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.9 One pair (one purge supply line and one purge exhaust line) of containment purge system lines may be open; the containment purge supply and exhaust isolation valves in all other containment purge lines shall be closed. Operation with purge supply or exhaust isolation valves open for either purging or venting shall be limited to less than or equal to 1000 hours per 365 days. The 365 day cumulative time period will begin every January 1.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With a purge supply or exhaust isolation valve open in excess of the above cumulative limit, or with more than one pair of containment purge system lines open, close the isolation valve(s) in the purge line(s) within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a containment purge supply and/or exhaust isolation valve having a measured leakage rate in excess of 0.05 L_a, restore the inoperable valve to OPERABLE status within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.9.1 The position of the containment purge supply and exhaust isolation valves shall be determined at least once per 31 days.

4.6.1.9.2 The cumulative time that the purge supply and exhaust isolation valves are open over a 365 day period shall be determined at least once per 7 days.

4.6.1.9.3 At least once per 3 months, each containment purge supply and exhaust isolation valve shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.05 L_a.*

*Results shall be evaluated against the acceptance criteria of Specification 4.6.1.1.c in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
C. PHASE "A" CONTAINMENT VENT ISOLATION (Cont.)			
13.	FCV-30-50	Upper Compt Purge Air Exh	4*
14.	FCV-30-51	Upper Compt Purge Air Exh	4*
15.	FCV-30-52	Upper Compt Purge Air Exh	4*
16.	FCV-30-53	Upper Compt Purge Air Exh	4*
17.	FCV-30-56	Lower Compt Purge Air Exh	4*
18.	FCV-30-57	Lower Compt Purge Air Exh	4*
19.	FCV-30-58	Inst Room Purge Air Exh	4*
20.	FCV-30-59	Inst Room Purge Air Exh	4*
21.	FCV-90-107	Cntmt Bldg LWR Compt Air Mon	5*
22.	FCV-90-108	Cntmt Bldg LWR Compt Air Mon	5*
23.	FCV-90-109	Cntmt Bldg LWR Compt Air Mon	5*
24.	FCV-90-110	Cntmt Bldg LWR Compt Air Mon	5*
25.	FCV-90-111	Cntmt Bldg LWR Compt Air Mon	5*
26.	FCV-90-113	Cntmt Bldg UPR Compt Air Mon	5*
27.	FCV-90-114	Cntmt Bldg UPR Compt Air Mon	5*
28.	FCV-90-115	Cntmt Bldg UPR Compt Air Mon	5*
29.	FCV-90-116	Cntmt Bldg UPR Compt Air Mon	5*
30.	FCV-90-117	Cntmt Bldg UPR Compt Air Mon	5*
D. OTHER			
1.	FCV-30-46	Vacuum Relief Isolation Valve	25
2.	FCV-30-47	Vacuum Relief Isolation Valve	25
3.	FCV-30-48	Vacuum Relief Isolation Valve	25
4.	FCV-62-90	Normal Charging Isolation Valve	12

*Provisions of LCO 3.0.4 are not applicable if valve is secured in its isolated position with power removed and leakage limits of Specification 4.6.1.1.c are satisfied. For purge valves, leakage limits under Surveillance Requirement 4.6.1.9.3 must also be satisfied.

#Provisions of LCO 3.0.4 are not applicable if valve is secured in its isolated position with power removed and either FCV-62-73 or FCV-62-74 is maintained operable.

**This valve is required after completion of the associated modification.

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

The safety design basis for primary containment is that the containment must withstand the pressures and temperatures of the limiting design basis accident (DBA) without exceeding the design leakage rates.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA), a steam line break, and a rod ejection accident (REA). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. This leakage rate limitation will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions. The containment was designed with an allowable leakage rate of 0.25 percent of containment air weight per day. This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting DBA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.25 percent per day in the safety analysis at P_a = 12.0 psig. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between tests.

Primary containment INTEGRITY or operability is maintained by limiting leakage to within the acceptance criteria of 10 CFR 50, Appendix J.

Individual leakage rates specified for the containment air lock (LCO 3.6.1.3), purge valves (LCO 3.6.1.9) and secondary bypass leakage (LCO 3.6.1.2) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits do not result in the primary containment being inoperable unless the leakage, when combined with other Type B and C test leakages, exceeds the acceptance criteria of Appendix J.

3/4.6.1.2 SECONDARY CONTAINMENT BYPASS LEAKAGE

The safety design basis for containment leakage assumes that 75 percent of the leakage from the primary containment enters the shield building annulus for filtration by the emergency gas treatment system. The remaining 25 percent of the primary containment leakage, which is considered to be bypassed to the auxiliary building, is assumed to exhaust directly to the atmosphere without filtration during the first 5 minutes of the accident. After 5 minutes, any bypass leakage to the auxiliary building is filtered by the auxiliary building gas treatment system. A tabulation of potential secondary containment bypass leakage paths to the auxiliary building is provided in Table 3.6-1. Restricting the leakage through the bypass leakage paths in Table 3.6-1 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.

3/4.6.1.7 SHIELD BUILDING STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment shield building will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to provide 1) protection for the steel vessel from external missiles, 2) radiation shielding in the event of a LOCA, and 3) and annulus surrounding the steel vessel that can be maintained at a negative pressure during accident conditions.

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. 167
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 March 10, 1993, which was amended by letter dated January 31, 1994, 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. 167, 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 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 10, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 167

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
VII	VII
1-2	1-2
3/4 6-1	3/4 6-1
3/4 6-2	3/4 6-2
3/4 6-3	3/4 6-3
3/4 6-4	3/4 6-4
3/4 6-8	3/4 6-8
3/4 6-11	3/4 6-11
3/4 6-12	3/4 6-12
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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.6 CONTAINMENT SYSTEMS

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 as provided in Table 3.6-2 of 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 ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative 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 as provided in Table 3.6-2 of 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 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.

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 identified in Table 3.6-1 as 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.

SECONDARY CONTAINMENT BYPASS LEAKAGE

SURVEILLANCE REQUIREMENTS

4.6.1.2 The secondary containment bypass leakage rates shall be demonstrated:

a. The combined bypass leakage rate to the auxiliary building shall be determined to be less than or equal to 0.25 L_a by applicable Type B and C tests at least once per 24 months except for penetrations which are not individually testable; penetrations not individually testable shall be determined to have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a, (12 psig) during each Type A test.*

b. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a (13.2 psig) and the seal system capacity is adequate to maintain system pressure (or fluid head for the containment spray system and RHR spray system valves at penetrations 48A, 48B, 49A and 49B) for at least 30 days.

c. The provisions of Specification 4.0.2 are not applicable.

*Results shall be evaluated against the acceptance criteria of Specification 4.6.1.1.c in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.

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SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. After each opening, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage less than or equal to 0.01 L_a as determined by precision flow measurements when measured for at least two minutes with the volume between the door seals at a pressure greater than or equal to 6 psig,

b. By conducting an overall air lock leakage test at not less than P_a (12 psig) and by verifying the overall air lock leakage rate is within the limit of Specification 3.6.1.3.b and the results evaluated in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions:#

1. At least once per six months, and

2. Prior to establishing CONTAINMENT INTEGRITY if opened when CONTAINMENT INTEGRITY was not required when maintenance has been performed on the air lock that could affect the air lock sealing capability.*

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

#The provisions of Specification 4.0.2 are not applicable.

*Exemption to Appendix "J" of 10 CFR 50.

CONTAINMENT SYSTEMS

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.6 The structural integrity of the containment vessel shall be determined during 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 vessel. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance of the surfaces or other abnormal degradation. Any abnormal degradation of the containment vessel detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.6.1.

SHIELD BUILDING STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

CONTAINMENT SYSTEMS

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.9 One pair (one purge supply line and one purge exhaust line) of containment purge system lines may be open; the containment purge supply and exhaust isolation valves in all other containment purge lines shall be closed. Operation with purge supply or exhaust isolation valves open for either purging or venting shall be limited to less than or equal to 1000 hours per 365 days. The 365 day cumulative time period will begin every January 1.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With a purge supply or exhaust isolation valve open in excess of the above cumulative limit, or with more than one pair of containment purge system lines open, close the isolation valve(s) in the purge line(s) within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a containment purge supply and/or exhaust isolation valve having a measured leakage rate in excess of 0.05 L_a, restore the inoperable valve to OPERABLE status within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.9.1 The position of the containment purge supply and exhaust isolation valves shall be determined at least once per 31 days.

4.6.1.9.2 The cumulative time that the purge supply and exhaust isolation valves are open over a 365 day period shall be determined at least once per 7 days.

4.6.1.9.3 At least once per 3 months, each containment purge supply and exhaust isolation valve shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.05 L_a.*

*Results shall be evaluated against the acceptance criteria of Specification 4.6.1.1.c in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
C.	PHASE "A" CONTAINMENT VENT ISOLATION (Cont.)		
13.	FCV-30-50	Upper Compt Purge Air Exh	4*
14.	FCV-30-51	Upper Compt Purge Air Exh	4*
15.	FCV-30-52	Upper Compt Purge Air Exh	4*
16.	FCV-30-53	Upper Compt Purge Air Exh	4*
17.	FCV-30-56	Lower Compt Purge Air Exh	4*
18.	FCV-30-57	Lower Compt Purge Air Exh	4*
19.	FCV-30-58	Inst Room Purge Air Exh	4*
20.	FCV-30-59	Inst Room Purge Air Exh	4*
21.	FCV-90-107	Cntmt Bldg LWR Compt Air Mon	5*
22.	FCV-90-108	Cntmt Bldg LWR Compt Air Mon	5*
23.	FCV-90-109	Cntmt Bldg LWR Compt Air Mon	5*
24.	FCV-90-110	Cntmt Bldg LWR Compt Air Mon	5*
25.	FCV-90-111	Cntmt Bldg LWR Compt Air Mon	5*
26.	FCV-90-113	Cntmt Bldg UPR Compt Air Mon	5*
27.	FCV-90-114	Cntmt Bldg UPR Compt Air Mon	5*
28.	FCV-90-115	Cntmt Bldg UPR Compt Air Mon	5*
29.	FCV-90-116	Cntmt Bldg UPR Compt Air Mon	5*
30.	FCV-90-117	Cntmt Bldg UPR Compt Air Mon	5*
D.	OTHER		
1.	FCV-30-46	Vacuum Relief Isolation Valve	2
2.	FCV-30-47	Vacuum Relief Isolation Valve	2
3.	FCV-30-48	Vacuum Relief Isolation Valve	2
4.	FCV-62-90	Normal Charging Isolation Valve	1

*Provisions of LCO 3.0.4 are not applicable if valve is secured in its isolated position with power removed and leakage limits of Specification 4.6.1.1.c are satisfied. For purge valves, leakage limits under surveillance Requirement 4.6.1.9.3 must also be satisfied.

#Provisions of LCO 3.0.4 are not applicable if valve is secured in its isolated position with power removed and either FCV-62-73 or FCV-62-74 is maintained operable.

**This valve is required after completion of the associated modification.

3/4.6.1 PRIMARY CONTAINMENT

The safety design basis for primary containment is that the containment must withstand the pressures and temperatures of the limiting design basis accident (DBA) without exceeding the design leakage rates.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA), a steam line break, and a rod ejection accident (REA). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. This leakage rate limitation will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions. The containment was designed with an allowable leakage rate of 0.25 percent of containment air weight per day. This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting DBA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.25 percent per day in the safety analysis at P_a = 12.0 psig. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between tests.

Primary containment INTEGRITY or operability is maintained by limiting leakage to within the acceptance criteria of 10 CFR 50, Appendix J.

Individual leakage rates specified for the containment air lock (LCO 3.6.1.3), purge valves (LCO 3.6.1.9) and secondary bypass leakage (LCO 3.6.1.2) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits do not result in the primary containment being inoperable unless the leakage, when combined with other Type B and C test leakages, exceeds the acceptance criteria of Appendix J.

3/4.6.1.2 SECONDARY CONTAINMENT BYPASS LEAKAGE

The safety design basis for containment leakage assumes that 75 percent of the leakage from the primary containment enters the shield building annulus for filtration by the emergency gas treatment system. The remaining 25 percent of the primary containment leakage, which is considered to be bypassed to the auxiliary building, is assumed to exhaust directly to the atmosphere without filtration during the first 5 minutes of the accident. After 5 minutes, any bypass leakage to the auxiliary building is filtered by the auxiliary building gas treatment system. A tabulation of potential secondary containment bypass leakage paths to the auxiliary building is provided in Table 3.6-1. Restricting the leakage through the bypass leakage paths in Table 3.6-1 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.

BASES

3/4.6.1.7 SHIELD BUILDING STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment shield building will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to provide 1) protection for the steel vessel from external missiles, 2) radiation shielding in the event of a LOCA, and 3) and annulus surrounding the steel vessel that can be maintained at a negative pressure during accident conditions.

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

ENCLOSURE 3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 176 TO FACILITY OPERATING LICENSE NO. DPR-77

AND AMENDMENT NO. 167 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 March 10, 1993, 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 add a reference to the test requirements of 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors" to the technical specifications at various locations, and remove the corresponding detailed test requirements and acceptance criteria. The licensee indicated that SQN TS 3.6.1.1 and TS 3.6.1.2 currently contain detailed containment leakage rate requirements, test requirements, test schedules, and test accuracies that are also required by 10 CFR Part 50, Appendix J. The proposed changes will remove the duplicate 10 CFR Part 50, Appendix J requirements from the TS. In addition to the specific changes, three other related TS, 4.6.1.6, 4.6.1.7 and 4.6.1.9.3 are being revised to remove references that will no longer be applicable. The licensee also proposed a change to a footnote in TS Table 3.6-2, "Containment Isolation Valves," that would clarify the additional testing requirements for the containment purge valves and correct an oversight from a previous TS change.

A supplemental letter dated January 31, 1994, supplied clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The licensee has proposed the following specific changes:

a. Revise definition for containment integrity (Definition 1.7, Item d) - Item d references the current TS 3.6.1.2 that governs containment leakage rate criteria. Proposed changes to TS 3.6.1.2 would revise Item d to reference the 10 CFR Part 50, Appendix J, containment leakage rate criteria that are provided by reference in TS 3.6.1.1.

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By letter dated January 31, 1994, the licensee proposed that a more appropriate reference is Specification 4.6.1.1.c, which contains the surveillance testing criteria, rather than 10 CFR 50, Appendix J. In addition, the containment integrity definition was expanded to include the secondary containment bypass leakage limit of Specification 3.6.1.2. These proposed changes clarify the intent of the TS requirements.

b. Revise Surveillance Requirement (SR) 4.6.1.1.c - This SR currently contains leak rate criteria for Type B and C penetrations. The Type B and C penetration leak rates are governed by 10 CFR Part 50, Appendix J. The licensee proposed to revise the SR to be consistent with standard requirements from NUREG-1431, "Standard Technical Specifications Westinghouse Plants."

By letter dated January 31, 1994, the licensee determined that in order to clarify the leak rate testing surveillance requirement, this step should also indicate that the testing is performed at the P_a pressure specified in 10 CFR Part 50, Appendix J.

c. Revise Limiting Condition for Operation (LCO) 3.6.1.2 - This LCO currently limits containment leakage rates in three categories: (a) overall integrity leakage rate, (b) combined leakage for Type B and C tests, and (c) combined leakage for secondary containment bypass leakage rates to the auxiliary building. The LCO would be revised to limit applicability to secondary containment bypass leakage rates (Category (c) only).

d. Delete LCO 3.6.1.2.a - This LCO currently limits the overall integrated containment leakage rate to less than or equal to a maximum allowable leakage rate (L_a). The 10 CFR 50, Appendix J test requirement referenced in SR 4.6.1.1.c maintains and governs the L_a limit.

e. Delete LCO 3.6.1.2.b - The LCO currently limits the containment combined leakage rate to less than or equal to 0.60 L_a for all penetrations and valves subject to Type B and C testing. The 10 CFR 50, Appendix J test requirements referenced in SR 4.6.1.1.c maintain this 0.60 L_a limit.

f. Revise LCO 3.6.1.2.c - The LCO currently limits the containment bypass leakage rate to less than or equal to 0.25 L_a. The secondary containment bypass leakage paths to the auxiliary building are specific to SQN and are not addressed by 10 CFR 50, Appendix J; therefore, this LCO is being retained under LCO 3.6.1.2.

g. Revise Action Statement for LCO 3.6.1.2 - The action statement for this LCO currently contains actions associated with three categories: (a) overall integrated leakage rate, (b) combined leakage rate for Type B and C penetrations, and (c) combined leakage for secondary containment bypass leakage paths to the auxiliary building. The licensee proposed reformatting this action statement to reflect its applicability to Category (c) only. Action requirements for Categories (a) and (b) are governed by 10 CFR Part 50, Appendix J (refer to SR 4.6.1.1.c).

h. Delete SR 4.6.1.2 (Items a, b, c, f, h, and i) - These SR items are associated with containment leakage rate criteria, test schedules, and accuracy requirements that are governed by 10 CFR Part 50, Appendix J. Items e, g, and j are associated with combined bypass leakage rates to auxiliary building and would be retained in the TS since they are not governed by Appendix J.

By letter dated January 31, 1994, the licensee proposed that a footnote be added that references Specification 4.6.1.2. The footnote would indicate that the results of the secondary containment bypass leakage tests shall be evaluated against the acceptance criteria of Specification 4.6.1.1.c in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions. The intent of this proposed change is to clarify the acceptance criteria.

i. By letter dated January 31, 1994, the licensee proposed that a clarification to the containment air lock operability surveillance requirement in SR 4.6.1.3.b should be incorporated by inserting a reference that indicates the results of the tests shall be evaluated against the acceptance criteria of Specification 3.6.1.3.b, in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.

j. Revise SR 4.6.1.6 and 4.6.1.7 - These SRs currently reference SR 4.6.1.2 for Type A containment leakage rate testing. This reference would no longer be applicable with implementation of the proposed changes to SR 4.6.1.2. The references would be changed to SR 4.6.1.1.c.

k. Revise SR 4.6.1.9.3 - This SR currently references SR 4.6.1.2.d, which would no longer be applicable with the proposed deletion of SR 4.6.1.2.d.

By letter dated January 31, 1994, the licensee proposed that this SR be clarified by indicating that the results shall be evaluated against the acceptance criteria of SR 4.6.1.1.c in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.

l. Revise footnote (*) to Table 3.6-2 - A reference to SR 4.6.3.4 is no longer applicable for defining the leakage limit on purge valves. The leakage limits for purge valves are governed by SR 4.6.1.9.3. Leakage limits for valves in Table 3.6-2 are also governed by 10 CFR Part 50, Appendix J. Therefore, the licensee proposed to revise the footnote accordingly. However, by letter dated January 31, 1994, the licensee indicated that a more appropriate reference than 10 CFR 50, Appendix J. is the specification itself, SR 4.6.1.1.c.

m. Revise Bases 3/4.6.1 - The proposed bases change would incorporate the NUREG-1431 wording to reflect 10 CFR Part 50, Appendix J leakage limits and acceptance criteria.

n. Revise Bases 3/4.6.1.2 - The proposed bases change would incorporate the secondary containment bypass leakage limitations.

The staff has reviewed the licensee's proposed TS changes as discussed above. The licensee's proposed TS changes continue to require that the containment integrity be maintained in accordance with 10 CFR Part 50, Appendix J. We find that the determination of containment leakage rates and offsite doses following an accident are not affected. SQN's current acceptance criteria governing containment leakage test limits (0.75 L_a for periodic Type A testing and 0.60 L_a for Types B and C testing) remain unchanged. Detailed test requirements, test schedules, and test accuracies that are being deleted from TS will remain governed by reference to 10 CFR Part 50, Appendix J. The proposed amendment does not affect the individual TS leakage rates associated with containment air lock, purge valves, or secondary bypass leakage to the auxiliary building, since these leakage rate limits are not specifically part of the acceptance criteria of 10 CFR Part 50, Appendix J. These individual leakage limits remain unchanged and are retained in TS. All other proposed changes are clarifications, including the revised wording for the footnote to TS Table 3.6-2 and those proposed by letter dated January 31, 1994. These clarifications do not impact the intent of the affected specifications and are administrative in nature. The proposed changes are considered to be a TS improvement, are consistent with the guidance contained in the NUREG-1431 and will not affect SQN's containment leakage test criteria, system conditions, plant configuration, and accident analysis. Making these changes now will reduce the potential for future TS changes and exemptions.

Based on the above evaluation, the staff concludes that the proposed changes to Sequoyah's Technical Specifications and its associated Bases for primary containment integrity to delete detail containment test requirements that are governed by 10 CFR Part 50, Appendix J, and other administrative clarifications are acceptable.

3.0 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.

4.0 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 and to the surveillance requirements. 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 (58 FR 28059). 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.

5.0 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: R. Goel

Dated: February 10, 1994