February 5, 1996

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

DISTRIBUTION W/ENCLOSURE
Docket Files
PUBLIC
SQN Rdg.
S. Varga O-14-E-4
C. Grimes O-11-E-22
G. Hill T-5-C-3(2 per docket)
ACRS
E. Merschoff RII
M. Lesser RII
R. Lobel

SUBJECT: ISSUANCE OF TECHNICAL SPECIFICATION AMENDMENTS FOR THE SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. M94239 AND M94240) (TS 95-24)

Dear Mr. Kingsley:

The Commission has issued the enclosed Amendment No. 217 to Facility Operating License No. DPR-77 and Amendment No. 207 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 December 8, 1995.

The amendments implement the change to 10 CFR Part 50, Appendix J to incorporate Option B, a voluntary performance-based option, for determining the frequency for performing Type A, B, and C Containment Leak Rate Testing.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register 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. 217 to License No. DPR-77
2. Amendment No. 207 to License No. DPR-79
3. Safety Evaluation

cc w/enclosures: See next page

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	SCSB		OGC		PDII-3/D	C
NAME	BClayton		DLaBarge		CBerlinger		EHoller		FHebdon	
DATE	1/23/96		1/24/96		1/25/96		2/2/96		2/5/96	

OFFICIAL RECORD COPY

140056

9602150323 960205
PDR ADOCK 05000327
P PDR

DF01

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

Mr. R. J. Adney, 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

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. 217
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 December 8, 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. 217, 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

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: February 5, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 217

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
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 6-7	3/4 6-7
3/4 6-8	3/4 6-8
3/4 6-11	3/4 6-11
3/4 6-15	3/4 6-15
3/4 6-17	3/4 6-17
B3/4 6-1	B3/4 6-1
B3/4 6-2	B3/4 6-2
---	6-18a

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. R16 R207

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3. R134

c. Perform required visual examinations and leakage rate testing in accordance with the Containment Leakage Rate Testing Program.

*Except valves, blind flanges, and deactivated automatic valves which are located inside the annulus or containment or the mainsteam 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. R195

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 percent L_a for all penetrations that are secondary containment BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING when pressurized to P_a.* R207

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION: R180

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.

*Enter the ACTION of LCO 3.6.1.1, "Primary Containment" when Secondary Containment Bypass Leakage results in exceeding the overall containment leakage rate acceptance criteria.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The secondary containment bypass leakage rates shall be demonstrated: R180

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 in accordance with the Containment Leakage Rate Test program, 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. R180

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.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE* with both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one containment air lock door inoperable:

   1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.

   2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

   3. Otherwise, be in at least HOT STANDBY within the next six hours and in COLD SHUTDOWN within the following 30 hours.

   4. The provisions of Specification 3.0.4 are not applicable.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next six hours and in COLD SHUTDOWN within the following 30 hours.

R16

*1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

2. Enter the ACTION of LCO 3.6.1.1, "Primary Containment" when air lock leakage results in exceeding the overall containment leakage rate acceptance criteria.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. By verifying leakage rates in accordance with the Containment Leakage Rate Test Program.

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

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.6 The structural integrity of the containment vessel shall be determined during shutdown by a visual inspection of the exposed accessible interior and exterior surfaces of the vessel. This inspection shall be performed in accordance with the Containment Leakage Rate Test Program to verify no apparent changes in appearance of the surfaces or other abnormal degradation. Any abnormal degradation of the containment vessel detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.6.1. R40

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. R22

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. R124

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. R124

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.* R180

*Enter the ACTION of LCO 3.6.1.1, "Primary Containment" when purge valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 Each containment isolation valve shall be OPERABLE.* R207

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: R207

   1. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or R201

   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. R207

c. The provisions of Specification 3.0.4 do not apply. R207

SURVEILLANCE REQUIREMENTS

4.6.3.1 Deleted R207

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

2. Enter the ACTION of LCO 3.6.1.1, "Primary Containment" when containment isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

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. R180

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 the Containment Leakage Rate Test Program, 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. R180

Primary containment INTEGRITY or operability is maintained by limiting leakage to within the acceptance criteria of the Containment Leakage Rate Test Program.

3/4.6.1.2 SECONDARY CONTAINMENT BYPASS LEAKAGE

R180

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 of 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 the Containment Leakage Rate Test Program. 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. |R180

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. |BR

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. |BR

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. |BR

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. Periodic visual inspections in accordance with the Containment Leakage Rate Test Program are sufficient to demonstrate this capability. |BR

h. Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50 Appendix J, Option B, as modified by approved exemptions. Visual examination and testing, including test intervals and extensions, shall be in accordance with Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 with exceptions provided in the site implementing instructions.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 12.0 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.25% of the primary containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment overall leakage rate acceptance criteria is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   2) For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to $\geq$ 6 psig for at least two minutes.

The provisions of SR 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

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. 207
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 December 8, 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. 207 , 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

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: February 5, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 207

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
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 6-7	3/4 6-7
3/4 6-8	3/4 6-8
3/4 6-11	3/4 6-11
3/4 6-15	3/4 6-15
3/4 6-17	3/4 6-17
B3/4 6-1	B3/4 6-1
B3/4 6-2	B3/4 6-2
6-19	6-19
---	6-19a

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 cntrol as permitted by Specification 3.6.3. |R193

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3. |R117

c. Perform required visual examinations and leakage rate testing in accordance with the Containment Leakage Rate Testing Program.

*Except valves, blind flanges, and deactivated automatic valves which are located inside the annulus or containment or the main steam valve vaults 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. |R183

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 presssurized to P_a.* R193

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION: R167

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.

*Enter the ACTION of LCO 3.6.1.1, "Primary Containment" when Secondary Containment Bypass Leakage results in exceeding the overall containment leakage rate acceptance criteria.

R167

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 in accordance with the Containment Leakage Rate Test Program, 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. R167

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. R167

c. The provisions of Specification 4.0.2 are not applicable.

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE* with both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

4. The provisions of Specification 3.0.4 are not applicable

b. With the containment air lock inoperable, except as the result of a inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

2. Enter the ACTION of LCO 3.6.1.1, "Primary Containment" when air lock leakage results in exceeding the overall containment leakage rate acceptance criteria.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. By verifying leakage rates in accordance with the Containment Leakage Rate Test Program.

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

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.6 The structural integrity of the containment vessel shall be determined during shutdown by a visual inspection of the exposed accessible interior and exterior surfaces of the vessel. This inspection shall be performed in accordance with the Containment Leakage Rate Test Program to verify no apparent changes in appearance of the surfaces or other abnormal degradation. Any abnormal degradation of the containment vessel detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.6.1. R28

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. R9

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. R109

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. R109

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. R9

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.* R167

*Enter the ACTION of LCO 3.6.1.1, "Primary Containment" when purge valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

R193

3.6.3 Each containment isolation valve shall be OPERABLE.*

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

R193

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: R188

   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.

R193

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. R188

R193

c. The provisions of Specification 3.0.4 do not apply.

SURVEILLANCE REQUIREMENTS

R193

4.6.3.1 Deleted

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

2. Enter the ACTION of LCO 3.6.1.1, "Primary Containment" when containment isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

R167

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 the Containment Leakage Rate Test Program, as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting DBA. The allowed leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. R167

Primary containment INTEGRITY or operability is maintained by limiting leakage to within the acceptance criteria of the Containment Leakage Rate Test Program.

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

R167

leakage paths to the auxiliary building is provided in the Containment Leakage Rate Test Program. 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. BR

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. BR

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. BR

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. Periodic visual inspections in accordance with the Containment Leakage Rate Test Program are sufficient to demonstrate this capability. BR

3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring. R134

h. Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50 Appendix J, Option B, as modified by approved exemptions. Visual examination and testing, including test intervals and extensions, shall be in accordance with Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 with exceptions provided in the site implementing instructions.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 12.0 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.25% of the primary containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment overall leakage rate acceptance criteria is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

2) For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to $\geq$ 6 psig for at least two minutes.

The provisions of SR 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted in accordance with 10 CFR 50.4. R64

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

STARTUP REPORT (continued)

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 217 TO FACILITY OPERATING LICENSE NO. DPR-77

AND AMENDMENT NO. 207 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

On September 12, 1995, the U.S. Nuclear Regulatory Commission (NRC) approved issuance of a revision to 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The NRC added Option B "Performance-Based Requirements" to allow licensees to voluntarily replace the prescriptive testing requirements of 10 CFR 50 Appendix J with testing requirements based on both overall performance and the performance of individual components.

By letter dated December 8, 1995, Tennessee Valley Authority, the licensee for Sequoyah Nuclear Plant Units 1 and 2, applied for amendments to Facility Licenses DPR-77 and DPR-78. The proposed changes would permit implementation of 10 CFR Part 50 Appendix J, Option B. The licensee has established a "Containment Leakage Rate Testing Program" and proposed adding this program to the technical specifications. The program references Regulatory Guide 1.163, September 1995, "Performance-Based Containment Leak Test Program," which specifies a method acceptable to the NRC for complying with Option B.

2.0 BACKGROUND

Compliance with Appendix J provides assurance that the primary containment, including those systems and components which penetrate the primary containment, do not exceed the allowable leakage rate specified in the technical specifications and bases. The allowable leakage rate is determined so that the leakage assumed in the safety analyses is not exceeded.

On February 4, 1992, the NRC published a notice in the Federal Register (57 FR 4166) discussing a planned initiative to begin eliminating requirements marginal to safety which impose a significant regulatory burden. 10 CFR 50 Appendix J, "Primary Containment Leakage Testing for Water-Cooled Power Reactors" was considered for this initiative and the staff undertook a study of possible changes to this regulation. The study examined the previous

ENCLOSURE 3

performance history of domestic containments and examined the effect on risk of a revision to the requirements of Appendix J. The results of this study are reported in NUREG-1493, "Performance-Based Leak-Test Program".

Based on the results of this study, the staff developed a performance-based approach to containment leakage rate testing. On September 12, 1995, the NRC approved issuance of this revision to 10 CFR Part 50, Appendix J, which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The revision added Option B "Performance-Based Requirements" to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance.

Regulatory Guide 1.163, September 1995, "Performance-Based Containment Leak Test Program," was developed as a method acceptable to the NRC staff for implementing Option B. This regulatory guide states that the Nuclear Energy Institute (NEI) guidance document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" provides methods acceptable to the NRC staff for complying with Option B with four exceptions which are described therein.

Option B requires that the regulatory guide or other implementation document used by a licensee to develop a performance-based leakage testing program must be included, by general reference, in the plant technical specifications in preparing amendment requests to implement Option B. The licensee has referenced Regulatory Guide 1.163 in the Sequoyah technical specifications.

Regulatory Guide 1.163 specifies an extension in Type A test frequency to at least one test in 10 years based upon two consecutive successful tests. Type B tests may be extended up to a maximum of 10 years based upon completion of two consecutive successful tests and Type C tests may be extended up to 5 years based on two consecutive successful tests.

By letter dated October 20, 1995, NEI proposed technical specifications to implement Option B. After some discussion, the staff and NEI agreed on a set of model technical specifications which were transmitted to NEI in a letter dated November 2, 1995. These technical specifications are to serve as a model for licensees to develop plant-specific technical specifications in preparing amendment requests to implement Option B.

In order for a licensee to determine the performance of each component, factors that are indicative of or affect performance, such as an administrative leakage limit must be established. The administrative limit is selected to be indicative of the potential onset of component degradation. Although these limits are subject to NRC inspection to assure that they are selected in a reasonable manner, they are not technical specifications requirements. Failure to meet an administrative limit requires the licensee to return to the minimum value of the test interval.

Option B requires that the licensee maintain records to show that the criteria for Type A, B and C tests have been met. In addition, the licensee must maintain comparisons of the performance of the overall containment system and the individual components to show that the test intervals are adequate. These records are subject to NRC inspection.

3.0 EVALUATION

The licensee's December 8, 1995, letter to the NRC proposes to establish a "Containment Leakage Rate Testing Program" and proposes to add this program to the technical specifications. The program references Regulatory Guide 1.163, September 1995, "Performance-Based Containment Leak Test Program," which specifies methods acceptable to the NRC for complying with Option B. This requires a change to existing technical specifications 4.6.1.1.c, 3.6.1.2, 4.6.1.2.a, 3.6.1.3, 4.6.1.3, 4.6.1.6, 3.6.1.9, and 3.6.3 and the addition of the program to Section 6.8.4 to add the "Containment Leakage Rate Testing Program" as item h. Corresponding bases were also modified.

Option B permits a licensee to choose Type A; or Type B and C; or Type A, B and C; testing to be done on a performance basis. The licensee has elected to perform Type A, B and C testing on a performance basis.

The technical specifications changes proposed by the licensee are in compliance with the requirements of Option B and consistent with the guidance of Regulatory Guide 1.163 and the generic technical specifications of the November 2, 1995 letter and are, therefore, acceptable.

In addition to changes related to the adoption of Option B to Appendix J, the licensee also proposes to add a note to Technical Specification 3.6.1.3 which states:

> An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

This note is included in the Improved Standard Technical Specifications, NUREG-1431. Since either air lock door is capable of providing a fission product barrier in the event of a design basis accident, there are situations in which one inoperable door does not result in the air lock being inoperable. The licensee provides two examples: (1) failure of an interlock mechanism and (2) seal leakage from a single door as instances which would not affect the integrity of the second air lock door or invalidate the previous overall air lock leakage test results. Since the change is consistent with the Improved Standard Technical Specifications and maintains the plant within its design basis, the staff finds this change acceptable.

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 (61 FR 182). 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: Richard M. Lobel

Dated: February 5, 1996