

March 2, 1989

Mr. Oliver D. Kingsley, Jr.  
Senior Vice President, Nuclear Power  
Tennessee Valley Authority  
6N 38A Lookout Place  
1101 Market Street  
Chattanooga, Tennessee 37402-2801

Dear Mr. Kingsley

SUBJECT: CONTAINMENT INTEGRATED LEAK RATE TEST (TAC R00503/R00504) (TS 88-32)  
SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

The Commission has issued the enclosed Amendment No. 102 to Facility Operating License No. DPR-77 and Amendment No. 91 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 2, 1988.

These amendments modify the Sequoyah Nuclear Plant, Units 1 and 2 Technical Specifications (TS). The changes are to revise surveillance requirement (SR) 4.6.1.2 and Bases 3/4.6.1.2 in the TS to permit the use of the mass point method for the Appendix J, Type A, containment integrated leak rate test. The request for an exemption to Appendix J of 10 CFR Part 50 in this application was withdrawn in your letter dated December 21, 1988.

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

Sincerely,

Original signed by

Suzanne Black, Assistant Director  
for Projects  
TVA Projects Division  
Office of Nuclear Reactor Regulation

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Enclosures:

1. Amendment No. 102 to License No. DPR-77
2. Amendment No. 91 to License No. DPR-79
3. Safety Evaluation

cc w/enclosures:  
See next page

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Mr. Oliver D. Kingsley, Jr.

-2-

Sequoyah Nuclear Plant

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

TENNESSEE VALLEY AUTHORITY  
DOCKET NO. 50-327  
SEQUOYAH NUCLEAR PLANT, UNIT 1  
AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 102  
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 2, 1988, 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.

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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. 102, 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.

FOR THE NUCLEAR REGULATORY COMMISSION



Suzanne Black, Assistant Director  
for Projects  
TVA Projects Division  
Office of Nuclear Reactor Regulation

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: March 2, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 102

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. Overleaf pages\* are provided to maintain document completeness.

REMOVE

3/4 6-3

3/4 6-4

B 3/4 6-1

B 3/4 6-2

INSERT

3/4 6-3

3/4 6-4\*

B 3/4 6-1

B 3/4 6-2\*

## CONTAINMENT SYSTEMS

### SURVEILLANCE REQUIREMENTS

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4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4-1972; however, the methods of ANSI/ANS 56.8-1987 for mass point data analysis may be used in lieu of the methods specified in ANSI N45.4-1972.

- a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at  $40 \pm 10$  month intervals during shutdown at  $P_a$  (12 psig) during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.
- b. If any periodic Type A test fails to meet  $0.75 L_a$ , the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet  $0.75 L_a$ , a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet  $0.75 L_a$  at which time the above test schedule may be resumed.
- c. The accuracy of each Type A test shall be verified by a supplemental test which:
  1. Confirms the accuracy of the Type A test by verifying that the difference between supplemental and Type A test data is within  $0.25 L_a$ .
  2. Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.
  3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage at  $P_a$  (12 psig).
- d. Type B and C tests shall be conducted with gas at  $P_a$  (12 psig) at intervals no greater than 24 months except for tests involving:
  1. Air locks,
  2. Penetrations using continuous leakage monitoring systems, and
  3. Valves pressurized with fluid from a seal system.

## CONTAINMENT SYSTEMS

### SURVEILLANCE REQUIREMENTS (Continued)

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- e. 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.
- f. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.
- g. 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.
- h. Type B tests for penetrations employing a continuous leakage monitoring system shall be conducted at  $P_a$  (12 psig) at intervals no greater than once per 3 years.
- i. All test leakage rates shall be calculated using observed data converted to absolute values. Error analyses shall be performed to select a balanced integrated leakage measurement system.
- j. The provisions of Specification 4.0.2 are not applicable.

## 3/4.6 CONTAINMENT SYSTEMS

### BASES

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#### 3/4.6.1 PRIMARY CONTAINMENT

#### 3/4.6.1 PRIMARY CONTAINMENT

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

#### 3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure,  $P_a$ . 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 leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix "J" of 10 CFR 50. Due to the increased accuracy of the mass point method for containment integrated leakage testing, the mass point method referenced in ANSI/ANS 56.8-1987 can be used in lieu of the methods described in ANSI N45.4-1972.

#### 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 SYSTEMS

### BASES

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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) an 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

TENNESSEE VALLEY AUTHORITY  
DOCKET NO. 50-328  
SEQUOYAH NUCLEAR PLANT, UNIT 2  
AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 91  
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 2, 1988, 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. 91, 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.

FOR THE NUCLEAR REGULATORY COMMISSION



Suzanne Black, Assistant Director  
for Projects  
TVA Projects Division  
Office of Nuclear Reactor Regulation

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: March 2, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 91

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

3/4 6-3

B 3/4 6-1

B 3/4 6-2

INSERT

3/4 6-3

B 3/4 6-1

B 3/4 6-2\*

## CONTAINMENT SYSTEMS

### SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4-1972; however, the methods of ANSI/ANS 56.8-1987 for mass point data analysis may be used in lieu of the methods specified in ANSI N45.4-1972.

- a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at  $40 \pm 10$ -month intervals during shutdown at  $P_a$ , 12 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.
- b. If any periodic Type A test fails to meet  $0.75 L_a$  the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet  $0.75 L_a$ , a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet  $0.75 L_a$  at which time the above test schedule may be resumed.
- c. The accuracy of each Type A test shall be verified by a supplemental test which:
  1. Confirms the accuracy of the Type A test by verifying that the difference between supplemental and Type A test data is within  $0.25 L_a$ ,
  2. Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.
  3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage at  $P_a$ , 12 psig.
- d. Type B and C tests shall be conducted with gas at  $P_a$ , 12 psig, at intervals no greater than 24 months except for tests involving:
  1. Air locks,
  2. Penetrations using continuous leakage monitoring systems, and

## 3/4.6 CONTAINMENT SYSTEMS

### BASES

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#### 3/4.6.1 PRIMARY CONTAINMENT

#### 3/4.6.1 PRIMARY CONTAINMENT

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

#### 3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure,  $P_a$ . 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 leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix "J" of 10 CFR 50. Due to the increased accuracy of the mass point method for containment integrated leakage testing, the mass point method referenced in ANSI/ANS 56.8-1987 can be used in lieu of the methods described in ANSI N45.4-1972.

#### 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 SYSTEMS

### BASES

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#### INTERNAL PRESSURE (Continued)

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) an 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

ENCLOSURE 3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION  
SUPPORTING AMENDMENT NO. 102 TO FACILITY OPERATING LICENSE NO. DPR-77  
AND AMENDMENT NO. 91 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 the letter dated December 2, 1988, the Tennessee Valley Authority (TVA) requested a change to the Sequoyah, Units 1 and 2 Technical Specifications (TS) involving the method used to calculate the leakage rates during a containment integrated leak rate test (Type A Test).

The changes are to revise surveillance requirement (SR) 4.6.1.2 and Bases 3/4.6.1.2 in the TS to permit the use of the mass point method for calculating containment leakage rates for the Appendix J, Type A test. The request for an exemption to Appendix J of 10 CFR Part 50 in this application was withdrawn in TVA's letter dated December 21, 1988. The proposed Bases 3/4.6.1.2 was revised to delete the reference to "an exemption to 10 CFR 50 Appendix J has been granted" because the request for the exemption was withdrawn.

2.0 EVALUATION

Originally, when the NRC issued the Containment Integrated Leakage Test Requirements in Appendix J to 10 CFR Part 50, the staff required licensees to use the state-of-the-art leakage test methodology as described in ANSI Standard N 45.4-1972 for the Type A Test. ANSI N 45.4-1972 accepted two techniques for calculating Type A test leakage rates; namely, the total-time method and the point-to-point method. Advances in technology have resulted in a third leakage rate calculation method called the mass point method. This method is specified in ANSI/ANS 56.8-1987. The NRC amended 10 CFR Part 50, Appendix J on November 15, 1988 to permit the use of the mass point method of calculating the Type A leakage rates if the test duration is at least 24 hours.

With the proposed TS change 88-32, TVA intends to use the mass point method for calculating the Type A test leakage rates in lieu of the two methods described in ANSI N 45.4-1972. TVA stated that it intends to use the mass point method within the 24-hour test duration as required by Appendix J to 10 CFR Part 50. Since TVA's TS change to use the mass point method to calculate Type A Test Leakage Rates conforms to the regulations in 10 CFR Part 50, Appendix J, the staff concludes that TS change 88-32 is acceptable.

### 3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The staff has determined that the amendments involve 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 these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendments meet 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 nor environmental assessment need be prepared in connection with the issuance of these amendments.

### 4.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (53 FR 53099) on December 30, 1988 and consulted with the State of Tennessee. No public comments were received and the State of Tennessee did not have any comments.

The staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendments will not be inimical to the common defense and security nor to the health and safety of the public.

Principal Contributor: P. Hearn

Dated: March 2, 1989