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February 3, 1988

Posted
Amdt. 137
to DPR-52

Docket Nos. 50-259/260/296

Mr. S. A. White
Manager of Nuclear Power
Tennessee Valley Authority
6N 38A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Mr. White:

SUBJECT: TECHNICAL SPECIFICATION CHANGE TO DELETE OPTIONAL REDUCED PRESSURE TEST FOR CONTAINMENT INTEGRATED LEAK RATE TEST (TS 230) (TAC R00028/29/30)

Re: Browns Ferry Nuclear Plant, Units 1, 2, and 3

The Commission has issued the enclosed Amendments Nos. 141, 137, and 112 to Facility Operating Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units 1, 2 and 3, respectively. These amendments are in response to your application dated May 15, 1987.

The amendments delete the option in the technical specifications to perform a reduced pressure test method for the integrated leak rate test and correct the acceptable leak rate limit of the drywell atmosphere to the suppression chamber with a 1 psi differential pressure.

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:
Gary G. Zech, Assistant Director
for Projects
TVA Projects Division
Office of Special Projects

Enclosures:
1. Amendment No. 141 to License No. DPR-33
2. Amendment No. 137 to License No. DPR-52
3. Amendment No. 112 to License No. DPR-68
4. Safety Evaluation

cc w/enclosures:
See next page

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* Conditions on review of Commission Decision [illegible]

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

February 3, 1988

Docket Nos. 50-259/260/296

Mr. S. A. White
Manager of Nuclear Power
Tennessee Valley Authority
6N 38A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Mr. White:

SUBJECT: TECHNICAL SPECIFICATION CHANGE TO DELETE OPTIONAL REDUCED PRESSURE TEST FOR CONTAINMENT INTEGRATED LEAK RATE TEST (TS 230) (TAC R00028/29/30)

Re: Browns Ferry Nuclear Plant, Units 1, 2, and 3

The Commission has issued the enclosed Amendments Nos. 141, 137, and 112 to Facility Operating Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units 1, 2 and 3, respectively. These amendments are in response to your application dated May 15, 1987.

The amendments delete the option in the technical specifications to perform a reduced pressure test method for the integrated leak rate test and correct the acceptable leak rate limit of the drywell atmosphere to the suppression chamber with a 1 psi differential pressure.

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Sincerely,

Gary G. Zech, Assistant Director
for Projects
TVA Projects Division
Office of Special Projects

Enclosures:
1. Amendment No. 141 to License No. DPR-33
2. Amendment No. 137 to License No. DPR-52
3. Amendment No. 112 to License No. DPR-68
4. Safety Evaluation

cc w/enclosures:
See next page

Mr. S. A. White
Tennessee Valley Authority

Browns Ferry Nuclear Plant
Units 1, 2, and 3

cc:
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Dr. Henry Myers, Science Advisor
Committee on Interior
and Insular Affairs
U.S. House of Representatives
Washington, D.C. 20515

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 141
License No. DPR-33

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated May 15, 1987, 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-33 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 141 , 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 and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Gary G. Zech, Assistant Director
for Projects
TVA Projects Division
Office of Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 3, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 141

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

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	INSERT
3.7/4.7-3	3.7/4.7-3
3.7/4.7-4	3.7/4.7-4
3.7/4.7-5	3.7/4.7-5
3.7/4.7-6	3.7/4.7-6*
3.7/4.7-11	3.7/4.7-11
3.7/4.7-12	3.7/4.7-12*
3.7/4.7-45	3.7/4.7-45
3.7/4.7-46	3.7/4.7-46

LIMITING CONDITIONS FOR OPERATION

3.7.A. Primary Containment

2.a. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics tests at power levels not to exceed 5 MW(t).

b. Primary containment integrity is confirmed if the maximum allowable integrated leakage rate, L_a, does not exceed the equivalent of 2 percent of the primary containment volume per 24 hours at the 49.6 psig design basis accident pressure, P_a.

c. If N_2 makeup to the primary containment averaged over 24 hours (corrected for pressure, temperature, and venting operations) exceeds 542 SCFH, it must be reduced to < 542 SCFH within 8 hours or the reactor shall be placed in Hot Shutdown within the next 16 hours.

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

2. Integrated Leak Rate Testing

Primary containment nitrogen consumption shall be monitored to determine the average daily nitrogen consumption for the last 24 hours. Excessive leakage is indicated by a N_2 consumption rate of > 2% of the primary containment free volume per 24 hours (corrected for drywell temperature, pressure, and venting operations) at 49.6 psig. Corrected to normal drywell operating pressure of 1.1 psig, this value is 542 SCFH. If this value is exceeded, the action specified in 3.7.A.2.C shall be taken.

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 to 10 CFR 50 using the methods and provisions of ANSI N45.4 (1972).

a. Three type A tests (overall integrated containment leakage rate) shall be conducted at 40 ± 10-month intervals during shutdown at P_a, 49.6 psig, during each 10-year plant inservice inspection.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

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 0.75L_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. 1. Test duration shall be at least 8 hours.

2. A 4-hour stabilization period will be required and the containment atmosphere will be considered stabilized when the change in weighted average air temperature averaged over an hour does not deviate by more than 0.5°R/hour from the average rate of change of temperature measured from the previous 4 hours.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

d. 1. At least 20 sets of data points at approximately equal time intervals and in no case at intervals greater than one hour shall be provided for proper statistical analysis.

2. The figure of merit for the instrumentation system shall never exceed 0.25 L_a.

e. The test shall not be concluded with an increasing calculated leak rate.

f. The accuracy of each type A test shall be verified by a supplemental test which:

1. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the type A test data is within 0.25 L_a.

2. Has 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 (49.6 psig).

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

g. Local leak rate tests (LLRTs) shall be performed on the primary containment testable penetrations and isolation valves, which are not part of a water-sealed system, at not less than 49.6 psig (except for the main steam isolation valves, see 4.7.A.2.i) and not less than 54.6 psig for water-sealed valves each operating cycle. Bolted double-gasketed seals shall be tested whenever the seal is closed after being opened and at least once per operating cycle. Acceptable methods of testing are halide gas detection, soap bubbles, pressure decay, hydrostatically pressurized fluid flow or equivalent.

The personnel air lock shall be tested at 6-month intervals at an internal pressure of not less than 49.6 psig. In addition, if the personnel air lock is opened during periods when containment integrity is not required, a test at the end of such a period will be conducted at not less than 49.6 psig. If the personnel air lock is opened during a period when containment integrity is required, a test at $\geq$ 2.5 psig shall be conducted within 3 days after being opened. If the air lock is opened more frequently than once every 3 days, the air lock shall be tested at least once every 3 days during the period of frequent openings.

LIMITING CONDITIONS FOR OPERATION

3.7.A Primary Containment

3.7.A.4.b (Cont'd)

as it is determined to be not more than 3° open as indicated by the position lights.

c. Two drywell-suppression chamber vacuum breakers may be determined to be INOPERABLE for opening.

d. If Specifications 3.7.A.4.a, 3.7.A.4.b, or 3.7.A.4.c. cannot be met, the unit shall be placed in a Cold Shutdown condition in an orderly manner within 24 hours.

5. Oxygen Concentration

a. After completion of the fire-related startup retesting program, containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100 psig, except as specified in 3.7.A.5.b.

b. Within the 24-hour period subsequent to placing the reactor in the RUN mode following a shut-down, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.

c. If Specification 3.7.A.5.a and 3.7.A.5.b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.A Primary Containment

4.7.A.4.b (Cont'd)

time when operability is required all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the INOPERABLE valve has been returned to normal service.

c. Once each operating cycle each vacuum breaker valve shall be inspected for proper operation of the valve and limit switches.

d. A leak test of the drywell to suppression chamber structure shall be conducted during each operating cycle. Acceptable leak rate is 0.09 lb/sec of primary containment atmosphere with 1 psi differential.

5. Oxygen Concentration

a. The primary containment oxygen concentration shall be measured and recorded daily. The oxygen measurement shall be adjusted to account for the uncertainty of the method used by adding a predetermined error function.

b. The methods used to measure the primary containment oxygen concentration shall be calibrated once every refueling cycle.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.A. Primary Containment	4.7.A. Primary Containment
6. Drywell-Suppression Chamber Differential Pressure	6. Drywell-Suppression Chamber Differential Pressure
a. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.1 psid except as specified in (1) and (2) below: (1) This differential shall be established within 24 hours of achieving operating temperature and pressure. The differential pressure may be reduced to less than 1.1 psid 24 hours prior to a scheduled shutdown. (2) This differential may be decreased to less than 1.1 psid for a maximum of four hours during required operability testing of the HPCI system, RCIC system and the drywell-pressure suppression chamber vacuum breakers.	a. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift.
b. If the differential pressure of Specification 3.7.A.6.a cannot be maintained and the differential pressure cannot be restored within the subsequent six-hour period, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours.	

be inoperable or the valve disc is stuck. For this case, a check light on and red light off confirms the disc is in a nearly closed position even if one of the indications is in error. Although the valve may be inoperable for full closure, it does not constitute a safety threat.

If the red light circuit alone is inoperable, the valve shall still be considered fully operable. If the green and red or the green light circuit alone is inoperable the valve shall be considered inoperable for opening. If the check and green or check light circuit alone is inoperable, the valve shall be considered inoperable for full closure. If the red and check light circuits are inoperable the valve shall be considered inoperable and open greater than 3°. For a light circuit to be considered operable the light must go on and off in proper sequence during the opening-closing cycle. If none of the lights change indication during the cycle, the valve shall be considered inoperable and open unless the check light stays on and the red light stays off in which case the valve shall be considered inoperable for opening.

The 12 drywell vacuum breaker valves which connect the suppression chamber and drywell are sized on the basis of the Bodega pressure suppression system tests. Ten operable to open vacuum breaker valves (18-inch) selected on this test basis and confirmed by the green lights are adequate to limit the pressure differential between the suppression chamber and drywell during postaccident drywell cooling operations to a value which is within suppression system design values.

The containment design has been examined to determine that a leakage equivalent to one drywell vacuum breaker opened to no more than a nominal 3° as confirmed by the red light is acceptable.

On this basis an indefinite allowable repair time for an inoperable red light circuit on any valve or an inoperable check and green or check light circuit alone or a malfunction of the operator or disc (if nearly closed) on one valve, or an inoperable green and red or green light circuit alone on two valves is justified.

During each operating cycle, a leak rate test shall be performed to verify that significant leakage flow paths do not exist between the drywell and suppression chamber. The drywell pressure will be increased by at least one psi with respect to the suppression chamber pressure and held constant. The two psig setpoint will not be exceeded. The subsequent suppression chamber pressure transient (if any) will be monitored with a sensitive pressure gauge. If the drywell pressure cannot be increased by one psi over the suppression chamber pressure it would be because a significant leakage path exists; in this event the leakage source will be identified and eliminated before power operation is resumed.

With a differential pressure of greater than one psig, the rate of change of the suppression chamber pressure must not exceed 0.25 inches of water per minute as measured over a 10-minute period, which corresponds to about 0.09 lb/sec of containment air. In the event the rate of change exceeds this value then the source of leakage will be identified and eliminated before power operation is resumed.

The water in the suppression chamber is used for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a daily check of the temperature and volume is adequate to assure that adequate heat removal capability is present.

The interior surfaces of the drywell and suppression chamber are coated as necessary to provide corrosion protection and to provide a more easily decontaminable surface. The surveillance inspection of the internal surfaces each operating cycle assures timely detection of corrosion. Dropping the torus water level to one foot below the normal operating level enables an inspection of the suppression chamber where problems would first begin to show.

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a LOCA. The peak drywell pressure would be about 49 psig which would rapidly reduce to less than 30 psig within 20 seconds following the pipe break. Following the pipe break, the suppression chamber pressure rises to 27 psig within 25 seconds, equalizes with drywell pressure, and decays with the drywell pressure decay.

The design pressure of the drywell and suppression chamber is 56 psig. The design leak rate is 0.5-percent per day at the pressure of 56 psig. As pointed out above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 25 seconds. Based on the calculated containment pressure response discussed above, the primary containment preoperational test pressures were chosen. Also based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The calculated radiological doses given in Section 14.9 of the FSAR were based on an assumed leakage rate of 0.635-percent at the maximum calculated pressure of 49.6 psig. The doses calculated by the NRC using this bases are 0.14 rem, whole body passing cloud gamma dose, and 15.0 rem, thyroid dose, which are respectively only 5×10^{-3} and 10^{-1} times the 10 CFR 100 reference doses. Increasing the assumed leakage rate at 49.6 psig to 2.0 percent as indicated in the specifications would increase these doses approximately a factor of three, still leaving a margin between the calculated dose and the 10 CFR 100 reference values.

Establishing the test limit of 2.0-percent/day provides an adequate margin of safety to assure the health and safety of the general public. It is further considered that the allowable leak rate should not deviate significantly from the containment design value to take advantage of the design leak-tightness capability of the structure over its service lifetime. Additional margin to maintain the containment in the "as built" condition is achieved by establishing the allowable operational leak rate. The allowable operational leak rate is derived by multiplying the maximum allowable leak rate by 0.75 thereby providing a 25-percent margin to allow for leakage deterioration which may occur during the period between leak rate tests.

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 137
License No. DPR-52

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated May 15, 1987, 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-52 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 137, 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 and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Gary G. Zech, Assistant Director
for Projects
TVA Projects Division
Office of Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 3, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 137

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

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	INSERT
3.7/4.7-3	3.7/4.7-3
3.7/4.7-4	3.7/4.7-4
3.7/4.7-5	3.7/4.7-5
3.7/4.7-6	3.7/4.7-6*
3.7/4.7-11	3.7/4.7-11
3.7/4.7-12	3.7/4.7-12*
3.7/4.7-45	3.7/4.7-45
3.7/4.7-46	3.7/4.7-46

LIMITING CONDITIONS FOR OPERATION

3.7.A. Primary Containment

2.a. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics tests at power levels not to exceed 5 MW(t).

b. Primary containment integrity is confirmed if the maximum allowable integrated leakage rate, L_a, does not exceed the equivalent of 2 percent of the primary containment volume per 24 hours at the 49.6 psig design basis accident pressure, P_a.

c. If N_2 makeup to the primary containment averaged over 24 hours (corrected for pressure, temperature, and venting operations) exceeds 542 SCFH, it must be reduced to < 542 SCFH within 8 hours or the reactor shall be placed in Hot Shutdown within the next 16 hours.

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

2. Integrated Leak Rate Testing

Primary containment nitrogen consumption shall be monitored to determine the average daily nitrogen consumption for the last 24 hours. Excessive leakage is indicated by a N_2 consumption rate of > 2% of the primary containment free volume per 24 hours (corrected for drywell temperature, pressure, and venting operations) at 49.6 psig. Corrected to normal drywell operating pressure of 1.1 psig, this value is 542 SCFH. If this value is exceeded, the action specified in 3.7.A.2.C shall be taken.

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 to 10 CFR 50 using the methods and provisions of ANSI N45.4 (1972).

a. Three type A tests (overall integrated containment leakage rate) shall be conducted at 40 ± 10-month intervals during shutdown at P_a, 49.6 psig, during each 10-year plant inservice inspection.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

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 0.75L_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. 1. Test duration shall be at least 8 hours.

2. A 4-hour stabilization period will be required and the containment atmosphere will be considered stabilized when the change in weighted average air temperature averaged over an hour does not deviate by more than 0.5°R/hour from the average rate of change of temperature measured from the previous 4 hours.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

d. 1. At least 20 sets of data points at approximately equal time intervals and in no case at intervals greater than one hour shall be provided for proper statistical analysis.

2. The figure of merit for the instrumentation system shall never exceed 0.25 L_a.

e. The test shall not be concluded with an increasing calculated leak rate.

f. The accuracy of each type A test shall be verified by a supplemental test which:

1. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the type A test data is within 0.25 L_a.

2. Has 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 (49.6 psig).

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

g. Local leak rate tests (LLRTs) shall be performed on the primary containment testable penetrations and isolation valves, which are not part of a water-sealed system, at not less than 49.6 psig (except for the main steam isolation valves, see 4.7.A.2.i) and not less than 54.6 psig for water-sealed valves each operating cycle. Bolted double-gasketed seals shall be tested whenever the seal is closed after being opened and at least once per operating cycle. Acceptable methods of testing are halide gas detection, soap bubbles, pressure decay, hydrostatically pressurized fluid flow or equivalent.

The personnel air lock shall be tested at 6-month intervals at an internal pressure of not less than 49.6 psig. In addition, if the personnel air lock is opened during periods when containment integrity is not required, a test at the end of such a period will be conducted at not less than 49.6 psig. If the personnel air lock is opened during a period when containment integrity is required, a test at $\geq$ 2.5 psig shall be conducted within 3 days after being opened. If the air lock is opened more frequently than once every 3 days, the air lock shall be tested at least once every 3 days during the period of frequent openings.

LIMITING CONDITIONS FOR OPERATION

3.7.A Primary Containment

3.7.A.4.b (Cont'd)

as it is determined to be not more than 3° open as indicated by the position lights.

c. Two drywell-suppression chamber vacuum breakers may be determined to be INOPERABLE for opening.

d. If Specifications 3.7.A.4.a, .b, or .c cannot be met, the unit shall be placed in a Cold Shutdown condition in an orderly manner within 24 hours.

5. Oxygen Concentration

a. After completion of the fire-related startup retesting program, containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100 psig, except as specified in 3.7.A.5.b.

b. Within the 24-hour period subsequent to placing the reactor in the RUN mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.

c. If Specification 3.7.A.5.a and 3.7.A.5.b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.A Primary Containment

4.7.A.4.b (Cont'd)

time when operability is required all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the INOPERABLE valve has been returned to normal service.

c. Once each operating cycle each vacuum breaker valve shall be inspected for proper operation of the valve and limit switches.

d. A leak test of the drywell to suppression chamber structure shall be conducted during each operating cycle. Acceptable leak rate is 0.09 lb/sec of primary containment atmosphere with 1 psi differential.

5. Oxygen Concentration

a. The primary containment oxygen concentration shall be measured and recorded daily. The oxygen measurement shall be adjusted to account for the uncertainty of the method used by adding a predetermined error function.

b. The methods used to measure the primary containment oxygen concentration shall be calibrated once every refueling cycle.

LIMITING CONDITIONS FOR OPERATION

3.7.A. Primary Containment

6. Drywell-Suppression Chamber Differential Pressure

a. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.1 psid except as specified in (1) and (2) below:

(1) This differential shall be established within 24 hours of achieving operating temperature and pressure. The differential pressure may be reduced to less than 1.1 psid 24 hours prior to a scheduled shutdown.

(2) This differential may be decreased to less than 1.1 psid for a maximum of four hours during required operability testing of the HPCI system, RCIC system and the drywell-pressure suppression chamber vacuum breakers.

b. If the differential pressure of Specification 3.7.A.6.a cannot be maintained and the differential pressure cannot be restored within the subsequent six-hour period, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

6. Drywell-Suppression Chamber Differential Pressure

a. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift.

be inoperable or the valve disc is stuck. For this case, a check light on and red light off confirms the disc is in a nearly closed position even if one of the indications is in error. Although the valve may be inoperable for full closure, it does not constitute a safety threat.

If the red light circuit alone is inoperable, the valve shall still be considered fully operable. If the green and red or the green light circuit alone is inoperable the valve shall be considered inoperable for opening. If the check and green or check light circuit alone is inoperable, the valve shall be considered inoperable for full closure. If the red and check light circuits are inoperable the valve shall be considered inoperable and open greater than 3°. For a light circuit to be considered operable the light must go on and off in proper sequence during the opening-closing cycle. If none of the lights change indication during the cycle, the valve shall be considered inoperable and open unless the check light stays on and the red light stays off in which case the valve shall be considered inoperable for opening.

The 12 drywell vacuum breaker valves which connect the suppression chamber and drywell are sized on the basis of the Bodega pressure suppression system tests. Ten operable to open vacuum breaker valves (18-inch) selected on this test basis and confirmed by the green lights are adequate to limit the pressure differential between the suppression chamber and drywell during postaccident drywell cooling operations to a value which is within suppression system design values.

The containment design has been examined to determine that a leakage equivalent to one drywell vacuum breaker opened to no more than a nominal 3° as confirmed by the red light is acceptable.

On this basis an indefinite allowable repair time for an inoperable red light circuit on any valve or an inoperable check and green or check light circuit alone or a malfunction of the operator or disc (if nearly closed) on one valve, or an inoperable green and red or green light circuit alone on two valves is justified.

During each operating cycle, a leak rate test shall be performed to verify that significant leakage flow paths do not exist between the drywell and suppression chamber. The drywell pressure will be increased by at least one psi with respect to the suppression chamber pressure and held constant. The two psig setpoint will not be exceeded. The subsequent suppression chamber pressure transient (if any) will be monitored with a sensitive pressure gauge. If the drywell pressure cannot be increased by one psi over the suppression chamber pressure it would be because a significant leakage path exists; in this event the leakage source will be identified and eliminated before power operation is resumed.

With a differential pressure of greater than one psig, the rate of change of the suppression chamber pressure must not exceed 0.25 inches of water per minute as measured over a 10-minute period, which corresponds to about 0.09 lb/sec of containment air. In the event the rate of change exceeds this value then the source of leakage will be identified and eliminated before power operation is resumed.

The water in the suppression chamber is used for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a daily check of the temperature and volume is adequate to assure that adequate heat removal capability is present.

The interior surfaces of the drywell and suppression chamber are coated as necessary to provide corrosion protection and to provide a more easily decontaminable surface. The surveillance inspection of the internal surfaces each operating cycle assures timely detection of corrosion. Dropping the torus water level to one foot below the normal operating level enables an inspection of the suppression chamber where problems would first begin to show.

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a LOCA. The peak drywell pressure would be about 49 psig which would rapidly reduce to less than 30 psig within 20 seconds following the pipe break. Following the pipe break, the suppression chamber pressure rises to 27 psig within 25 seconds, equalizes with drywell pressure, and decays with the drywell pressure decay.

The design pressure of the drywell and suppression chamber is 56 psig. The design leak rate is 0.5-percent per day at the pressure of 56 psig. As pointed out above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 25 seconds. Based on the calculated containment pressure response discussed above, the primary containment preoperational test pressures were chosen. Also based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The calculated radiological doses given in Section 14.9 of the FSAR were based on an assumed leakage rate of 0.635-percent at the maximum calculated pressure of 49.6 psig. The doses calculated by the NRC using this bases are 0.14 rem, whole body passing cloud gamma dose, and 15.0 rem, thyroid dose, which are respectively only 5×10^{-3} and 10^{-1} times the 10 CFR 100 reference doses. Increasing the assumed leakage rate at 49.6 psig to 2.0 percent as indicated in the specifications would increase these doses approximately a factor of three, still leaving a margin between the calculated dose and the 10 CFR 100 reference values.

Establishing the test limit of 2.0-percent/day provides an adequate margin of safety to assure the health and safety of the general public. It is further considered that the allowable leak rate should not deviate significantly from the containment design value to take advantage of the design leak-tightness capability of the structure over its service lifetime. Additional margin to maintain the containment in the "as built" condition is achieved by establishing the allowable operational leak rate. The allowable operational leak rate is derived by multiplying the maximum allowable leak rate by 0.75 thereby providing a 25-percent margin to allow for leakage deterioration which may occur during the period between leak rate tests.

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 112
License No. DPR-68

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated May 15, 1987, 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-68 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 112, 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 and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Gary G. Zech, Assistant Director
for Projects
TVA Projects Division
Office of Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 3, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 112

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

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	INSERT
3.7/4.7-3	3.7/4.7-3
3.7/4.7-4	3.7/4.7-4
3.7/4.7-5	3.7/4.7-5
3.7/4.7-6	3.7/4.7-6*
3.7/4.7-11	3.7/4.7-11
3.7/4.7-12	3.7/4.7-12*
3.7/4.7-43	3.7/4.7-43
3.7/4.7-44	3.7/4.7-44

LIMITING CONDITIONS FOR OPERATION

3.7.A. Primary Containment

2.a. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics tests at power levels not to exceed 5 MW(t).

b. Primary containment integrity is confirmed if the maximum allowable integrated leakage rate, L_a, does not exceed the equivalent of 2 percent of the primary containment volume per 24 hours at the 49.6 psig design basis accident pressure, P_a.

c. If N_2 makeup to the primary containment averaged over 24 hours (corrected for pressure, temperature, and venting operations) exceeds 542 SCFH, it must be reduced to < 542 SCFH within 8 hours or the reactor shall be placed in Hot Shutdown within the next 16 hours.

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

2. Integrated Leak Rate Testing

Primary containment nitrogen consumption shall be monitored to determine the average daily nitrogen consumption for the last 24 hours. Excessive leakage is indicated by a N_2 consumption rate of > 2% of the primary containment free volume per 24 hours (corrected for drywell temperature, pressure, and venting operations) at 49.6 psig. Corrected to normal drywell operating pressure of 1.1 psig, this value is 542 SCFH. If this value is exceeded, the action specified in 3.7.A.2.c shall be taken.

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 to 10 CFR 50 using the methods and provisions of ANSI N45.4 (1972).

a. Three type A tests (overall integrated containment leakage rate) shall be conducted at 40 ± 10-month intervals during shutdown at P_a, 49.6 psig, during each 10-year plant inservice inspection.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

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. 1. Test duration shall be at least 8 hours.

2. A 4-hour stabilization period will be required and the containment atmosphere will be considered stabilized when the change in weighted average air temperature averaged over an hour does not deviate by more than 0.5°R/hour from the average rate of change of temperature measured from the previous 4 hours.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

d. 1. At least 20 sets of data points at approximately equal time intervals and in no case at intervals greater than one hour shall be provided for proper statistical analysis.

2. The figure of merit for the instrumentation system shall never exceed 0.25 L_a.

e. The test shall not be concluded with an increasing calculated leak rate.

f. The accuracy of each type A test shall be verified by a supplemental test which:

1. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the type A test data is within 0.25 L_a.

2. Has 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 (49.6 psig).

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

g. Local leak rate tests (LLRTs) shall be performed on the primary containment testable penetrations and isolation valves, which are not part of a water-sealed system, at not less than 49.6 psig (except for the main steam isolation valves, see 4.7.A.2.i) and not less than 54.6 psig for water-sealed valves each operating cycle. Bolted double-gasketed seals shall be tested whenever the seal is closed after being opened and at least once per operating cycle. Acceptable methods of testing are halide gas detection, soap bubbles, pressure decay, hydrostatically pressurized fluid flow or equivalent.

The personnel air lock shall be tested at 6-month intervals at an internal pressure of not less than 49.6 psig. In addition, if the personnel air lock is opened during periods when containment integrity is not required, a test at the end of such a period will be conducted at not less than 49.6 psig. If the personnel air lock is opened during a period when containment integrity is required, a test at $\geq$ 2.5 psig shall be conducted within 3 days after being opened. If the air lock is opened more frequently than once every 3 days, the air lock shall be tested at least once every 3 days during the period of frequent openings.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.A Primary Containment	4.7.A Primary Containment
3.7.A.4.b (Cont'd)	4.7.A.4.b (Cont'd)
as it is determined to be not more than 3° open as indicated by the position lights.	time when operability is required, all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the INOPERABLE valve has been returned to normal service.
c. Two drywell-suppression chamber vacuum breakers may be determined to be INOPERABLE for opening.	c. Once each operating cycle, each vacuum breaker valve shall be inspected for proper operation of the valve and limit switches.
d. If Specifications 3.7.A.4.a, 3.7.A.4.b, or 3.7.A.4.c, cannot be met, the unit shall be placed in a Cold Shutdown condition in an orderly manner within 24 hours.	d. A leak test of the drywell to suppression chamber structure shall be conducted during each operating cycle. Acceptable leak rate is 0.09 lb/sec of primary containment atmosphere with 1 psi differential.
5. Oxygen Concentration	5. Oxygen Concentration
a. After completion of the 300-hour warranty run, containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100 psig, except as specified in 3.7.A.5.b.	a. The primary containment oxygen concentration shall be measured and recorded daily. The oxygen measurement shall be adjusted to account for the uncertainty of the method used by adding a predetermined error function.
b. Within the 24-hour period subsequent to placing the reactor in the RUN mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.	b. The methods used to measure the primary containment oxygen concentration shall be calibrated once every refueling cycle.
c. If the specifications of 3.7.A.5.a through 3.7.A.5.b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.A. Primary Containment	4.7.A. Primary Containment
6. Drywell-Suppression Chamber Differential Pressure	6. Drywell-Suppression Chamber Differential Pressure
a. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.1 psid except as specified in (1) and (2) below: (1) This differential shall be established within 24 hours of achieving operating temperature and pressure. The differential pressure may be reduced to less than 1.1 psid 24 hours prior to a scheduled shutdown. (2) This differential may be decreased to less than 1.1 psid for a maximum of four hours during required operability testing of the HPCI system, RCIC system and the drywell-pressure suppression chamber vacuum breakers.	a. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift.
b. If the differential pressure of Specification 3.7.A.6.a cannot be maintained and the differential pressure cannot be restored within the subsequent six-hour period, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours.	

be inoperable or the valve disc is stuck. For this case, a check light on and red light off confirms the disc is in a nearly closed position even if one of the indications is in error. Although a valve may be inoperable for full closure, it does not constitute a safety threat.

If the red light circuit alone is inoperable, the valve shall still be considered fully operable. If the green and red or the green light circuit alone is inoperable the valve shall be considered inoperable for opening. If the check and green or check light circuit alone is inoperable, the valve shall be considered inoperable for full closure. If the red and check light circuits are inoperable the valve shall be considered inoperable and open greater than 3°. For a light circuit to be considered operable the light must go on and off in proper sequence during the opening-closing cycle. If none of the lights change indication during the cycle, the valve shall be considered inoperable and open unless the check light stays on and the red light stays off in which case the valve shall be considered inoperable for opening.

The 12 drywell vacuum breaker valves which connect the suppression chamber and drywell are sized on the basis of the Bodega pressure suppression system tests. Ten operable to open vacuum breaker valves (18-inch) selected on this test basis and confirmed by the green lights are adequate to limit the pressure differential between the suppression chamber and drywell during postaccident drywell cooling operations to a value which is within suppression system design values.

The containment design has been examined to determine that a leakage equivalent to one drywell vacuum breaker opened to no more than a nominal 3° as confirmed by the red light is acceptable.

On this basis an indefinite allowable repair time for an inoperable red light circuit on any valve or an inoperable check and green or check light circuit alone or a malfunction of the operator or disc (if nearly closed) on one valve, or an inoperable green and red or green light circuit alone on two valves is justified.

During each operating cycle, a leak rate test shall be performed to verify that significant leakage flow paths do not exist between the drywell and suppression chamber. The drywell pressure will be increased by at least one psi with respect to the suppression chamber pressure and held constant. The two psig setpoint will not be exceeded. The subsequent suppression chamber pressure transient (if any) will be monitored with a sensitive pressure gauge. If the drywell pressure cannot be increased by one psi over the suppression chamber pressure it would be because a significant leakage path exists; in this event the leakage source will be identified and eliminated before power operation is resumed.

With a differential pressure of greater than one psig, the rate of change of the suppression chamber pressure must not exceed 0.25 inches of water per minute as measured over a 10-minute period, which corresponds to about 0.09 lb/sec of containment air. In the event the rate of change exceeds this value then the source of leakage will be identified and eliminated before power operation is resumed.

The water in the suppression chamber is used for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a daily check of the temperature and volume is adequate to assure that adequate heat removal capability is present.

The interior surfaces of the drywell and suppression chamber are coated as necessary to provide corrosion protection and to provide a more easily decontaminable surface. The surveillance inspection of the internal surfaces each operating cycle assures timely detection of corrosion. Dropping the torus water level to one foot below the normal operating level enables an inspection of the suppression chamber where problems would first begin to show.

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a loss-of-coolant accident. The peak drywell pressure would be about 49 psig which would rapidly reduce to less than 30 psig within 20 seconds following the pipe break. Following the pipe break, the suppression chamber pressure rises to 27 psig within 25 seconds, equalizes with drywell pressure, and decays with the drywell pressure decay.

The design pressure of the drywell and suppression chamber is 56 psig. The design leak rate is 0.5-percent per day at the pressure of 56 psig. As pointed out above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 25 seconds. Based on the calculated containment pressure response discussed above, the primary containment preoperational test pressures were chosen. Also based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The calculated radiological doses given in Section 14.9 of the FSAR were based on an assumed leakage rate of 0.635-percent at the maximum calculated pressure of 49.6 psig. The doses calculated by the NRC using this Bases are 0.14 rem, whole body passing cloud gamma dose, and 15.0 rem, thyroid dose, which are respectively only 5×10^{-3} and 10^{-1} times the 10 CFR 100 reference doses. Increasing the assumed leakage rate at 49.6 psig to 2.0 percent as indicated in the specifications would increase these doses approximately a factor of three, still leaving a margin between the calculated dose and the 10 CFR 100 reference values.

Establishing the test limit of 2.0-percent/day provides an adequate margin of safety to assure the health and safety of the general public. It is further considered that the allowable leak rate should not deviate significantly from the containment design value to take advantage of the design leak-tightness capability of the structure over its service lifetime. Additional margin to maintain the containment in the "as built" condition is achieved by establishing the allowable operational leak rate. The allowable operational leak rate is derived by multiplying the maximum allowable leak rate by 0.75 thereby providing a 25-percent margin to allow for leakage deterioration which may occur during the period between leak rate tests.

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

SAFETY EVALUATION BY THE OFFICE OF SPECIAL PROJECTS

SUPPORTING AMENDMENT NO. 141 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 137 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 112 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2 AND 3

DOCKET NOS. 50-259, 50-260 AND 50-296

1.0 INTRODUCTION

By letter dated May 15, 1987, the Tennessee Valley Authority (the licensee) requested changes to the Browns Ferry Nuclear Plant, Units 1, 2 and 3 Technical Specifications (TS). The first change involves deleting the option in the TS to perform a reduced pressure test method for the integrated leak rate test (ILRT). The second change involves a correction of the acceptable leak rate limit of the drywell atmosphere to the suppression chamber with a 1 psi differential pressure.

2.0 EVALUATION

The proposed amendments would change the Browns Ferry Units 1, 2, and 3 TS to delete references to an optional reduced pressure test method for the ILRT. The full pressure test requirements would not be affected. This revision would be made in Surveillance Requirements (SR) 4.7.A.2.a, 4.7.A.2.b, 4.7.A.2.f.1, and 4.7.A.2.f.3. It is also reflected in the bases for Section 4.7.A.

It was formerly believed that a correlation existed between data obtained from the performance of a reduced pressure test (25 psig) and a full pressure test (49.6 psig), allowing a prediction of the full pressure leakage by the performance of a reduced pressure test only. However, experience at Browns Ferry has shown that no correlation exists between the results of each test. Therefore, the licensee proposes to delete the low pressure test for clarity and conciseness. The controlling factor for leak rate testing at Browns Ferry is the requirement to meet 10 CFR Part 50, Appendix J. Although Appendix J currently allows periodic ILRT to be conducted at a reduced test pressure, TVA currently conducts all periodic containment ILRT at full test pressure in compliance with the requirements of Appendix J to 10 CFR Part 50. TVA has conducted preoperational ILRT at both pressures with the purpose of establishing a relationship between the measured leakage at half pressure and full pressure. However, these test results do not show a clear correlation between the reduced pressure and full pressure leakage. In fact, test results

showed increased leakage occurring due to the inception of leaks brought on by increasing pressure, demonstrating that leakage paths closed at lower pressure may open at full pressure. Deleting the option to perform an ILRT at reduced pressure is a conservative change that would not reduce the margin of safety.

A second change would be made to SR 4.7.A.4.d and the bases to correct the acceptable leak rate of drywell atmosphere to the suppression chamber with a 1 psi differential pressure. The limits currently stated in the TS are 0.38 inch of water per minute pressure change in the suppression pool, which corresponds to the 0.14 pound per second of containment air leakage specified in SR 4.7.A.4.d. The correct limits should be 0.25 inch of water per minute and 0.09 pound per second of containment air. The change to the acceptable leakage rate is to correct an error in accordance with TVA's response, (U5.1-5, Section 3.3) to the Atomic Energy Commission (AEC) (predecessor to the Nuclear Regulatory Commission), question 5.1 dated December 6, 1971.

The original TS listed the acceptable drywell to suppression chamber leakage as 0.25 inch of water per minute pressure change and 0.14 pound per second. These numbers were supposed to correspond to each other, but they do not. The licensee realized that an inconsistency existed and therefore, the TS were changed by amendment nos. 114, 108, and 82, for Browns Ferry, Units 1, 2, and 3, respectively. However, this change was based on the assumption that the limit of 0.14 pound per second given in TS 4.7.A.4.d was correct and the 0.25 inch of water per minute stated in the bases was wrong. The opposite of that previous assumption is actually true. These amendments are based on TVA's response to the AEC question 5.1 which states a limit of 0.25 inch of water per minute in the acceptability section (U5.1-5, Section 3.3). It should also be noted that the proposed changes are in the conservative direction in that it would allow less leakage. Therefore, this change would not reduce the margin of safety.

Based on the above evaluation, the staff finds the proposed changes to the TS are conservative and will increase the margin of safety. Therefore, the proposed changes to the TS are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

The amendments involve a change to 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 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 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: John Stang

Dated: February 3, 1988