DISTRIBUTION
Docket File
NRC PDR
Local PDR
OBR#2 Rdg.
D. Eisenhut
D. Clark
S. Norris
OELD
SECY
L. J. Harmon 6
T. Barnhart 12
L. Schneider
D. Brinkman
ACRS 10
OPA Clare Miles
R. Diggs
NSIC
ASLAB
Gray
5 extra

February 14, 1983

Docket Nos. 50-259
50-260
and 50-296

Mr. Hugh G. Parris
Manager of Power
Tennessee Valley Authority
500 Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Dear Mr. Parris:

The Commission has issued the enclosed Amendment Nos. 86, 83 and 57 to Facility License Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Unit Nos. 1, 2 and 3. These amendments are in response to your application dated August 19, 1980 (TVA BFNP TS 146), as supplemented by your resubmittal of December 3, 1982.

The amendments change the Technical Specifications to (1) reduce the duration of containment integrated leak rate tests by following the procedures outlined in Bechtel Topical Report BN-TOP-1, (2) revise the requirements in Sections 3.7.2 and 4.7.2 to conform the leak rate testing to the BWR/4 Standard Technical Specifications and (3) update the tables on isolation valves that are required to be tested.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 86 to DPR-33
2. Amendment No. 83 to DPR-52
3. Amendment No. 57 to DPR-68
4. Safety Evaluation
5. Notice

cc w/enclosures
See next page

AMDT. + FRN ONLY

OFFICE	DL:ORB#2	DL:ORB#2	DL:ORB#2	DL:OR	OELD		
SURNAME	S.Norris	D.Clark:pr	D.Vassallo	G.Lainas	Goddard		
DATE	2/14/83	2/14/83	2/14/83	2/14/83	2/14/83		

cc:

H. S. Sanger, Jr., Esquire
General Counsel
Tennessee Valley Authority
400 Commerce Avenue
E 11B 33C
Knoxville, Tennessee 37902

Mr. Ron Rogers
Tennessee Valley Authority
400 Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Mr. Charles R. Christopher
Chairman, Limestone County Commission
P. O. Box 188
Athens, Alabama 35611

Ira L. Myers, M.D.
State Health Officer
State Department of Public Health
State Office Building
Montgomery, Alabama 36104

Mr. H. N. Culver
249A HBD
400 Commerce Avenue
Tennessee Valley Authority
Knoxville, Tennessee 37902

U. S. Environmental Protection
Agency
Region IV Office
Regional Radiation Representative
345 Courtland Street
Atlanta, Georgia 30308

Resident Inspector
U. S. Nuclear Regulatory Commission
Route 2, Box 311
Athens, Alabama 35611

Mr. Donald L. Williams, Jr.
Tennessee Valley Authority
400 West Summit Hill Dr., W10B85
Knoxville, Tennessee 37902

George Jones
Tennessee Valley Authority
P. O. Box 2000
Decatur, Alabama 35602

Mr. Oliver Havens
U.S. Nuclear Regulatory Commission
Reactor Training Center
Osborne Office Center, Suite 200
Chattanooga, Tennessee 37411

James P. O'Reilly
Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 86
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 August 19, 1980, as supplemented by letter dated December 3, 1982, 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 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. 86, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Within 90 days after the effective date of this amendment, or such later time as the Commission may specify, the Licensee shall satisfy any applicable requirement of P.L. 97-425 related to pursuing an agreement with the Secretary of Energy for the disposal of high-level radioactive waste and spent nuclear fuel.

4. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

DBVassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 14, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 86

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Revise Appendix A as follows:

1. Remove the following pages and replace with the identically numbered pages.

229
230
231
232
258
259
260
261
261a
262
263
264
267

The marginal lines on these pages denote the area being changed.

2. Add the following new pages:

229a
232a
259a

3.7 CONTAINMENT SYSTEMS

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 5MW(t).

b. Primary containment integrity is confirmed if the maximum allowable integrated leakage rate, La, 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, Pa.

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.

4.7 CONTAINMENT SYSTEMS

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 is542 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 either P_a, 49.6 psig, or at P_t,25 psig, during each 10-year plant inservice inspection.

b. If any periodic type A test fails to meet either 0.75 L_a or 0.75 L_t 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 either 0.75L_a or 0.75 L_t, a type A test shall be performed at least every 18 months until two consecutive

LIMITING CONDITIONS FOR OPERATION

3.7 CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS

type A tests meet either 0.75 L_a or $0.75L_t$, 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

3.7 CONTAINMENT SYSTEMS

4.7 CONTAINMENT SYSTEMS

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.

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 or 0.25 L_t.

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), or P_t (25 psig).

3.7 CONTAINMENT SYSTEMS

4.7 CONTAINMENT SYSTEMS

g. Local Leak rate tests (LLRT's) 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 a pressure of 49.6 psig during each operating cycle. In addition, following each opening, the personnel air lock shall be leak tested at a pressure of ≥2.5 psig within 72 hours of the first of each series of openings whenever containment integrity is required. The personnel air lock shall be leak tested at a pressure of ≥2.5 psig at least once every 6 months from the first of each series of openings to verify the

3.7 CONTAINMENT SYSTEMS

4.7 CONTAINMENT SYSTEMS

condition of the air lock assembly whenever containment integrity is required. The total leakage from all penetrations and isolation valves shall not exceed 60 percent of L_a per 24 hours. Leakage from containment isolation valves that terminate below suppression pool water level may be excluded from the total leakage provided a sufficient fluid inventory is available to ensure the sealing function for at least 30 days at a pressure of 54.6 psig. Leakage from containment isolation valves that are in closed-loop, seismic class I lines that will be water sealed during a DBA will be measured but will be excluded when computing the total leakage. Penetrations and isolation valves are identified as follows:

(1) Testable penetrations with double O-ring seals - Table 3.7.B,

(2) Testable penetrations with testable bellows Table 3.7.C,

(3) Isolation valves without fluid seal - Table 3.7.D,

(4) Testable electrical penetrations - Table 3.7.H, and

(5) Isolation valves sealed with fluid - Tables 3.7.E, and 3.7.F.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7 CONTAINMENT SYSTEMS	4.7 CONTAINMENT SYSTEMS
	h. (1) If at any time it is determined that the criterion of 4.7.A.2.g is exceeded, repairs shall be initiated immediately.
	(2) If conformance to the criterion of 4.7.A.2.g is not demonstrated

TABLE 3.7.D

AIR TESTED ISOLATION VALVES

Valve	Valve Identification
1-14	Main Steam
1-15	Main Steam
1-26	Main Steam
1-27	Main Steam
1-37	Main Steam
1-38	Main Steam
1-51	Main Steam
1-52	Main Steam
1-55	Main Steam Drain
1-56	Main Steam Drain
2-1192	Service Water
2-1383	Service Water
3-554	Feedwater
3-558	Feedwater
3-568	Feedwater
3-572	Feedwater
32-62	Drywell Compressor Suction
32-63	Drywell Compressor Suction
32-336	Drywell Compressor Return
32-2163	Drywell Compressor Return
33-1070	Service Air
33-785	Service Air
43-13	Reactor Water Sample Lines
43-14	Reactor Water Sample Lines
63-525	Standby Liquid Control Discharge
63-526	Standby Liquid Control Discharge
64-17	Drywell and Suppression Chamber Air Purge Inlet
64-18	Drywell Air Purge Inlet
64-19	Suppression Chamber Air Purge Inlet
64-20	Suppression Chamber Vacuum Relief
64-c.v.	Suppression Chamber Vacuum Relief
64-21	Suppression Chamber Vacuum Relief
64-c.v.	Suppression Chamber Vacuum Relief
64-29	Drywell Main Exhaust
64-30	Drywell Main Exhaust
64-32	Suppression Chamber Main Exhaust
64-33	Suppression Chamber Main Exhaust
64-31	Drywell exhaust to Standby Gas Treatment
64-34	Suppression Chamber to Standby Gas Treatment
64-139	Drywell pressurization, Compressor Suction
64-140	Drywell pressurization, Compressor Discharge
68-508	CRD to RC Pump Seals
68-523	CRD to RC Pump Seals
68-550	CRD to RC Pump Seals
68-555	CRD to RC Pump Seals

TABLE 3.7.D (Continued)

Valve	Valve Identification
69-1	RWCU Supply
69-2	RWCU Supply
69-579	RWCU Return
71-2	RCIC Steam Supply
71- 3	RCIC Steam Supply
71-39	RCIC Pump Discharge
71-40	RCIC Pump Discharge
73-2	RCIC Steam Supply
73-3	RCIC Steam Supply
73-44	HPCI Pump Discharge
73-45	HPCI Pump Discharge
73-81	HPCI Steam Supply Bypass
74-47	RHR Shutdown Suction
74-48	RHR Shutdown Suction
74-661	RHR Shutdown Suction
74-662	RHR Shutdown Suction
76-17	Drywell/Suppression Chamber Nitrogen Purge
76-18	Drywell Nitrogen Purge Inlet
76-19	Suppression Chamber Purge Inlet
76-24	Drywell/Suppression Chamber Nitrogen Purge
76-49	Containment Atmospheric Monitor
76-50	Containment Atmospheric Monitor
76-51	Containment Atmospheric Monitor
76-52	Containment Atmospheric Monitor
76-53	Containment Atmospheric Monitor
76-54	Containment Atmospheric Monitor
76-55	Containment Atmospheric Monitor
76-56	Containment Atmospheric Monitor
76-57	Containment Atmospheric Monitor
76-58	Containment Atmospheric Monitor
76-59	Containment Atmospheric Monitor
76-60	Containment Atmospheric Monitor
76-61	Containment Atmospheric Monitor
76-62	Containment Atmospheric Monitor
76-63	Containment Atmospheric Monitor
76-64	Containment Atmospheric Monitor
76-65	Containment Atmospheric Monitor
76-66	Containment Atmospheric Monitor
76-67	Containment Atmospheric Monitor
76-68	Containment Atmospheric Monitor
77-2A	Drywell Floordrain Sump
77-2B	Drywell Floordrain Sump
77-15A	Drywell Equipment Drain Sump
77-15B	Drywell Equipment Drain Sump
84-8A	Containment Atmospheric Dilution
84-8B	Containment Atmospheric Dilution
84-8C	Containment Atmospheric Dilution
84-8D	Containment Atmospheric Dilution
84-19	Containment Atmospheric Dilution
84-20	Main Exhaust to Standby Gas Treatment
84-600	Main Exhaust to Standby Gas Treatment
84-601	Main Exhaust to Standby Gas Treatment
84-602	Main Exhaust to Standby Gas Treatment

TABLE 3.7.D (Continued)

Valve	Valve Identification
84-603	Main Exhaust to Standby Gas Treatment
85-576	CRD Hydraulic Return
90-254A	Radiation Monitor Suction
90-254B	Radiation Monitor Suction
90-255	Radiation Monitor Suction
90-257A	Radiation Monitor Discharge
90-257B	Radiation Monitor Discharge

(DELETED)

TABLE 3.7.E

PRIMARY CONTAINMENT ISOLATION VALVES WHICH TERMINATE BELOW THE SUPPRESSION POOL WATER LEVEL

Valve	Valve Identification
12-738	Auxiliary Boiler to RCIC
12-741	Auxiliary Boiler to RCIC
43-28A	RHR Suppression Chamber Sample Lines
43-28B	RHR Suppression Chamber Sample Lines
43-29A	RHR Suppression Chamber Sample Lines
43-29B	RHR Suppression Chamber Sample Lines
2-1143	Demineralized Water
71-14	RCIC Turbine Exhaust
71-32	RCIC Vacuum Pump Discharge
71-580	RCIC Turbine Exhaust
71-592	RCIC Vacuum Pump Discharge
73-23	HPCI Turbine Exhaust
73-24	HPCI Turbine Exhaust Drain
73-603	HPCI Turbine Exhaust
73-609	HPCI Exhaust Drain
74-722	RHR
75-57	Core Spray to Auxiliary Boiler
75-58	Core Spray to Auxiliary Boiler
	Core Spray to Auxiliary Boiler

TABLE 3.7.F

PRIMARY CONTAINMENT ISOLATION VALVES LOCATED IN
WATER SEALED SEISMIC CLASS 1 LINES

Valve	Valve Identification
74-53	RHR LPCI Discharge
74-54	RHR
74-57	RHR Suppression Chamber Spray
74-58	RHR Suppression Chamber Spray
74-60	RHR Drywell Spray
74-61	RHR Drywell Spray
74-67	RHR LPCI Discharge
74-68	RHR LPCI Discharge
74-71	RHR Suppression Chamber Spray
74-72	RHR Suppression Chamber Spray
74-74	RHR Drywell Spray
74-75	RHR Drywell Spray
74-77	RHR Head Spray
74-78	RHR Head Spray
75-25	Core Spray Discharge
75-26	Core Spray Discharge
75-53	Core Spray Discharge
75-54	Core Spray Discharge

TABLE 3.7.G

(This table intentionally left blank)

3.7.A & 4.7.A Primary Containment

The integrity of the primary containment and operation of the core standby cooling system in combination, ensure 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 Part 100 during accident conditions.

During initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required, there will be no pressure on the system thus greatly reducing the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect to minimize the probability of an accident occuring.

The limitations on primary 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 of 49.6 psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited 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 Part 50 (type A, B, and C tests).

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat release during primary system blowdown from 1,035 psig. Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting form isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water levels given in the specification, containment pressure during the design basis accident is approximately 49 psig, which is below the maximum of 62 psig. The maximum water level indications of -1 inch corresponds to a downcomer submergence of 3 feet 7 inches and a water volume of 127,800 cubic feet with or 128,700 cubic feet without the drywell-suppression chamber differential pressure control. The minimum water level indication of -6.25 inches with differential pressure control and -7.25 inches without differential pressure control corresponds to a downcomer submergence of approximately 3 feet and water volume of approximately 123,000 cubic feet. Maintaining the water level between these levels will ensure that the torus water volume and downcomer submergence are within the aforementioned limits during normal plant operation. Alarms, adjusted for instrument error, will notify the operator when the limits of the torus water level are approached. The majority of the Bodega tests were run with a submerged length of 4 feet and with complete condensation. Thus, with response to downcomer submergence, this specification is adequate. The maximum temperature at the end of blowdown tested during the Humboldt Bay and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 83
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 August 19, 1980, as supplemented by letter dated December 3, 1982, 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 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. 83, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Within 90 days after the effective date of this amendment, or such later time as the Commission may specify, the Licensee shall satisfy any applicable requirement of P.L. 97-425 related to pursuing an agreement with the Secretary of Energy for the disposal of high-level radioactive waste and spent nuclear fuel.

4. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 14, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 83

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise Appendix A as follows:

1. Remove the following pages and replace with identically numbered pages.

229
230
231
232
267

The marginal lines on these pages denote the area being changed.

2. Add the following new pages.

229a
232a

3.7 CONTAINMENT SYSTEMS

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 5MW(t).

b. Primary containment integrity is confirmed if the maximum allowable integrated leakage rate, La, 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, Pa.

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.

4.7 CONTAINMENT SYSTEMS

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 either P_a, 49.6 psig, or at P_t,25 psig, during each 10-year plant inservice inspection.

b. If any periodic type A test fails to meet either 0.75 L_a or 0.75 L_t 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 either 0.75L_a or 0.75 L_t, a type A test shall be performed at least every 18 months until two consecutive

3.7 CONTAINMENT SYSTEMS

4.7 CONTAINMENT SYSTEMS

type A tests meet either 0.75 L_a or $0.75L_t$, 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

3.7 CONTAINMENT SYSTEMS

4.7 CONTAINMENT SYSTEMS

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.

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 or 0.25 L_t.

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), or P_t (25 psig).

3.7 CONTAINMENT SYSTEMS

4.7 CONTAINMENT SYSTEMS

g. Local Leak rate tests (LLRT's) 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 a pressure of 49.6 psig during each operating cycle. In addition, following each opening, the personnel air lock shall be leak tested at a pressure of ≥2.5 psig within 72 hours of the first of each series of openings whenever containment integrity is required. The personnel air lock shall be leak tested at a pressure of ≥2.5 psig at least once every 6 months from the first of each series of openings to verify the

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7 CONTAINMENT SYSTEMS	4.7 CONTAINMENT SYSTEMS

condition of the air lock assembly whenever containment integrity is required. The total leakage from all penetrations and isolation valves shall not exceed 60 percent of L_a per 24 hours. Leakage from containment isolation valves that terminate below suppression pool water level may be excluded from the total leakage provided a sufficient fluid inventory is available to ensure the sealing function for at least 30 days at a pressure of 54.6 psig. Leakage from containment isolation valves that are in closed-loop, seismic class I lines that will be water sealed during a DBA will be measured but will be excluded when computing the total leakage. Penetrations and isolation valves are identified as follows:

(1) Testable penetrations with double O-ring seals - Table 3.7.B,

(2) Testable penetrations with testable bellows Table 3.7.C,

(3) Isolation valves without fluid seal - Table 3.7.D,

(4) Testable electrical penetrations - Table 3.7.H, and

(5) Isolation valves sealed with fluid - Tables 3.7.E, and 3.7.F.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7 CONTAINMENT SYSTEMS	4.7 CONTAINMENT SYSTEMS
	h. (1) If at any time it is determined that the criterion of 4.7.A.2.g is exceeded, repairs shall be initiated immediately.
	(2) If conformance to the criterion of 4.7.A.2.g is not demonstrated

3.7.A & 4.7.A Primary Containment

The integrity of the primary containment and operation of the core standby cooling system in combination, ensure 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 Part 100 during accident conditions.

During initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required, there will be no pressure on the system thus greatly reducing the changes of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect to minimize the probability of an accident occuring.

The limitations on primary 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 of 49.6 psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited 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 Part 50 (type A, B, and C tests).

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat release during primary system blowdown from 1,035 psig. Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting form isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water levels given in the specification, containment pressure during the design basis accident is approximately 49 psig, which is below the maximum of 62 psig. The maximum water level indications of -1 inch corresponds to a downcomer submergence of 4 feet 7 inches and a water volume of 129,000 cubic feet with or without the drywell-suppression chamber differential pressure control. The minimum water level indication of -7 inches with differential pressure control and -8 inches without differential pressure control corresponds to a downcomer submergence of approximately 3 feet and water volume of approximately 123,000 cubic feet. Maintaining the water level between these levels will ensure that the torus water volume and downcomer submergence are within the aforementioned limits during normal plant operation. Alarms, adjusted for instrument error, will notify the operator when the limits of the torus water level are approached. The majority of the Bodega tests were run with a submerged length of 4 feet and with complete condensation. Thus, with response to downcomer submergence, this specification is adequate. The maximum temperature at the end of blowdown tested during the Humboldt Bay and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 57
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 August 19, 1980, as supplemented by letter dated December 3, 1982, 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 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. 57, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Within 90 days after the effective date of this amendment, or such later time as the Commission may specify, the Licensee shall satisfy any applicable requirement of P.L. 97-425 related to pursuing an agreement with the Secretary of Energy for the disposal of high-level radioactive waste and spent nuclear fuel.

4. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

DBVassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 14, 1983

ATTACHMENT TO LICENSE AMENDMENT NO.

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise A-pendix A as follows:

1. Remove the following pages and replace with identically numbered pages:

233
234
235
236
237
238
239
240
241
285

The marginal lines on the above pages indicate the area being changed.

3.7 CONTAINMENT SYSTEMS

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 5MW(t).

b. Primary containment integrity is confirmed if the maximum allowable integrated leakage rate, La, 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, Pa.

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.

4.7 CONTAINMENT SYSTEMS

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 either P_a, 49.6 psig, or at P_t,25 psig, during each 10-year plant inservice inspection.

b. If any periodic type A test fails to meet either 0.75 L_a or 0.75 L_t 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 either $0.75L_a$ or 0.75 L_t, a type A test shall be performed at least every 18 months until two consecutive

3.7 CONTAINMENT SYSTEMS

4.7 CONTAINMENT SYSTEMS

type A tests meet either 0.75 L_a or $0.75L_t$, 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

3.7 CONTAINMENT SYSTEMS

4.7 CONTAINMENT SYSTEMS

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.

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 or 0.25 L_t.

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), or P_t (25 psig).

3.7 CONTAINMENT SYSTEMS

4.7 CONTAINMENT SYSTEMS

g. Local Leak rate tests (LLRT's) 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 then 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 ≥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

3.7 CONTAINMENT SYSTEMS

4.7 CONTAINMENT SYSTEMS

3 days during the period of frequent openings.

The total leakage from all penetrations and isolation valves shall not exceed 60 percent of L_a per 24 hours. Leakage from containment isolation valves that terminate below suppression pool water level may be excluded from the total leakage provided a sufficient fluid inventory is available to ensure the sealing function for at least 30 days at a pressure of 54.6 psig. Leakage from containment isolation valves that are in closed-loop, seismic class I lines that will be water sealed during a DBA will be measured but will be excluded when computing the total leakage. Penetrations and isolation valves are identified as follows:

(1) Testable penetrations with double O-ring seals - Table 3.7.B,

(2) Testable penetrations with testable bellows Table 3.7.C,

(3) Isolation valves without fluid seal - Table 3.7.D,

(4) Testable electrical penetrations - Table 3.7.H, and

(5) Isolation valves sealed with fluid - Tables 3.7.E, and 3.7.F.

(DELETED)

3.7 CONTAINMENT SYSTEMS

4.7 CONTAINMENT SYSTEMS

h. (1) If at any time it is determined that the criterion of 4.7.A.2.g is exceeded, repairs shall be initiated immediately.

(2) If conformance to the criterion of 4.7.A.2.g is not demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by re-test.

i. The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling outage. If the leakage rate of 11.5 scf/hr for any one main steamline isolation valve is exceeded, repairs and retest shall be performed to correct the condition.

j. Continuous Leak Rate Monitoring

When the primary containment is inerted, the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. This monitoring

BASES

3.7.A & 4.7.A Primary Containment

The integrity of the primary containment and operation of the core standby cooling system in combination, ensure 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 Part 100 during accident conditions.

During initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required, there will be no pressure on the system thus greatly reducing the changes of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect to minimize the probability of an accident occuring.

The limitations on primary 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 of 49.6 psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited 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 Part 50 (type A, B, and C tests).

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat release during primary system blowdown from 1,035 psig. Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting form isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water levels given in the specification, containment pressure during the design basis accident is approximately 49 psig, which is below the maximum of 62 psig. The maximum water level indications of -1 inch corresponds to a downcomer submergence of 3 feet 7 inches and a water volume of 127,800 cubic feet with or 128,700 cubic feet without the drywell-suppression chamber differential pressure control. The minimum water level indication of -6.25 inches with differential pressure control and -7.25 inches without differential pressure control corresponds to a downcomer submergence of approximately 3 feet and water volume of approximately 123,000 cubic feet. Maintaining the water level between these levels will ensure that the torus water volume and downcomer submergence are within the aforementioned limits during normal plant operation. Alarms, adjusted for instrument error, will notify the operator when the limits of the torus water level are approached. The majority of the Bodega tests were run with a submerged length of 4 feet and with complete condensation. Thus, with response to downcomer submergence, this specification is adequate. The maximum temperature at the end of blowdown tested during the Humboldt Bay and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 86 TO FACILITY OPERATION LICENSE NO. DPR-33

AMENDMENT NO. 83 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 57 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT NOS. 1, 2 AND 3

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

1.0 Introduction

By letter dated August 19, 1980, the Tennessee Valley Authority (the licensee or TVA) requested amendments to Facility Operating License Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Unit Nos. 1, 2 and 3. In response to discussions with the TVA staff, the application was supplemented by a resubmittal on December 3, 1982. The proposed amendments would change the Technical Specifications to: (1) reduce the duration of containment integrated leak rate tests by following the procedures outlined in Bechtel Topical Report BN-TOP-1, (2) revise the requirements in Sections 3.7.2 and 4.7.2 to conform the leak rate testing to the BWR/4 Standard Technical Specifications and (3) update the tables on isolation valves that are required to be tested.

2.0 Evaluation

One of the proposed changes would reduce the duration of the Containment Integrated Leak Rate Test (CILRT) from 24 hours to 8 hours. The Licensee has committed to follow the procedures outlined in Bechtel Topical Report BN-TOP-1, dated November 1, 1972, which allows for an 8-hour CILRT. This report was evaluated and approved by the staff in a Topical Report Evaluation dated February 1, 1973. The licensee's submittal satisfies the leak testing requirements of 10 CFR 50, Appendix J and ANSI N45.4-1972 with regard to test methods, procedures and frequency of testing. We, therefore, conclude that the proposed change to the Technical Specifications reducing the duration of the CILRT from 24 hours to 8 hours is acceptable.

In conjunction with the above change, TVA has proposed revisions and reformatting the limiting conditions for operation (LCO) and surveillance requirements on leak rate testing in Sections 3.7.2 and 4.7.2 of the Technical Specifications. The staff concludes these proposed changes are acceptable since they are consistent with the BWR/4 Standard Technical Specifications.

There was one change in the present requirements (Section 3.7.2.c, P229) which is to change the drywell operating pressure from 1.5 psig to 1.1 psig and the allowable leakage from 549 SCFH to 542 SCFH (2% of containment volume). This adds conservatism to the allowable leakage rate. This is done to ensure that the test method does not produce nonconservative results. The change also reflects the extensive modifications made to the units as a result of the generic Mark I torus modifications. The proposed change is more conservative and is acceptable.

The proposed changes to Tables 3.7.D, E, and F, which list the containment isolation valves, were discussed with the Licensee in a telephone conference on January 31, 1983. The Licensee stated that valve number 64-141, on the Drywall Pressurization Compressor Line, was deleted from the original tables, since it had been incorrectly identified as an isolation valve. The correct isolation valves for that line were confirmed to be numbers 64-139 and 64-140 in Table 3.7.D by the Licensee, during the telephone conference. Six additional valves have been classified as isolation valves in the updated tables, including valve numbers 2-1143, 2-1192, 2-1383, 33-785, 33-1070, and 74-722. The Licensee indicated that these revisions were made as a result of recent plant modifications.

The Licensee has also proposed to air test certain isolation valves that were previously water tested. Appendix J 10 CFR 50 specifies air testing as the recommended leak testing method except for those valves that are fluid sealed. In addition, the staff considers air testing of valves to be a more conservative method than water testing. On the basis of the requirements of 10 CFR 50, Appendix J and clarifications provided by the Licensee during the telephone conference of January 31, 1983,the staff concludes that the above proposed changes to the Technical Specifications are acceptable.

The staff concludes, on the bases detailed above, that the proposed changes to Sections 3.7.2 and 4.7.2 of the Technical Specifications for Browns Ferry, Units 1, 2, and 3, as detailed in the Licensee's submittal of December 3, 1982, are acceptable.

3.0 Environmental Considerations

We have determined that these amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that these amendments involve an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR Section 51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

4.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated, do not create the possibility of an accident of a type different from any evaluated previously, and do not involve a significant reduction in a margin of safety, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: J. R. Hall

Dated: February 14, 1983

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 86 to Facility Operating License No. DPR-33, Amendment No. 83 to Facility Operating License No. DPR-52 and Amendment No. 57 to Facility Operating License No. DPR-68 issued to Tennessee Valley Authority (the licensee), for operation of the Browns Ferry Nuclear Plant, Unit Nos. 1, 2 and 3, located in Limestone County, Alabama. The amendments are effective as of the date of issuance.

The amendments change the Technical Specifications to: (1) reduce the duration of containment integrated leak rate tests by following the procedures outlined in Bechtel topical report BN-TOP-1, (2) revise the requirements in Sections 3.7.2 and 4.7.2 to conform the leak rate testing to the BWR/4 Standard Technical Specifications and (3) update the tables on isolation valves that are required to be tested.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR Section 51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated August 19, 1980, as supplemented by letter dated December 3, 1982, (2) Amendment No. 86 to License No. DPR-33, Amendment No. 83 to License No. DPR-52, and Amendment No. 57to License No. DPR-68, and (3) the Commission's related Safety Evaluation. All of these are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Athens Public Library, South and Forrest, Athens, Alabama 35611. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 14th day of February, 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

DBVassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing