ENCLOSURE

PROPOSED TECHNICAL SPECIFICATIONS
BROWNS FERRY NUCLEAR PLANT
(TVA BFNP TS 146)

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 549 SCFH, it must be reduced to 549 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 once each 8-hour shift 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 1.5 psig, this value is 549SCFH. 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 contairment 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.

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
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 Inlet
76-18	Drywell Nitrogen Purge Inlet
76-19	Suppression Chamber Purge Inlet
76-24	Drywell/Suppression Chamber Nitrogen Purge Inlet
76-215	Containment Atmospheric Monitor
76-217	Containment Atmospheric Monitor
76-220	Containment Atmospheric Monitor
76-222	Containment Atmospheric Monitor
76-225	Containment Atmospheric Monitor
76-226	Containment Atmospheric Monitor
76-229	Containment Atmospheric Monitor
76-230	Containment Atmospheric Monitor
76-237	Containment Atmospheric Monitor
76-239	Containment Atmospheric Monitor
76-242	Containment Atmospheric Monitor
76-243	Containment Atmospheric Monitor
76-248	Containment Atmospheric Monitor
76-250	Containment Atmospheric Monitor
76-253	Containment Atmospheric Monitor
76-254	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
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)

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 released 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 from 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 indication 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 4 feet and water volume of approximately 123,000 cubic eet. Maintaining the water level between these levels will assure that the torus water volume and downcomer submergency are with 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.

UNIT 2

PROPOSED CHANGES

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 549 SCFH, it must be reduced to 549 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 once each 8-hour shift 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 1.5 psig, this value is 549SCFH. 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

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

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
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 Inlet
76-18	Drywell Nitrogen Purge Inlet
76-19	Suppression Chamber Purge Inlet
76-24	Drywell/Suppression Chamber Nitrogen Purge Inlet
76-215	Containment Atmospheric Monitor
76-217	Containment Atmospheric Monitor
76-220	Containment Atmospheric Monitor
76-222	Containment Atmospheric Monitor
76-225	Containment Atmospheric Monitor
76-226	Containment Atmospheric Monitor
76-229	Containment Atmospheric Monitor
76-230	Containment Atmospheric Monitor
76-237	Containment Atmospheric Monitor
76-239	Containment Atmospheric Monitor
76-242	Containment Atmospheric Monitor
76-243	Containment Atmospheric Monitor
76-248	Containment Atmospheric Monitor
76-250	Containment Atmospheric Monitor
76-253	Containment Atmospheric Monitor
76-254	Containment Atmospheric Monitor
77-2A	Drywell Floor Drain Sump
77-2B	Drywell Floor Drain 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
84-603	Main Exhaust to Standby Gas Treatment

TABLE 3.7.D (continued)

Valve	Valve Identification
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)

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 released 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 from 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 indication 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 4 feet and water volume of approximately 123,000 cubic eet. Maintaining the water level between these levels will assure that the torus water volume and downcomer submergency are with 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.

3.7 CONTAINMENT SYSTEMS	4.7 CONTAINMENT SYSTEMS

UNIT 3

PROPOSED CHANGES

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 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.

(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

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
69-624	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
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 Inlet
76-18	Drywell Nitrogen Purge Inlet
76-19	Suppression Chamber Purge Inlet
76-24	Drywell/Suppression Chamber Nitrogen Purge Inlet
76-49	Containment Inerting
76-50	Containment Inerting
76-51	Containment Inerting
76-52	Containment Inerting
76-59	Containment Inerting
76-60	Containment Inerting
76-61	Containment Inerting
76-62	Containment Inerting
76-63	Containment Inerting
76-64	Containment Inerting
76-65	Containment Inerting
76-66	Containment Inerting
76-67	Containment Inerting
76-68	Containment Inerting
76-215	Containment Atmospheric Monitor
76-217	Containment Atmospheric Monitor
76-220	Containment Atmospheric Monitor
76-222	Containment Atmospheric Monitor
76-225	Containment Atmospheric Monitor
76-226	Containment Atmospheric Monitor
76-229	Containment Atmospheric Monitor
76-230	Containment Atmospheric Monitor
76-237	Containment Atmospheric Monitor
76-239	Containment Atmospheric Monitor

TABLE 3.7.D (continued)

Valve	Valve Identification
76-242	Containment Atmospheric Monitor
76-243	Containment Atmospheric Monitor
76-248	Containment Atmospheric Monitor
76-250	Containment Atmospheric Monitor
76-253	Containment Atmospheric Monitor
76-254	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
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)

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 released 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 from 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 indication 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 4 feet and water volume of approximately 123,000 cubic feet. Maintaining the water level between these levels will assure that the torus water volume and downcomer submergency are with 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.