ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS-364
MARKED PAGES

I. **AFFECTED PAGE LIST**

Unit 1	Unit 2	Unit 3
3.7/4.7-3	3.7/4.7-3	3.7/4.7-3
3.7/4.7-4	3.7/4.7-4	3.7/4.7-4
3.7/4.7-5	3.7/4.7-5	3.7/4.7-5
3.7/4.7-6	3.7/4.7-6	3.7/4.7-6
3.7/4.7-7	3.7/4.7-7	3.7/4.7-7
3.7/4.7-25	3.7/4.7-25	3.7/4.7-24
6.0-24	6.0-23c	6.0-23c

II. **MARKED PAGES**

See attached.

DEC 0 7 1994

LIMITING CONDITIONS FOR OPERATION

3.7.A. Primary Containment

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

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

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

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

2. Integrated Leak Rate Testing

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

A

~~The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in accordance with Appendix J to 10 CFR 50 as modified by approved exemptions.~~

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

Insert A

INSERT A

TS 4.7.A.2 SECOND PARAGRAPH

Perform leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

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LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

b. If any periodic type A test fails to meet 0.75 L_a, the test schedule for subsequent type A tests shall be reviewed and approved by the Commission.

If two consecutive type A tests fail to meet $0.75L_a$, a type A test shall be performed at least every 18 months until two consecutive type A tests meet 0.75 L_a, at which time the above test schedule may be resumed.

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

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

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

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

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

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

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

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

2. Has duration sufficient to establish accurately the change in leakage rate between the type A test and the supplemental test.

3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage at P_a (49.6 psig).

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

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

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

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2.g (Cont'd)

~~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.~~

Insert B

INSERT B

PARAGRAPH 4.7.A.2.g

Perform required local leak rate tests, including the primary containment air lock leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

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

The acceptance criteria for air lock testing are: (1) Overall air lock leakage rate is ≤ (0.05 L_a) when tested at ≥ Pa. (2) For door seal leakage, the overall air lock leakage rate is ≤ (0.02 L_a) when the air lock is pressurized to (≥ 2.5 psig for at least 15 minutes).

 NOV 1 6 1992

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 occurring.

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 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 indications of -1 inch corresponds to a downcomer submergence of three feet seven 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 three feet and water volume of approximately 123,000 cubic feet.

accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

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6.8.5 PROGRAMS

Postaccident Sampling

Postaccident sampling activities will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. These activities shall include the following:

(i) Training of personnel,
(ii) Procedures for sampling and analysis,
(iii) Provisions for maintenance of sampling and analysis.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Director of the Regional Office of NRC, unless otherwise noted.

6.9.1.1 STARTUP REPORT

a. A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a

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Section 6.8.4.3 PRIMARY CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test program, dated September 1995".

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 49.6 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 2% of primary containment air weight per day.

Leakage Rate acceptance criteria are:

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

b. Air lock testing acceptance criteria are:

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

   2) Air lock door seals leakage rate is $\leq 0.02\ L_a$ when the overall air lock is pressurized to ≥ 2.5 psig for at least 15 minutes.

DEC 0 7 1994

LIMITING CONDITIONS FOR OPERATION

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

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

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

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~~The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in accordance with Appendix J to 10 CFR 50 as modified by approved exemptions.~~

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

Insert A

INSERT A

TS 4.7.A.2 SECOND PARAGRAPH

Perform leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

b. If any periodic type A test fails to meet 0.75 L_a, the test schedule for subsequent type A tests shall be reviewed and approved by the Commission.

If two consecutive type A tests fail to meet $0.75L_a$, a type A test shall be performed at least every 18 months until two consecutive type A tests meet 0.75 L_a, at which time the above test schedule may be resumed.

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

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

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

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

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

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

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

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

2. ~~Has duration sufficient to establish accurately the change in leakage rate between the type A test and the supplemental test.~~

3. ~~Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage at P_a (49.6 psig).~~

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

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

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

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2.g (Cont'd)

~~The total path 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.~~

Insert B

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PARAGRAPH 4.7.A.2.g

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

The acceptance criteria for air lock testing are: (1) Overall air lock leakage rate is ≤ (0.05 L_a) when tested at ≥ Pa. (2) For door seal leakage, the overall air lock leakage rate is ≤ (0.02 L_a) when the air lock is pressurized to (≥ 2.5 psig for at least 15 minutes).

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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 occurring.

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.

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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 three feet seven 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 three feet and a water volume of approximately 123,000 cubic feet.

j. Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

6.8.4.2 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

a. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,

b. A Land Use Census to ensure that changes in the use of area at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

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

→ Insert C

6.8.5 PROGRAMS

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Postaccident sampling activities will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and

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Leakage Rate acceptance criteria are:

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

b. Air lock testing acceptance criteria are:

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

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LIMITING CONDITIONS FOR OPERATION

3.7.A. Primary Containment

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

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

SURVEILLANCE REQUIREMENTS

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Perform leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

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4.7.A.2. (Cont'd)

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LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

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4.7.A.2. (Cont'd)

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LIMITING CONDITIONS FOR OPERATION

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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 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 indications of -1 inch corresponds to a downcomer submergence of three feet seven 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 three feet and water volume of approximately 123,000 cubic feet.

than 8 days in gaseous effluents released from each unit to areas beyond the ~~SITE~~ BOUNDARY conforming to Appendix I to 10 CFR Part 50.

j. Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

6.8.4.2 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

a. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,

b. A Land Use Census to ensure that changes in the use of area at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

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

→ Insert C

INSERT C

Section 6.8.4.3 PRIMARY CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test program, dated September 1995".

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 49.6 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 2% of primary containment air weight per day.

Leakage Rate acceptance criteria are:

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

b. Air lock testing acceptance criteria are:

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

   2) Air lock door seals leakage rate is $\leq 0.02\ L_a$ when the overall air lock is pressurized to $\geq$ 2.5 psig for at least 15 minutes.

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS-364
REVISED PAGES

I. AFFECTED PAGE LIST

Unit 1	Unit 2	Unit 3
3.7/4.7-3	3.7/4.7-3	3.7/4.7-3
3.7/4.7-4	3.7/4.7-4	3.7/4.7-4
3.7/4.7-5	3.7/4.7-5	3.7/4.7-5
3.7/4.7-6	3.7/4.7-6	3.7/4.7-6
3.7/4.7-7	3.7/4.7-7	3.7/4.7-7
3.7/4.7-25	3.7/4.7-25	3.7/4.7-24
6.0-23d	6.0-23c	6.0-23d
6.0-23e	6.0-23d	6.0-23e
	6.0-23e	

II. REVISED PAGES

See attached.

LIMITING CONDITIONS FOR OPERATION

3.7.A. Primary Containment

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

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

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

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

2. Integrated Leak Rate Testing

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

Perform leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	4.7.A. Primary Containment 4.7.A.2. (Cont'd) b. Deleted c. Deleted

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	4.7.A. Primary Containment 4.7.A.2. (Cont'd) d. Deleted e. Deleted f. Deleted

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	4.7.A. Primary Containment

4.7.A.2. (Cont'd)

g. Perform required local leak rate tests, including the primary containment air lock leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

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

The acceptance criteria for air lock testing are: (1) Overall air lock leakage rate is $\leq$ (0.05 L_a) when tested at $\geq$ Pa. (2) For door seal leakage, the overall air lock leakage rate is $\leq$ (0.02 L_a) when the air lock is pressurized to ($\geq$ 2.5 psig for at least 15 minutes).

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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 occurring.

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 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 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 indications of -1 inch corresponds to a downcomer submergence of three feet seven 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 three feet and water volume of approximately 123,000 cubic feet.

accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

6.8.4.3 PRIMARY CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test program, dated September 1995".

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 49.6 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 2% of primary containment air weight per day.

Leakage Rate acceptance criteria are:

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

b. Air lock testing acceptance criteria are:

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

   (2) Air lock door seals leakage rate is $\leq 0.02\ L_a$ when the overall air lock is pressurized to ≥ 2.5 psig for at least 15 minutes.

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6.8.5 PROGRAMS

Postaccident Sampling

Postaccident sampling activities will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. These activities shall include the following:

(i) Training of personnel,
(ii) Procedures for sampling and analysis,
(iii) Provisions for maintenance of sampling and analysis.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Director of the Regional Office of NRC, unless otherwise noted.

6.9.1.1 STARTUP REPORT

a. A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a

LIMITING CONDITIONS FOR OPERATION

3.7.A. Primary Containment

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

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

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

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

2. Integrated Leak Rate Testing

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

Perform leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	4.7.A. Primary Containment
	4.7.A.2. (Cont'd)
	b. Deleted
	c. Deleted

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	4.7.A. Primary Containment 4.7.A.2. (Cont'd) d. Deleted e. Deleted f. Deleted

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

g. Perform required local leak rate tests, including the primary containment air lock leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

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

The acceptance criteria for air lock testing are: (1) Overall air lock leakage rate is $\leq$ (0.05 L_a) when tested at $\geq$ Pa. (2) For door seal leakage, the overall air lock leakage rate is $\leq$ (0.02 L_a) when the air lock is pressurized to ($\geq$ 2.5 psig for at least 15 minutes).

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3.7/4.7 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 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 occurring.

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 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 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 indications of -1 inch corresponds to a downcomer submergence of three feet seven 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 three feet and a water volume of approximately 123,000 cubic feet.

j. Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

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a. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,

b. A Land Use Census to ensure that changes in the use of area at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

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

6.8.4.3 PRIMARY CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test program, dated September 1995".

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 49.6 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 2% of primary containment air weight per day.

Leakage Rate acceptance criteria are:

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

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

(2) Air lock door seals leakage rate is $\leq 0.02\ L_a$ when the overall air lock is pressurized to $\geq$ 2.5 psig for at least 15 minutes.

6.8.5 PROGRAMS

<u>Postaccident Sampling</u>

Postaccident sampling activities will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and

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LIMITING CONDITIONS FOR OPERATION

3.7.A. Primary Containment

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SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

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Perform leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	4.7.A. Primary Containment 4.7.A.2. (Cont'd) b. Deleted c. Deleted

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
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LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
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The acceptance criteria for air lock testing are: (1) Overall air lock leakage rate is $\leq$ (0.05 L_a) when tested at $\geq$ Pa. (2) For door seal leakage, the overall air lock leakage rate is $\leq$ (0.02 L_a) when the air lock is pressurized to ($\geq$ 2.5 psig for at least 15 minutes).

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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.

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