UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 228
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 December 8, 1995, and supplemented January 10, 1996, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-33 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 228, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Peter S. Tam for

Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: February 22, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 228

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. *Overleaf pages are provided to maintain document completeness.

REMOVE	INSERT
3.7/4.7-3	3.7/4.7-3
3.7/4.6-4	3.7/4.6-4
3.7/4.7-5	3.7/4.7-5
3.7/4.7-6	3.7/4.7-6
3.7/4.7-7	3.7/4.7-7
3/7/4.7-8	3.7/4.7-8*
3.7/4.7-25	3.7/4.7-25
3.7/4.7-26	3.7/4.7-26*
---	6.0-23d
---	6.0-23e
6.0-24	6.0-24
6.0-25	6.0-25*

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 door 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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LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

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 retest.

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.

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.

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 maximum permissible bulk pool temperature is limited by the potential for stable and complete condensation of steam discharged from safety relief valves and adequate core spray pump net positive suction head. At reactor vessel pressures above approximately 555 psig, the bulk pool temperature shall not exceed 180°F. At pressures below approximately 240 psig, the bulk temperature may be as much as 184°F. At intermediate pressures, linear interpolation of the bulk temperature is permitted.

They also represent the bounding upper limits that are used in suppression pool temperature response analyses for safety relief valve discharge and loss-of-coolant accident (LOCA) cases. The actions required by Specification 3.7.c., -d. -e. and -f., assure the reactor can be depressurized in a timely manner to avoid exceeding the maximum bulk suppression pool water limits. Furthermore, the 184°F limit provides that adequate RHR and core spray pump NPSH will be available without dependency on containment overpressure.

Should it be necessary to drain the suppression chamber, this should only be done when there is no requirement for core standby cooling systems OPERABILITY. Under full power operation conditions, blowdown from an initial suppression chamber water temperature of 95°F results in a peak long term water temperature which is sufficient for complete condensation.

Limiting suppression pool temperature to 105°F during RCIC, HPCI, or relief valve operation when decay heat and stored energy is removed from the primary system by discharging reactor steam directly to the suppression chamber ensures adequate margin for controlled blowdown anytime during RCIC operation and ensures margin for complete condensation of steam from the design basis LOCA.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

If a LOCA were to occur when the reactor water temperature is below approximately 330°F, the containment pressure will not exceed the 62 psig code permissible pressures even if no condensation were to occur. The maximum allowable pool temperature, whenever the reactor is above 212°F, shall be governed by this specification. Thus, specifying water volume-temperature requirements applicable for reactor-water temperature above 212°F provides additional margin above that available at 330°F.

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 different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

b. Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

6.9.1.2 ANNUAL OPERATING REPORT*

a. A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) for whom monitoring was required receiving annual deep dose equivalent exposures greater than 100 mrem and their associated man rem exposure according to work and job functions, **e.g., reactor operations and surveillance,

*A single submittal may be made for a multiple unit station.
**This tabulation supplements the requirements of 20.2206 of 10 CFR Part 20.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 243
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 December 8, 1995, and supplemented January 10, 1996, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-52 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 243, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Peter S. Tam for

Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: February 22, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 243

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. *Overleaf pages are provided to maintain document completeness.

REMOVE	INSERT
3.7/4.7-3	3.7/4.7-3
3.7/4.6-4	3.7/4.6-4
3.7/4.7-5	3.7/4.7-5
3.7/4.7-6	3.7/4.7-6
3.7/4.7-7	3.7/4.7-7
3.7/4.7-8	3.7/4.7-8*
3.7/4.7-25	3.7/4.7-25
3.7/4.7-26	3.7/4.7-26*
6.0-23b	6.0-23b*
6.0-23c	6.0-23c
---	6.0-23d
---	6.0-23e

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 door 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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LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

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 retest.

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.

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.

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 maximum permissible bulk pool temperature is limited by the potential for stable and complete condensation of steam discharged from safety relief valves and adequate core spray pump net positive suction head. At reactor vessel pressures above approximately 555 psig, the bulk pool temperature shall not exceed 180°F. At pressures below approximately 240 psig, the bulk temperature may be as much as 184°F. At intermediate pressures, linear interpolation of the bulk temperature is permitted.

They also represent the bounding upper limits that are used in suppression pool temperature response analyses for safety relief valve discharge and loss-of-coolant accident (LOCA) cases. The actions required by Specifications 3.7.C. - 3.7.F. assure the reactor can be depressurized in a timely manner to avoid exceeding the maximum bulk suppression pool water limits. Furthermore, the 184°F limit provides that adequate RHR and core spray pump NPSH will be available without dependency on containment overpressure.

Should it be necessary to drain the suppression chamber, this should only be done when there is no requirement for Core Standby Cooling Systems OPERABILITY. Under full power operation conditions, blowdown from an initial suppression chamber water temperature of 95°F results in a peak long term water temperature which is sufficient for complete condensation.

Limiting suppression pool temperature to 105°F during RCIC, HPCI, or relief valve operation when decay heat and stored energy is removed from the primary system by discharging reactor steam directly to the suppression chamber ensures adequate margin for controlled blowdown anytime during RCIC operation and ensures margin for complete condensation of steam from the design basis loss-of-coolant accident (LOCA).

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

If a LOCA were to occur when the reactor water temperature is below approximately 330°F, the containment pressure will not exceed the 62 psig code permissible pressures even if no condensation were to occur. The maximum allowable pool temperature, whenever the reactor is above 212°F, shall be governed by this specification. Thus, specifying water volume-temperature requirements applicable for reactor-water temperature above 212°F provides additional margin above that available at 330°F.

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current year in accordance with the methodology and parameters in the ODCM at least every 31 days.

f. Limitations on the OPERABILITY and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50.

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the SITE BOUNDARY shall be limited to the following:

   1. For noble gas: less than or equal to a dose rate of 500 mrem/yr to the total body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and

   2. For Iodine-131, Iodine-133, tritium, and for all radionuclides in particulate form with half lives greater than 8 days: less than or equal to a dose rate of 1500 mrem/yr to any organ.

h. Limitations of the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50.

i. Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater 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.

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.

6.8.5 PROGRAMS

Postaccident Sampling

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

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 203
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 December 8, 1995, and supplemented January 10, 1996, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-68 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 203 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Peter S. Tam for

Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: February 22, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 203

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. *Overleaf pages are provided to maintain document completeness.

REMOVE	INSERT
3.7/4.7-3	3.7/4.7-3
3.7/4.6-4	3.7/4.6-4
3.7/4.7-5	3.7/4.7-5
3.7/4.7-6	3.7/4.7-6
3.7/4.7-7	3.7/4.7-7
3.7/4.7-8	3.7/4.7-8*
3.7/4.7-24	3.7/4.7-24
3.7/4.7-25	3.7/4.7-25*
---	6.0-23d
---	6.0-23e

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. 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 door 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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LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

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 retest.

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.

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.

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 maximum permissible bulk pool temperature is limited by the potential for stable and complete condensation of steam discharged from safety relief valves and adequate core spray pump net positive suction head. At reactor vessel pressures above approximately 555 psig, the bulk pool temperature shall not exceed 180°F. At pressures below approximately 240 psig, the bulk temperature may be as much as 184°F. At intermediate pressures, linear interpolation of the bulk temperature is permitted.

They also represent the bounding upper limits that are used in suppression pool temperature response analyses for safety relief valve discharge and loss-of-coolant accident (LOCA) cases. The actions required by Specifications 3.7.C. - 3.7.F. assure the reactor can be depressurized in a timely manner to avoid exceeding the maximum bulk suppression pool water limits. Furthermore, the 184°F limit provides that adequate RHR and core spray pump NPSH will be available without dependency on containment overpressure.

Should it be necessary to drain the suppression chamber, this should only be done when there is no requirement for Core Standby Cooling Systems OPERABILITY. Under full power operation conditions, blowdown from an initial suppression chamber water temperature of 95°F results in a peak long term water temperature which is sufficient for complete condensation.

Limiting suppression pool temperature to 105°F during RCIC, HPCI, or relief valve operation when decay heat and stored energy is removed from the primary system by discharging reactor steam directly to the suppression chamber assures adequate margin for controlled blowdown anytime during RCIC operation and ensures margin for complete condensation of steam from the design basis loss-of-coolant accident (LOCA).

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

If a LOCA were to occur when the reactor water temperature is below approximately 330°F, the containment pressure will not exceed the 62 psig code permissible pressures even if no condensation were to occur. The maximum allowable pool temperature, whenever the reactor is above 212°F, shall be governed by this specification. Thus, specifying water volume-temperature requirements applicable for reactor-water temperature above 212°F provides additional margin above that available at 330°F.

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.

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