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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 176
License No. DPR-77

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated March 10, 1993, which was amended by letter dated January 31, 1994, 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-77 is hereby amended to read as follows:

   (2) Technical Speciiications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 176, 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, to be implemented within 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Mohan C. Thadani / for

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: February 10, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 176

FACILITY OPERATING LICENSE NO. DPR-77

DOCKET NO. 50-327

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.

REMOVE	INSERT
VII	VII
1-2	1-2
3/4 6-1	3/4 6-1
3/4 6-2	3/4 6-2
3/4 6-3	3/4 6-3
3/4 6-4	3/4 6-4
3/4 6-8	3/4 6-8
3/4 6-11	3/4 6-11
3/4 6-12	3/4 6-12
3/4 6-15	3/4 6-15
3/4 6-23	3/4 6-23
B3/4 6-1	B3/4 6-1
B3/4 6-2	B3/4 6-2
----	B3/4 6-2a

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

c. Digital channels - the injection of a simulated signal into the channel as close to the sensor input to the process racks as practicable to verify OPERABILITY including alarm and/or trip functions.

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:
   1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or
   2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-2 of Specification 3.6.3.

b. All equipment hatches are closed and sealed.

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are within the limits of Specification 4.6.1.1.c,

e. The sealing mechansim associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE, and

f. Secondary containment bypass leakage is within the limits of Specification 3.6.1.2.

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMIT REPORT

1.10 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.14. Unit operation within these operating limits is addressed in individual specifications.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-2 of Specification 3.6.3.

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

c. Perform required visual examinations and leakage rate testing at P_a in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions. The maximum allowable leakage rate, L_a, is 0.25% of containment air weight per day at the calculated peak containment pressure P_a, 12 psig.

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*Except valves, blind flanges, and deactivated automatic valves which are located inside the annulus or containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

CONTAINMENT SYSTEMS

SECONDARY CONTAINMENT BYPASS LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Secondary Containment bypass leakage rates shall be limited to a combined bypass leakage rate of less than or equal to 0.25 L_a for all penetrations identified in Table 3.6-1 as secondary containment BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the combined bypass leakage rate exceeding 0.25 L_a for BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING, restore the combined bypass leakage rate from BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING to less than or equal to 0.25 L_a within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SECONDARY CONTAINMENT BYPASS LEAKAGE

SURVEILLANCE REQUIREMENTS

4.6.1.2 The secondary containment bypass leakage rates shall be demonstrated:

a. The combined bypass leakage rate to the auxiliary building shall be determined to be less than or equal to 0.25 L_a by applicable Type B and C tests at least once per 24 months except for penetrations which are not individually testable; penetrations not individually testable shall be determined to have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a (12 psig) during each Type A test.*

b. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a (13.2 psig) and the seal system capacity is adequate to maintain system pressure (or fluid head for the containment spray system and RHR spray system valves at penetrations 48A, 48B, 49A and 49B) for at least 30 days.

c. The provisions of Specification 4.0.2 are not applicable.

*Results shall be evaluated against the acceptance criteria of Specification 4.6.1.1.c in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.

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SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. After each opening, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage less than or equal to 0.01 L_a as determined by precision flow measurements when measured for at least two minutes with the volume between the door seals at a pressure greater than or equal to 6 psig,

b. By conducting an overall air lock leakage test at not less than P_a (12 psig) and by verifying the overall air lock leakage rate is within the limit of Specification 3.6.1.3.b and the results evaluated in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions:#

   1. At least once per six months, and

   2. Prior to establishing CONTAINMENT INTEGRITY if opened when CONTAINMENT INTEGRITY was not required when maintenance has been performed on the air lock that could affect the air lock sealing capability.*

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

#The provisions of Specification 4.0.2 are not applicable.

*Exemption to Appendix "J" of 10 CFR 50.

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the containment vessel not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The structural integrity of the containment vessel shall be determined during the shutdown for each Type A containment leakage rate test (Specification 4.6.1.1.c) by a visual inspection of the exposed accessible interior and exterior surfaces of the vessel. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance of the surfaces or other abnormal degradation. Any abnormal degradation of the containment vessel detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.6.1.

SHIELD BUILDING STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.7 The structural integrity of the shield building shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.7.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the shield building not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The structural integrity of the shield building shall be determined during the shutdown for each Type A containment leakage rate test (Specification 4.6.1.1.c) by a visual inspection of the exposed accessible interior and exterior surfaces of the shield building and verifying no apparent changes in appearance of the concrete surfaces or other abnormal degradation. Any abnormal degradation of the shield building detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.6.1.

CONTAINMENT SYSTEMS

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.9 One pair (one purge supply line and one purge exhaust line) of containment purge system lines may be open; the containment purge supply and exhaust isolation valves in all other containment purge lines shall be closed. Operation with purge supply or exhaust isolation valves open for either purging or venting shall be limited to less than or equal to 1000 hours per 365 days. The 365 day cumulative time period will begin every January 1.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With a purge supply or exhaust isolation valve open in excess of the above cumulative limit, or with more than one pair of containment purge system lines open, close the isolation valve(s) in the purge line(s) within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a containment purge supply and/or exhaust isolation valve having a measured leakage rate in excess of 0.05 L_a, restore the inoperable valve to OPERABLE status within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.9.1 The position of the containment purge supply and exhaust isolation valves shall be determined at least once per 31 days.

4.6.1.9.2 The cumulative time that the purge supply and exhaust isolation valves are open over a 365 day period shall be determined at least once per 7 days.

4.6.1.9.3 At least once per 3 months, each containment purge supply and exhaust isolation valve shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.05 L_a.*

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*Results shall be evaluated against the acceptance criteria of Specification 4.6.1.1.c in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
C.	PHASE "A" CONTAINMENT VENT ISOLATION (Cont.)		
13.	FCV-30-50	Upper Compt Purge Air Exh	4*
14.	FCV-30-51	Upper Compt Purge Air Exh	4*
15.	FCV-30-52	Upper Compt Purge Air Exh	4*
16.	FCV-30-53	Upper Compt Purge Air Exh	4*
17.	FCV-30-56	Lower Compt Purge Air Exh	4*
18.	FCV-30-57	Lower Compt Purge Air Exh	4*
19.	FCV-30-58	Inst Room Purge Air Exh	4*
20.	FCV-30-59	Inst Room Purge Air Exh	4*
21.	FCV-90-107	Cntmt Bldg LWR Compt Air Mon	5*
22.	FCV-90-108	Cntmt Bldg LWR Compt Air Mon	5*
23.	FCV-90-109	Cntmt Bldg LWR Compt Air Mon	5*
24.	FCV-90-110	Cntmt Bldg LWR Compt Air Mon	5*
25.	FCV-90-111	Cntmt Bldg LWR Compt Air Mon	5*
26.	FCV-90-113	Cntmt Bldg UPR Compt Air Mon	5*
27.	FCV-90-114	Cntmt Bldg UPR Compt Air Mon	5*
28.	FCV-90-115	Cntmt Bldg UPR Compt Air Mon	5*
29.	FCV-90-116	Cntmt Bldg UPR Compt Air Mon	5*
30.	FCV-90-117	Cntmt Bldg UPR Compt Air Mon	5*
D.	OTHER		
1.	FCV-30-46	Vacuum Relief Isolation Valve	25
2.	FCV-30-47	Vacuum Relief Isolation Valve	25
3.	FCV-30-48	Vacuum Relief Isolation Valve	25
4.	FCV-62-90	Normal Charging Isolation Valve	12

*Provisions of LCO 3.0.4 are not applicable if valve is secured in its isolated position with power removed and leakage limits of Specification 4.6.1.1.c are satisfied. For purge valves, leakage limits under Surveillance Requirement 4.6.1.9.3 must also be satisfied.

#Provisions of LCO 3.0.4 are not applicable if valve is secured in its isolated position with power removed and either FCV-62-73 or FCV-62-74 is maintained operable.

**This valve is required after completion of the associated modification.

3/4.6.1 PRIMARY CONTAINMENT

The safety design basis for primary containment is that the containment must withstand the pressures and temperatures of the limiting design basis accident (DBA) without exceeding the design leakage rates.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA), a steam line break, and a rod ejection accident (REA). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. This leakage rate limitation will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions. The containment was designed with an allowable leakage rate of 0.25 percent of containment air weight per day. This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting DBA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.25 percent per day in the safety analysis at P_a = 12.0 psig. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between tests.

Primary containment INTEGRITY or operability is maintained by limiting leakage to within the acceptance criteria of 10 CFR 50, Appendix J.

Individual leakage rates specified for the containment air lock (LCO 3.6.1.3), purge valves (LCO 3.6.1.9) and secondary bypass leakage (LCO 3.6.1.2) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits do not result in the primary containment being inoperable unless the leakage, when combined with other Type B and C test leakages, exceeds the acceptance criteria of Appendix J.

3/4.6.1.2 SECONDARY CONTAINMENT BYPASS LEAKAGE

The safety design basis for containment leakage assumes that 75 percent of the leakage from the primary containment enters the shield building annulus for filtration by the emergency gas treatment system. The remaining 25 percent of the primary containment leakage, which is considered to be bypassed to the auxiliary building, is assumed to exhaust directly to the atmosphere without filtration during the first 5 minutes of the accident. After 5 minutes, any bypass leakage to the auxiliary building is filtered by the auxiliary building gas treatment system. A tabulation of potential secondary containment bypass leakage paths to the auxiliary building is provided in Table 3.6-1. Restricting the leakage through the bypass leakage paths in Table 3.6-1 to 0.25 L_a provides assurance that the leakage fraction assumptions used in the evaluation of site boundary radiation doses remain valid.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provide assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the annulus atmosphere of 0.5 psig and 2) the containment peak pressure does not exceed the maximum allowable internal pressure of 12 psig during LOCA conditions.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that 1) the containment air mass is limited to an initial mass sufficiently low to prevent exceeding the maximum allowable internal pressure during LOCA conditions and 2) the ambient air temperature does not exceed that temperature allowable for the continuous duty rating specified for equipment and instrumentation located within containment.

The containment pressure transient is sensitive to the initially contained air mass during a LOCA. The contained air mass increases with decreasing temperature. The lower temperature limits of 100°F for the lower compartment, 85°F for the upper compartment, and 60°F when less than or equal to 5% of RATED THERMAL POWER will limit the peak pressure to an acceptable value. The upper temperature limit influences the peak accident temperature slightly during a LOCA; however, this limit is based primarily upon equipment protection and anticipated operating conditions. Both the upper and lower temperature limits are consistent with the parameters used in the accident analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 12 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

BASES

3/4.6.1.7 SHIELD BUILDING STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment shield building will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to provide 1) protection for the steel vessel from external missiles, 2) radiation shielding in the event of a LOCA, and 3) and annulus surrounding the steel vessel that can be maintained at a negative pressure during accident conditions.

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 167
License No. DPR-77

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated March 10, 1993, which was amended by letter dated January 31, 1994, 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-79 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 167, 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, to be implemented within 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Mohan C. Thadani for

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: February 10, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 167

FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

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.

REMOVE	INSERT
VII	VII
1-2	1-2
3/4 6-1	3/4 6-1
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B3/4 6-1	B3/4 6-1
B3/4 6-2	B3/4 6-2
----	B3/4 6-2a

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

c. Digital channels - the injection of a simulated signal into the channel as close to the sensor input to the process racks as practicable to verify OPERABILITY including alarm and/or trip functions.

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-2 of Specification 3.6.3.

b. All equipment hatches are closed and sealed.

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are within the limits of Specification 4.6.1.1.c,

e. The sealing mechansim associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE, and

f. Secondary containment bypass leakage is within the limits of Specification 3.6.1.2.

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMIT REPORT

1.10 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.14. Unit operation within these operating limits is addressed in individual specifications.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-2 of Specification 3.6.3.

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

c. Perform required visual examinations and leakage rate testing at P_a in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions. The maximum allowable leakage rate, L_a, is 0.25% of containment air weight per day at the calculated peak containment pressure P_a, 12 psig.

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*Except valves, blind flanges, and deactivated automatic valves which are located inside the annulus or containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

SECONDARY CONTAINMENT BYPASS LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Secondary Containment bypass leakage rates shall be limited to a combined bypass leakage rate of less than or equal to 0.25 L_a for all penetrations identified in Table 3.6-1 as secondary containment BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the combined bypass leakage rate exceeding 0.25 L_a for BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING, restore the combined bypass leakage rate from BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING to less than or equal to 0.25 L_a within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SECONDARY CONTAINMENT BYPASS LEAKAGE

SURVEILLANCE REQUIREMENTS

4.6.1.2 The secondary containment bypass leakage rates shall be demonstrated:

a. The combined bypass leakage rate to the auxiliary building shall be determined to be less than or equal to 0.25 L_a by applicable Type B and C tests at least once per 24 months except for penetrations which are not individually testable; penetrations not individually testable shall be determined to have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a, (12 psig) during each Type A test.*

b. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a (13.2 psig) and the seal system capacity is adequate to maintain system pressure (or fluid head for the containment spray system and RHR spray system valves at penetrations 48A, 48B, 49A and 49B) for at least 30 days.

c. The provisions of Specification 4.0.2 are not applicable.

*Results shall be evaluated against the acceptance criteria of Specification 4.6.1.1.c in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.

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SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. After each opening, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage less than or equal to 0.01 L_a as determined by precision flow measurements when measured for at least two minutes with the volume between the door seals at a pressure greater than or equal to 6 psig,

b. By conducting an overall air lock leakage test at not less than P_a (12 psig) and by verifying the overall air lock leakage rate is within the limit of Specification 3.6.1.3.b and the results evaluated in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions:#

   1. At least once per six months, and

   2. Prior to establishing CONTAINMENT INTEGRITY if opened when CONTAINMENT INTEGRITY was not required when maintenance has been performed on the air lock that could affect the air lock sealing capability.*

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

#The provisions of Specification 4.0.2 are not applicable.

*Exemption to Appendix "J" of 10 CFR 50.

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the containment vessel not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The structural integrity of the containment vessel shall be determined during the shutdown for each Type A containment leakage rate test (Specification 4.6.1.1.c) by a visual inspection of the exposed accessible interior and exterior surfaces of the vessel. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance of the surfaces or other abnormal degradation. Any abnormal degradation of the containment vessel detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.6.1.

SHIELD BUILDING STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.7 The structural integrity of the shield building shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.7.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the shield building not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The structural integrity of the shield building shall be determined during the shutdown for each Type A containment leakage rate test (Specification 4.6.1.1.c) by a visual inspection of the exposed accessible interior and exterior surfaces of the shield building and verifying no apparent changes in appearance of the concrete surfaces or other abnormal degradation. Any abnormal degradation of the shield building detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.6.1.

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.9 One pair (one purge supply line and one purge exhaust line) of containment purge system lines may be open; the containment purge supply and exhaust isolation valves in all other containment purge lines shall be closed. Operation with purge supply or exhaust isolation valves open for either purging or venting shall be limited to less than or equal to 1000 hours per 365 days. The 365 day cumulative time period will begin every January 1.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With a purge supply or exhaust isolation valve open in excess of the above cumulative limit, or with more than one pair of containment purge system lines open, close the isolation valve(s) in the purge line(s) within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a containment purge supply and/or exhaust isolation valve having a measured leakage rate in excess of 0.05 L_a, restore the inoperable valve to OPERABLE status within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.9.1 The position of the containment purge supply and exhaust isolation valves shall be determined at least once per 31 days.

4.6.1.9.2 The cumulative time that the purge supply and exhaust isolation valves are open over a 365 day period shall be determined at least once per 7 days.

4.6.1.9.3 At least once per 3 months, each containment purge supply and exhaust isolation valve shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.05 L_a.*

*Results shall be evaluated against the acceptance criteria of Specification 4.6.1.1.c in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
C. PHASE "A" CONTAINMENT VENT ISOLATION (Cont.)		
13. FCV-30-50	Upper Compt Purge Air Exh	4*
14. FCV-30-51	Upper Compt Purge Air Exh	4*
15. FCV-30-52	Upper Compt Purge Air Exh	4*
16. FCV-30-53	Upper Compt Purge Air Exh	4*
17. FCV-30-56	Lower Compt Purge Air Exh	4*
18. FCV-30-57	Lower Compt Purge Air Exh	4*
19. FCV-30-58	Inst Room Purge Air Exh	4*
20. FCV-30-59	Inst Room Purge Air Exh	4*
21. FCV-90-107	Cntmt Bldg LWR Compt Air Mon	5*
22. FCV-90-108	Cntmt Bldg LWR Compt Air Mon	5*
23. FCV-90-109	Cntmt Bldg LWR Compt Air Mon	5*
24. FCV-90-110	Cntmt Bldg LWR Compt Air Mon	5*
25. FCV-90-111	Cntmt Bldg LWR Compt Air Mon	5*
26. FCV-90-113	Cntmt Bldg UPR Compt Air Mon	5*
27. FCV-90-114	Cntmt Bldg UPR Compt Air Mon	5*
28. FCV-90-115	Cntmt Bldg UPR Compt Air Mon	5*
29. FCV-90-116	Cntmt Bldg UPR Compt Air Mon	5*
30. FCV-90-117	Cntmt Bldg UPR Compt Air Mon	5*
D. OTHER		
1. FCV-30-46	Vacuum Relief Isolation Valve	2
2. FCV-30-47	Vacuum Relief Isolation Valve	2
3. FCV-30-48	Vacuum Relief Isolation Valve	2
4. FCV-62-90	Normal Charging Isolation Valve	1

*Provisions of LCO 3.0.4 are not applicable if valve is secured in its isolated position with power removed and leakage limits of Specification 4.6.1.1.c are satisfied. For purge valves, leakage limits under surveillance Requirement 4.6.1.9.3 must also be satisfied.

#Provisions of LCO 3.0.4 are not applicable if valve is secured in its isolated position with power removed and either FCV-62-73 or FCV-62-74 is maintained operable.

**This valve is required after completion of the associated modification.

BASES

3/4.6.1 PRIMARY CONTAINMENT

The safety design basis for primary containment is that the containment must withstand the pressures and temperatures of the limiting design basis accident (DBA) without exceeding the design leakage rates.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA), a steam line break, and a rod ejection accident (REA). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. This leakage rate limitation will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions. The containment was designed with an allowable leakage rate of 0.25 percent of containment air weight per day. This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting DBA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.25 percent per day in the safety analysis at P_a = 12.0 psig. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between tests.

Primary containment INTEGRITY or operability is maintained by limiting leakage to within the acceptance criteria of 10 CFR 50, Appendix J.

Individual leakage rates specified for the containment air lock (LCO 3.6.1.3), purge valves (LCO 3.6.1.9) and secondary bypass leakage (LCO 3.6.1.2) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits do not result in the primary containment being inoperable unless the leakage, when combined with other Type B and C test leakages, exceeds the acceptance criteria of Appendix J.

3/4.6.1.2 SECONDARY CONTAINMENT BYPASS LEAKAGE

The safety design basis for containment leakage assumes that 75 percent of the leakage from the primary containment enters the shield building annulus for filtration by the emergency gas treatment system. The remaining 25 percent of the primary containment leakage, which is considered to be bypassed to the auxiliary building, is assumed to exhaust directly to the atmosphere without filtration during the first 5 minutes of the accident. After 5 minutes, any bypass leakage to the auxiliary building is filtered by the auxiliary building gas treatment system. A tabulation of potential secondary containment bypass leakage paths to the auxiliary building is provided in Table 3.6-1. Restricting the leakage through the bypass leakage paths in Table 3.6-1 to 0.25 L_a provides assurance that the leakage fraction assumptions used in the evaluation of site boundary radiation doses remain valid.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provide assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the annulus atmosphere of 0.5 psig and 2) the containment peak pressure does not exceed the maximum allowable internal pressure of 12 psig during LOCA conditions.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that 1) the containment air mass is limited to an initial mass sufficiently low to prevent exceeding the maximum allowable internal pressure during LOCA conditions and 2) the ambient air temperature does not exceed that temperature allowable for the continuous duty rating specified for equipment and instrumentation located within containment.

The containment pressure transient is sensitive to the initially contained air mass during a LOCA. The contained air mass increases with decreasing temperature. The lower temperature limits of 100°F for the lower compartment, 85°F for the upper compartment, and 60°F when less than or equal to 5% of RATED THERMAL POWER will limit the peak pressure to an acceptable value. The upper temperature limit influences the peak accident temperature slightly during a LOCA; however, this limit is based primarily upon equipment protection and anticipated operating conditions. Both the upper and lower temperature limits are consistent with the parameters used in the accident analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 12 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.7 SHIELD BUILDING STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment shield building will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to provide 1) protection for the steel vessel from external missiles, 2) radiation shielding in the event of a LOCA, and 3) and annulus surrounding the steel vessel that can be maintained at a negative pressure during accident conditions.