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

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 103
License No. NPF-30

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

   A. The application for amendment filed by Union Electric Company (UE, the licensee) dated May 20, 1994, as supplemented March 29, 1995, 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. NPF-30 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 103 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into the license. UE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance. The Technical Specifications are to be implemented within 120 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

L. Raynard Wharton

L. Raynard Wharton, Project Manager
Project Directorate IV-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of issuance: October 20, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 103

OPERATING LICENSE NO. NPF-30

DOCKET NO. 50-483

Revise 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 vertical lines indicating the area of change.

REMOVE	INSERT
IV	IV
VI	VI
VII	VII
VIII	VIII
IX	IX
X	X
XI	XI
XII	XII
XIII	XIII
XIV	XIV
XV	XV
XVI	XVI
XVII	XVII
XX	XX
1-2	1-2
3/4 1-1	3/4 1-1
3/4 1-2	3/4 1-2
3/4 1-3	3/4 1-3
3/4 1-7	3/4 1-7
3/4 1-8	3/4 1-8 through 3/4 1-13
3/4 1-9	--
3/4 1-10	--
3/4 1-11	--
3/4 1-12	--
3/4 1-13	--
3/4 1-14	3/4 1-14
3/4 1-15	3/4 1-15
3/4 1-18	3/4 1-18 through 3/4 1-19
3/4 1-19	--
3/4 1-21	3/4 1-21
3/4 3-42	3/4 3-42 through 3/4 3-48
3/4 3-43	--
3/4 3-44	--
3/4 3-45	--
3/4 3-46	--
3/4 3-47	--
3/4 3-48	--

REMOVE	INSERT
3/4 3-52	3/4 3-52
3/4 3-53	3/4 3-53
3/4 3-54	3/4 3-54
3/4 3-55	3/4 3-55
3/4 3-56	3/4 3-56
3/4 3-57 through 3/4 3-61	3/4 3-57 through 3/4 3-76
3/4 3-62	--
3/4 3-63 through 3/4 3-74	--
3/4 3-75	--
3/4 3-76	--
3/4 4-7	3/4 4-7
3/4 4-22	3/4 4-22 through 3/4 4-24
3/4 4-23	--
3/4 4-24	--
3/4 4-32	3/4 4-32 through 3/4 4-33
3/4 4-33	--
3/4 4-37	3/4 4-37 through 3/4 4-38
3/4 4-38	--
3/4 5-9	3/4 5-9
3/4 6-1	3/4 6-1
3/4 6-2	3/4 6-2 through 3/4 6-3
3/4 6-2a	--
3/4 6-3	--
3/4 6-8	3/4 6-8 through 3/4 6-10a
3/4 6-9	--
3/4 6-10	--
3/4 6-10a	--
3/4 6-12	3/4 6-12
3/4 6-31	3/4 6-31
3/4 7-10	3/4 7-10
3/4 7-19	3/4 7-19 through 3/4 7-38
3/4 7-20	--
3/4 7-21	--
3/4 7-22	--
3/4 7-23	--
3/4 7-23a	--
3/4 7-23b	--
3/4 7-24	--
3/4 7-25	--
3/4 7-26	--
3/4 7-27 through 3/4 7-36	--
3/4 7-37	--
3/4 7-38	--
3/4 8-16	3/4 8-16 through 3/4 8-17
3/4 8-17	--
3/4 9-5	3/4 9-5 through 3/4 9-8
3/4 9-6	--
3/4 9-7	--
3/4 9-8	--

REMOVE	INSERT
3/4 9-13	3/4 9-13
3/4 10-1	3/4 10-1
3/4 10-5	3/4 10-5
3/4 11-1	3/4 11-1 through 3/4 11-18
3/4 11-2	--
3/4 11-3	--
3/4 11-4 through 3/4 11-18	--
B 3/4 1-2	B 3/4 1-2
B 3/4 1-3	B 3/4 1-3
B 3/4 1-4	B 3/4 1-4
--	B 3/4 1-5
B 3/4 3-3	B 3/4 3-3
B 3/4 3-4	B 3/4 3-4
B 3/4 3-5	B 3/4 3-5
B 3/4 4-2	B 3/4 4-2
B 3/4 4-5	B 3/4 4-5
B 3/4 4-7	B 3/4 4-7
B 3/4 4-15	B 3/4 4-15
B 3/4 4-17	B 3/4 4-17
B 3/4 6-1	B 3/4 6-1
B 3/4 6-1a	B 3/4 6-1a
B 3/4 6-2	B 3/4 6-2
B 3/4 6-3	B 3/4 6-3
B 3/4 6-4	B 3/4 6-4
B 3/4 7-3	B 3/4 7-3
B 3/4 7-3a	B 3/4 7-3a
B 3/4 7-5	B 3/4 7-5 through B 3/4 7-8
B 3/4 7-6	--
B 3/4 7-7	--
B 3/4 7-8	--
B 3/4 8-3	B 3/4 8-3
B 3/4 9-1	B 3/4 9-1
B 3/4 9-2	B 3/4 9-2
B 3/4 10-1	B 3/4 10-1
B 3/4 11-1	B 3/4 11-1
6-15	6-15
6-17	6-17
6-18	6-18
6-19	6-19
--	6-19a
--	6-19b
6-23	6-23

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

BASES

BASES

BASES

ADMINISTRATIVE CONTROLS

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 3.6-1 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 sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

e. The containment leakage rates are determined per Specification 4.6.1.1.d and are within the limits listed in the Bases of Specification 3.6.1.1.

f. Structural integrity is assured via the program described in Specification 6.8.5.c.

CONTROLLED LEAKAGE

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

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor 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 LIMITS 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. The cycle specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.9. Plant operation within these operating limits is addressed in individual specifications.

DOSE EQUIVALENT I-131

1.11 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} > 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.3% Δk/k.

APPLICABILITY: MODES 3 and 4.

ACTION:

With the SHUTDOWN MARGIN less than 1.3% Δk/k, within 15 minutes initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.3% Δk/k:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);

b. At least once per 24 hours by consideration of the following factors:

    1) Reactor Coolant System boron concentration,

    2) Control rod position,

    3) Reactor Coolant System average temperature,

    4) Fuel burnup based on gross thermal energy generation,

    5) Xenon concentration, and

    6) Samarium concentration.

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SHUTDOWN MARGIN - $T_{avg} \leq 200°F$

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1% $\Delta k/k$.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1% $\Delta k/k$, within 15 minutes initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1% $\Delta k/k$:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);

b. At least once per 24 hours by consideration of the following factors:

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

CORE REACTIVITY

LIMITING CONDITION FOR OPERATION

3.1.1.5 The measured core reactivity shall be within ±1% Δk/k of predicted values.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the measured core reactivity not within limits, within 72 hours:

a. reevaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation, and

b. establish appropriate administrative operating restrictions and Surveillance Requirements, or

c. be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.5.1 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ±1% Δk/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.b. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

4.1.1.5.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.3% Δk/k prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1.b, with the control banks at the maximum insertion limit of Specification 3.1.3.6.

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3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full-length shutdown and control rods shall be OPERABLE and positioned within ±12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*.

ACTION:

The ACTION to be taken is based on the cause of inoperability of control rods as follows:

Any immovability of a control rod initially invokes ACTION Statement 3.1.3.1.a. Subsequently, ACTION Statement 3.1.3.1.a may be exited and ACTION Statement 3.1.3.1.d invoked if either the rod control urgent failure alarm is illuminated or an electrical problem is detected in the rod control system.

CAUSE OF INOPERABILITY	ACTION One Rod	ACTION More Than One Rod
1. Immovable as a result of excessive friction or mechanical interference or known to be untrippable.	(a)	(a)
2. Misaligned by more than ±12 steps (indicated position) from its group step counter demand height or from any other rod in its group.	(c)	(b)
3. Inoperable due to a rod control urgent failure alarm or other electrical problem in the rod control system, but trippable.	(d)	(d)

ACTION a - 1. Determine that the SHUTDOWN MARGIN is greater than or equal to 1.3% Δk/k, with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s), within 1 hour, and

2. Be in HOT STANDBY within 6 hours.

ACTION b - Be in HOT STANDBY within 6 hours.

ACTION c - POWER OPERATION may continue provided that within 1 hour:

1. The rod is restored to OPERABLE status within the above alignment requirements, or

* See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ±12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Specification 3.1.3.6. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

3. The rod is declared inoperable and the SHUTDOWN MARGIN is greater than or equal to 1.3% Δk/k. POWER OPERATION may then continue provided that:

   a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions;

   b) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F_{\Delta H}^N$ are verified to be within their limits within 72 hours; and

   c) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the High Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER.

ACTION d - Restore the inoperable rods to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full-length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the rod position deviation monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full-length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

4.1.3.1.3 Prior to reactor criticality, the rod drop time of the individual full-length shutdown and control rods from the fully withdrawn position shall be demonstrated to be less than or equal to 2.7 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry with $T_{avg} \geq 551°F$ and all reactor coolant pumps operating:

a. For all rods following each removal of the reactor vessel head, and

b. For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System which could affect the drop time of those specific rods.

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CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as specified in the Core Operating Limits Report (COLR).

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the control banks inserted beyond the insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2:

a. Within 1 hour, verify that the SHUTDOWN MARGIN is greater than or equal to 1.3% $\Delta k/k$ or initiate boration until the SHUTDOWN MARGIN is restored to greater than or equal to 1.3% $\Delta k/k$, and

b. Restore the control banks to within the limits within 2 hours, or

c. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the insertion limits specified in the COLR, or

d. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6.1 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

4.1.3.6.2 When in MODE 2 with K_{eff} less than 1, verify that the predicted critical control rod position is within insertion limits within 4 hours prior to achieving reactor criticality.

* See Special Test Exceptions Specifications 3.10.2 and 3.10.3.
# With K_{eff} greater than or equal to 1.

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ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 30 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the following 14 days outlining the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the channels to OPERABLE status.

b. With the number of OPERABLE accident monitoring instrumentation channels, except for instrument functions 10, 16 and 18 (Containment Hydrogen Concentration Level, Containment Radiation Level, and the Reactor Vessel Level Indicating System), less than the Minimum Channels OPERABLE requirements of Table 3.3-10, restore one channel to OPERABLE status within 7 days; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With the number of OPERABLE channels for instrument functions 16 or 18 (Containment Radiation Level or the Reactor Vessel Level Indicating System) less than the Minimum Channels OPERABLE requirements of Table 3.3-10, initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours and either restore one inoperable channel to OPERABLE status within 7 days, or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the following 14 days outlining the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the channels to OPERABLE status.

d. With the number of OPERABLE channels for the Containment Hydrogen Concentration Level monitors less than the Minimum Channels OPERABLE requirement of Table 3.3-10, restore one channel to OPERABLE status within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

e. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 4.3-7.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure - Normal Range	2	1
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	2	1
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	2	1
4. Reactor Coolant Pressure - Wide Range	2	1
5. Pressurizer Water Level	2	1
6. Steam Line Pressure	2/steam generator	1/steam generator
7. Steam Generator Water Level - Narrow Range	2/steam generator	1/steam generator
8. Steam Generator Water Level - Wide Range	1/steam generator	1/steam generator
9. Refueling Water Storage Tank Water Level	2	1
10. Containment Hydrogen Concentration Level	2	1
11. Auxiliary Feedwater Flow Rate	1/steam generator	1/steam generator
12. Deleted		
13. Deleted		
14. Neutron Flux	2	1
15. Containment Normal Sump Water Level	2	1
16. Containment Radiation Level (High Range, GT-RIC-59, -60)	2	1
17. Thermocouple/Core Cooling Detection System	4/core quadrant	2/core quadrant
18. Reactor Vessel Level Indicating System	2	1

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TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure - Normal Range	M	R
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	M	R
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	M	R
4. Reactor Coolant Pressure - Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Line Pressure	M	R
7. Steam Generator Water Level - Narrow Range	M	R
8. Steam Generator Water Level - Wide Range	M	R
9. Refueling Water Storage Tank Water Level	M	R
10. Containment Hydrogen Concentration Level	M	R
11. Auxiliary Feedwater Flow Rate	M	R
12. Deleted		
13. Deleted		
14. Neutron Flux	M	R (1)
15. Containment Normal Sump Level	M	R
16. Containment Radiation Level (High Range, GT-RIC-59, -60)	M	R (2)
17. Thermocouple/Core Cooling Detection System	M	R
18. Reactor Vessel Level Indicating System	M	R

TABLE 4.3-7 (Continued)

TABLE NOTATIONS

(1) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(2) CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10R/h and a one point calibration check of the detector below 10R/h with an installed or portable gamma source.

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3/4.5.4 ECCS SUBSYSTEMS - $T_{avg} \leq 200°F$

LIMITING CONDITION FOR OPERATION

3.5.4 All Safety Injection pumps and one centrifugal charging pump shall be inoperable.

APPLICABILITY: MODE 5 and MODE 6 with the reactor vessel head on.*

ACTION:

a. With a Safety Injection pump OPERABLE, restore all Safety Injection pumps to an inoperable status within 4 hours.

b. With two centrifugal charging pumps OPERABLE, restore one of the centrifugal charging pumps to an inoperable status within 4 hours.

SURVEILLANCE REQUIREMENTS

4.5.4.1 All Safety Injection pumps shall be demonstrated inoperable** by verifying that the motor circuit breakers are secured in the open position at least once per 31 days.

4.5.4.2 One centrifugal charging pump shall be demonstrated inoperable** by verifying that the motor circuit breakers are secured in the open position at least once per 31 days.

* When the RCS water level is below the top of the reactor vessel flange, both Safety Injection pumps may be OPERABLE for the purpose of protecting the decay heat removal function.

** An inoperable pump may be energized for testing or for filling accumulators provided the discharge at the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

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 1 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 manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 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. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at a pressure not less than P_a, 48.1 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.1.d for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a;

d. By performing containment leakage rate testing, except for containment air locks, in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions; and

e. By verifying containment structural integrity in accordance with the Containment Tendon Surveillance Program of Specification 6.8.5.c.

* Except valves, blind flanges, and deactivated automatic valves which are located inside the 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.

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SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each 36-inch containment shutdown purge supply and exhaust isolation valve(s)* shall be verified blank flanged and closed at least once per 31 days.

4.6.1.7.2 Each 36-inch containment shutdown purge supply and exhaust isolation valve and its associated blank flange shall be leak tested at least once per 24 months and following each reinstallation of the blank flange when pressurized to P_a, 48.1 psig, and verifying that when the measured leakage rate for these valves and flanges, including stem leakage, is added to the leakage rates determined pursuant to Specification 4.6.1.1.d for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a.

4.6.1.7.3 The cumulative time that all 18-inch containment mini-purge supply and exhaust isolation valves have been open during a calendar year shall be determined at least once per 7 days.

4.6.1.7.4 At least once per 3 months each 18-inch containment mini-purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than 0.05 L_a when pressurized to P_a.

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* Except valves and flanges which are located inside containment. These valves shall be verified to be closed with their blank flanges installed prior to entry into MODE 4 following each COLD SHUTDOWN.

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MODERATOR TEMPERATURE COEFFICIENT (Continued)

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting End of Life (EOL) MTC value. The 300 ppm surveillance limit MTC value represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting EOL MTC value.

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 551°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.1.5 CORE REACTIVITY

When measured core reactivity is within ±1% Δk/k of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the limit are normally detected by comparing predicted and measured steady state RCS critical boron concentrations, the difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the RCS boron concentration are unlikely.

The acceptance criteria for core reactivity (±1% Δk/k of the predicted value) ensures plant operation is maintained within the assumptions of the safety analyses.

CORE REACTIVITY (Continued)

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation is, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod ejection accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance additionally ensures that the nuclear methods provide an accurate representation of the core reactivity.

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the RCS boron concentration requirements for reactivity control during fuel depletion.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at beginning of cycle (BOC) do not agree, then the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted boron letdown curve that develop during fuel depletion may be an indication that the calculational model is not adequate for core burnups beyond BOC, or that an unexpected change in core conditions has occurred.

The normalization of predicted RCS boron concentration to the measured value shall be performed after reaching RTP following startup from a refueling outage, with the control rods in their normal positions for power operation. The normalization is performed at BOC conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle.

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required completion time of 72 hours is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

CORE REACTIVITY (Continued)

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core conditions at the time of RCS boron concentration sampling, then a recalculation of the RCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the boron letdown curve may be renormalized and power operation may continue. If operational restrictions or additional surveillance requirements are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required completion time of 72 hours is adequate for preparing whatever operating restrictions or surveillances that may be required to allow continued reactor operation.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. Verification that the Digital Rod Position Indicator agrees with the demanded position within ±12 steps at 24, 48, 120 and 228 steps withdrawn for the Control Banks and 18, 210 and 228 steps withdrawn for the Shutdown Banks provides assurance that the Digital Rod Position Indicator is operating correctly over the full range of indication. Since the Digital Rod Position System does not indicate the actual shutdown rod position between 18 steps and 210 steps, only points in the indicated ranges are picked for verification of agreement with demanded position. Shutdown and control rods are positioned at 225 steps or higher for fully withdrawn.

For purposes of determining compliance with Specification 3.1.3.1, any immovability of a control rod initially invokes ACTION statement 3.1.3.1.a. Subsequently, ACTION statement 3.1.3.1.a may be exited and ACTION statement 3.1.3.1.d invoked if either the rod control urgent failure alarm is illuminated or an electrical problem is detected in the rod control system. The rod is considered trippable if the rod was demonstrated OPERABLE during the last performance of Surveillance Requirement 4.1.3.1.2 and met the rod drop time criteria during the last performance of Surveillance Requirement 4.1.3.1.3.

MOVABLE CONTROL ASSEMBLIES (Continued)

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The power reduction and shutdown time limits given in ACTION statements 3.1.3.2.a.2, 3.1.3.2.b.2, and 3.1.3.2.c.2, respectively, are initiated at the time of discovery that the compensatory actions required for POWER OPERATION can no longer be met.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 551°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a Reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

Engineered Safety Features Actuation System Interlocks

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip, closes main feedwater valves on T_{avg} below setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level signal, allows Safety Injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 On increasing pressure P-11 automatically reinstates Safety Injection actuation on low pressurizer pressure and low steam line pressure and automatically blocks steam line isolation on negative steam line pressure rate. On decreasing pressure, P-11 allows the manual block of Safety Injection on low pressurizer pressure and low steam line pressure and allows steam line isolation on negative steam line pressure rate to become active upon manual block of low steam line pressure SI.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated action will be initiated when the radiation level monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance. The radiation monitors for plant operations senses radiation levels in selected plant systems and locations and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms or automatic isolation action and actuation of Emergency Exhaust or Control Room Emergency Ventilation Systems.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the Remote Shutdown System ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the facility from locations outside of the control room and that a fire will not preclude achieving safe shutdown. The Remote Shutdown System transfer switches, power circuits, and control circuits are independent of areas where a fire could damage systems normally used to shutdown the reactor. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 3 and 19 and Appendix R of 10 CFR Part 50.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1980, and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

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3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam.

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss-of-load assuming no Reactor trip and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.3 PRESSURIZER

The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure and prevent a high pressurizer pressure reactor trip during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer Code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

The PORVs are equipped with automatic actuation circuitry and manual control capability. Because no credit for automatic operation is taken in the FSAR analyses for MODE 1, 2 and 3 transients where operation of the PORVs has a beneficial impact on the results of the analysis, the PORVs are considered OPERABLE in either the manual or automatic mode. The automatic mode is the preferred configuration, as this provides pressure relieving capability without reliance on operation action.

OPERATIONAL LEAKAGE (Continued)

The leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which would result in a LOCA that bypasses containment, these valves should be tested periodically to ensure low probability of gross failure.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady state reactor-to-secondary steam generator leakage rate of 1 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Callaway site, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

PRESSURE/TEMPERATURE LIMITS (Continued)

2. These limit lines shall be calculated periodically using methods provided below.

3. System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the 1972 Winter Addenda to Section III of the ASME Boiler and Pressure Vessel Code.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 17 effective full power years (EFPY) of service life. The 17 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and copper content and phosphorus content of the material in question, can be predicted using Figure B 3/4.4-1 and the largest value of ΔRT_{NDT} computed by either Regulatory Guide 1.99, Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," or the Westinghouse Copper Trend Curves shown in Figure B 3/4.4-2. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 17 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10 CFR Part 50, Appendix H. The lead factor represents the

HEATUP (Continued)

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

The OPERABILITY of two PORVs, two RHR suction relief valves, one RHR suction relief valve and one PORV, or an RCS vent opening of at least 2 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 368°F. Either PORV or either RHR suction relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the start of a centrifugal charging pump and its injection into a water-solid RCS.

In addition to opening RCS vents to meet the requirement of Specification 3.4.9.3c., it is acceptable to remove a pressurizer Code safety valve, open a PORV block valve and remove power from the valve operator in conjunction with disassembly of a PORV and removal of its internals, or otherwise open the RCS.

COLD OVERPRESSURE

The Maximum Allowed PORV Setpoint for the Cold Overpressure Mitigation System (COMS) is derived by analysis which models the performance of the COMS assuming various mass input and heat input transients. Operation with a PORV setpoint less than or equal to the maximum setpoint ensures that Appendix G criteria will not be violated with consideration for 1) a maximum pressure overshoot beyond the PORV setpoint which can occur as a result of time delays in signal processing and valve opening; 2) a 50°F heat transport effect made

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3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions.

Containment leakage rates shall be within the following limits:

1) An overall integrated leakage rate of less than or equal to L_a, 0.20% by weight of the containment air per 24 hours at P_a, 48.1 psig.

2) A combined leakage rate of less than 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to P_a, 48.1 psig.

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 provides 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 outside atmosphere of 3.0 psig, and (2) the containment peak pressure does not exceed the design pressure of 60 psig during steam line break conditions.

The maximum peak pressure expected to be obtained from a steam line break event is 48 psig. The limit of 1.5 psig for initial positive containment pressure will limit the total pressure to 49.5 psig, which is less than design pressure and is consistent with the safety analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the safety analysis for a steam line break accident. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 36-inch containment purge supply and exhaust isolation valves are required to be closed and blank flanged during plant operations since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves closed and blank flanged during plant operation ensures that excessive quantities of radioactive material will not be released via the Containment Purge System. To provide assurance that the 36-inch containment purge valves cannot be inadvertently opened, the valves are blank flanged.

The use of the containment mini-purge lines is restricted to the 18-inch purge supply and exhaust isolation valves since, unlike the 36-inch valves, the 18-inch valves are capable of closing during a LOCA or steam line break accident. Therefore, the SITE BOUNDARY dose guideline values of 10 CFR Part 100 would not be exceeded in the event of an accident during containment purging operation. Operation will be limited to 2000 hours during a calendar year. The total time the Containment Purge (vent) System isolation valves may be open during MODES 1, 2, 3, and 4 in a calendar year is a function of anticipated need and operating experience. Only safety-related reasons; e.g., containment pres-sure control or the reduction of airborne radioactivity to facilitate personnel access for surveillance and maintenance activities, should be used to support additional time requests. Only safety-related reasons should be used to justify the opening of these isolation valves during MODES 1, 2, 3, and 4 in any calendar year regardless of the allowable hours.

Leakage integrity tests with a maximum allowable leakage rate for containment purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before gross leakage failures could develop. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a LOCA or steam line break. The pressure reduction and resultant lower containment leakage rate are co·istent with the assumptions used in the safety analyses.

The Containmen· ray System and the Containment Cooling System are redundant to each other in p· ·ing post-accident cooling of the Containment atmosphere. However, the Conta· t Spray System also provides a mechanism for removing iodine from the c· ·ent atmosphere and therefore the time requirements for restoring an inop· spray system to OPERABLE status have been maintained consistent with t· ·ssigned other inoperable ESF equipment.

3/4.6.2.2 RECIRCULATION FLUID pH CONTROL (RFPC) SYSTEM

The operability of the RFPC System ensures that there exists adequate TSP-C in the containment such that a post-LOCA equilibrium sump pH of greater than or equal to 7.1 is maintained during the recirculation phase. The minimum depth of 30" ensures that 9000 lbm of TSP-C is available for dissolution to yield a minimum equilibrium sump pH of 7.1 This pH level minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The upper limit of 36.8" corresponds to the basket design capacity.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the Containment Cooling System ensures that: (1) the containment air temperature will be maintained within limits during normal operation, and (2) adequate heat removal capacity is available when operated in conjunction with the Containment Spray Systems during post-LOCA conditions.

The Containment Cooling System and the Containment Spray System are redundant to each other in providing post-accident cooling of the Containment atmosphere. As a result of this redundancy in cooling capability, the allowable out-of-service time requirements for the Containment Cooling System have been appropriately adjusted. However, the allowable out-of-service time requirements for the Containment Spray System have been maintained consistent with that assigned other inoperable ESF equipment since the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 thru 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit (or the Purge System) is capable of controlling the expected hydrogen generation associated with: (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. The Hydrogen Purge Subsystem discharges directly to the Emergency Exhaust System. Operation of the Emergency Exhaust System with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," Revision 2, November 1978.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to: (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam line isolation valves within the closure times of the Surveillance Requirements is consistent with the assumptions used in the safety analyses.

3/4.7.1.6 MAIN FEEDWATER ISOLATION VALVES

The OPERABILITY of the main feedwater isolation valves: (1) provides a pressure boundary to permit auxiliary feedwater addition in the event of a main steam or feedwater line break; (2) limits the RCS cooldown and the mass and energy releases for secondary line breaks inside containment; and (3) mitigates steam generator overfill events such as a feedwater malfunction, with protection provided by feedwater isolation via the steam generator high-high level trip signal. The OPERABILITY of the main feedwater isolation valves within the closure times of the Surveillance Requirements is consistent with the assumptions used in the safety analyses.

3/4.7.1.7 STEAM GENERATOR ATMOSPHERIC STEAM DUMP VALVES

The OPERABILITY of the steam generator atmospheric steam dump valves (ASD's) ensures that the reactor decay heat can be dissipated to the atmosphere in the event of a steam generator tube rupture and loss of offsite power and that the Reactor Coolant System can be cooled down for Residual Heat Removal System operation. The number of required ASD's assures that the subcooling can be achieved, consistent with the assumptions used in the steam generator tube rupture analysis, to facilitate equalizing pressures between the Reactor Coolant System and the faulted steam generator. For cooling the plant to RHR initiation conditions, only one ASD is required. In this case, with three ASD's OPERABLE, if the single failure of one ASD occurs and another ASD is assumed to be associated with the faulted steam generator, one ASD remains available for required heat removal.

Each ASD is equipped with a manual block valve (in the auxiliary building) to provide a positive shutoff capability should an ASD develop leakage. Closure of the block valves of all ASD's because of excessive seat leakage does not endanger the reactor core; consistent with plant

3/4.7.1.7 STEAM GENERATOR ATMOSPHERIC STEAM DUMP VALVES (Continued)

accident and transient analyses, decay heat can be dissipated with the main steamline safety valves or a block valve can be opened manually in the auxiliary building and the ASD can be used to control release of steam to the atmosphere. For the steam generator tube rupture event, primary to secondary leakage can be terminated by depressurizing the Reactor Coolant System with the pressurizer power operated relief valves.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the Component Cooling Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses. Each independent CCW loop contains two 100% capacity pumps and, therefore, the failure of one pump does not affect the OPERABILITY of that loop.

3/4.7.4 ESSENTIAL SERVICE WATER SYSTEM

The OPERABILITY of the Essential Service Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.5 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level and temperature ensure that sufficient cooling capacity is available either to: (1) provide normal cooldown of the facility, or (2) mitigate the effects of accident conditions within acceptable limits.

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3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. The limitation on K_{eff} of no greater than 0.95 is sufficient to prevent reactor criticality during refueling operations. The locking closed of the required valves during refueling operations precludes the possibility of uncontrolled boron dilution of the filled portions of the Reactor Coolant System via the CVCS blending tee. This action prevents flow to the RCS of unborated water by closing all automatic flow paths from sources of unborated water. Administrative controls will limit the volume of unborated water which can be added to the refueling pool for decontamination activities in order to prevent diluting the refueling pool below the limits specified in the LCO. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Source Range Neutron Flux Monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the fuel handling accident radiological consequence and spent fuel pool thermal-hydraulic analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

The OPERABILITY of this system ensures the containment purge penetrations will be automatically isolated upon detection of high radiation levels within containment. The OPERABILITY of this system is required to restrict the release of radioactive materials from the containment atmosphere to the environment.

The restriction on the setpoint for GT-RE-22 and GT-RE-33 is based on a fuel handling accident inside the Containment Building with resulting damage to one fuel rod and subsequent release of 0.1% of the noble gas gap activity, except for 0.3% of the Kr-85 gap activity. The setpoint concentration of 5E-3 μCi/cc is equivalent to approximately 150 mR/hr submersion dose rate.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that: (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to maintain a 1000 gpm flowrate ensures that there is adequate flow to prevent boron stratification. The RHR flow to the RCS will provide adequate cooling to prevent exceeding 140°F and to allow flowrates which provide additional margin against vortexing at the RHR pump suction while in partial drain operation.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual control rods to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to: (1) measure control rod worth, and (2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 PHYSICS TESTS

This special test exception permits PHYSICS TESTS to be performed at less than or equal to 5% of RATED THERMAL POWER with the RCS T_{avg} slightly lower than normally allowed so that the fundamental nuclear characteristics of the core and related instrumentation can be verified. In order for various characteristics to be accurately measured, it is at times necessary to operate outside the normal restrictions of these Technical Specifications. For instance, to measure the moderator temperature coefficient at BOL, it is necessary to position the various control rods at heights which may not normally be allowed by Specification 3.1.3.6 which in turn may cause the RCS T_{avg} to fall slightly below the minimum temperature of Specification 3.1.1.4.

3/4.10.4 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

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SAFETY LIMIT VIOLATION (Continued)

c. The Safety Limit Violation Report shall be submitted to the Commission, the NSRB and the Senior Vice President-Nuclear within 14 days of the violation; and

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Section 7.1 of Generic Letter No. 82-33;

c. Plant Security Plan implementation;

d. Radiological Emergency Response Plan implementation;

e. PROCESS CONTROL PROGRAM implementation;

f. OFFSITE DOSE CALCULATION MANUAL implementation;

g. Quality Assurance Program implementation for effluent and environmental monitoring; and

h. Fire Protection Program implementation.

6.8.2 Each procedure and administrative policy of Specification 6.8.1 above, and changes thereto, including temporary changes shall be reviewed prior to implementation as set forth in Specification 6.5 above.

6.8.3 The plant Administrative Procedures and changes thereto shall be reviewed in accordance with Specification 6.5.1.6 and approved in accordance with Specification 6.5.3.1. The associated implementing procedures and changes thereto shall be reviewed and approved in accordance with Specification 6.5.3.1.

6.8.4 The following programs shall be established, implemented and maintained:

a. Reactor Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the recirculation portion of the Containment Spray System, Safety Injection System, Chemical and Volume Control System, and RHR System. The program shall include the following:

1) Preventive maintenance and periodic visual inspection requirements, and

PROCEDURES AND PROGRAMS (Continued)

e. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 CFR Part 20, Appendix B, Table II, Column 2,

3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 and with the methodology and parameters in the ODCM,

4) Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,

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

PROCEDURES AND PROGRAMS (Continued)

e. Radioactive Effluent Controls Program (Continued)

6) 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,

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to the doses associated with 10 CFR Part 50, Appendix B, Table II, Column 1,

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

9) 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 to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

10) 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.

f. 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:

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

2) A Land Use Census to ensure that changes in the use of areas 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

PROCEDURES AND PROGRAMS (Continued)

f. Radiological Environmental Monitoring Program (Continued)

3) Participation in a 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.

6.8.5 The following programs, relocated from the Technical Specifications to FSAR Chapter 16, shall be implemented and maintained:

a. Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the WASTE GAS HOLDUP SYSTEM, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The program shall include:

1. The limits for concentrations of hydrogen and oxygen in the WASTE GAS HOLDUP SYSTEM and a surveillance program to ensure the limits are maintained.

2. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of $\geq$ 0.5 rem to a MEMBER OF THE PUBLIC at the nearest SITE BOUNDARY in the event of an uncontrolled release of the tanks' contents, consistent with Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases due to Waste Gas System Leak or Failure," in NUREG-0800, July 1981.

3. A surveillance program to ensure that the quantity of radioactivity contained in the following outdoor liquid radwaste tanks, that are not surrounded by liners, dikes, or walls capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste system, is less than the amount that would result in concentrations less than the limits of 10 CFR Part 20.1 -20.602, Appendix B (redesignated at 56FR23391, May 21, 1991) at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA, in the event of an uncontrolled release of the tanks' contents:

   a. Reactor Makeup Water Storage Tank,
   b. Refueling Water Storage Tank,
   c. Condensate Storage Tank, and
   d. Outside temporary tanks, excluding demineralizer vessels and the liner being used to solidify radioactive waste.

PROCEDURES AND PROGRAMS (Continued)

a. Explosive Gas and Storage Tank Radioactivity Monitoring Program (Cont'd)

The provisions of Specifications 4.0.2 and 4.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

b. Reactor Coolant Pump Flywheel Inspection Program

Each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, dated August 1975.

c. Containment Tendon Surveillance Program

This program provides controls for monitoring tendon performance, including the effectiveness of the tendon corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial plant operation as well as periodic testing thereafter. The Containment Tendon Surveillance Program, and its inspection frequencies and acceptance criteria, shall be in accordance with the Callaway position on proposed Revision 3 of Regulatory Guide 1.35 dated April 1979.

The provisions of Specifications 4.0.2 and 4.0.3 are applicable to the Containment Tendon Surveillance Program inspection frequencies.

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 reports shall be submitted to the Regional Administrator of the NRC Regional Office unless otherwise noted.

STARTUP REPORT

6.9.1.1 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.

6.9.1.2 The Startup 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.

REPORTING REQUIREMENTS (Continued)

6.9.1.3 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 operation) supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS

6.9.1.4 Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

RECORD RETENTION (Continued)

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;

k. Records of meetings of the ORC and the NSRB;

l. Records of the service lives of all hydraulic and mechanical snubbers including the date at which the service life commences and associated installation and maintenance records;

m. Records of secondary water sampling and water quality;

n. Records of analysis required by the Radiological Environmental Monitoring Program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed; and

o. Records of reviews performed for changes made to APA-ZZ-01003, the OFFSITE DOSE CALCULATION MANUAL and APA-ZZ-01011, the PROCESS CONTROL PROGRAM.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 Pursuant to Paragraph 20.203(c)(5) of 10 CFR Part 20, in lieu of the "control device" or "alarm signal" required by Paragraph 20.203(c)(2) each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is equal to or less than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Health Physics Technician) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mR/h, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area, or