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

MAR 1 1985

Docket No. 50-482

Mr. Glenn L. Koester
Vice President - Nuclear
Kansas Gas and Electric Company
201 North Market Street
Post Office Box 208
Wichita, Kansas 67201

Dear Mr. Koester:

Subject: Revised Draft License for Wolf Creek

By letter dated December 10, 1984, you submitted a final draft copy of proposed Technical Specifications for the Wolf Creek Generating Station. Your letter also submitted proposed changes to the Technical Specifications and certified that the proposed Technical Specifications accurately reflected the plant design, FSAR and SER. By letters dated January 2, 1985, January 18, 1985, January 25, 1985, February 1, 1985, February 19, 1985 and February 25, 1985, you proposed additional changes to the Wolf Creek Technical Specifications.

We have incorporated many, but not all, of the changes that you have proposed. Therefore, it is necessary for you to determine whether the FSAR and SER need to be updated in accordance with these technical specification changes. If not, your earlier certification should be confirmed as applicable to the revised Technical Specifications. If updating of the FSAR is required, you should recertify the applicability when the changes to the FSAR have been effected.

The changes that have been incorporated are on the final draft replacement pages in the enclosure.

Additionally, the staff has prepared a revised draft license for the Wolf Creek Generating Station. We had previously sent you the prior draft license in our letter dated December 4, 1984. Enclosed is a draft copy of this revised license (without attachments and appendices) for your information. We have incorporated some of your comments on the earlier draft into this draft. We believe that this present draft accurately reflects the commitments required of you as described in the FSAR, SER and other documents.

Sincerely,

T M Novak

Thomas M. Novak, Assistant Director
for Licensing
Division of Licensing

Enclosure: As stated

cc: See next page

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Sincerely,

Original signed by:
Thomas M. Novak

Thomas M. Novak, Assistant Director
for Licensing
Division of Licensing

Enclosure: As stated

cc: See next page

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WOLF CREEK

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

KANSAS GAS AND ELECTRIC COMPANY

KANSAS CITY POWER & LIGHT COMPANY

KANSAS ELECTRIC POWER COOPERATIVE, INC.

DOCKET NO. STN 50-482

WOLF CREEK GENERATING STATION, UNIT NO. 1

FACILITY OPERATING LICENSE

License No. NPF-32

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

   A. The application for license filed by Kansas Gas and Electric Company, Kansas City Power & Light Company, and Kansas Electric Power Cooperative, Inc. (licensees), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the Wolf Creek Generating Station, Unit No. 1 (the facility) has been substantially completed in conformity with Construction Permit No. CPPR-147 and the application, as amended, the provisions of the Act, and the regulations of the Commission;

   C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission, (except as exempted from compliance in Section 2.D below);

   D. There is reasonable assurance: (i) that the activities authorized by this operating license 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 set forth in 10 CFR Chapter I, (except as exempted from compliance in Section 2D below);

   E. Kansas Gas and Electric Company* is technically qualified to engage in the activities authorized by this license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;

*Kansas Gas and Electric Company is authorized to act as agent for the Kansas City Power & Light Company and the Kansas Electric Power Cooperative, Inc., and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

F. The licensees have satisfied the applicable provisions of 10 CFR Part 140 "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

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

H. After weighing the environmental, economic, technical and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of this Facility Operating License No. NPF-32, subject to the conditions for protection of the environment set forth in the Environmental Protection Plan attached as Appendix B, is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

I. The receipt, possession, and use of source, byproduct and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70.

2. Based on the foregoing findings regarding this facility, Facility Operating License No. NPF-32 is hereby issued to Kansas Gas and Electric Company, Kansas City Power & Light Company, and Kansas Electric Power Cooperative, Inc. (the licensees) to read as follows:

A. The license applies to the Wolf Creek Generating Station, Unit No. 1, a pressurized water nuclear reactor and associated equipment (the facility), owned by Kansas Gas and Electric Company, Kansas City Power & Light Company, and Kansas Electric Power Cooperative, Inc. The facility is located in Coffey County, Kansas, approximately 28 miles east-southeast of Emporia, Kansas, and is described in the licensees' "Final Safety Analysis Report", as supplemented and amended, and in the licensees' Environmental Report, as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses Kansas Gas and Electric Company (KG&E), Kansas City Power & Light Company (KCPL) and Kansas Electric Power Cooperative, Inc. (KEPCO).

(1) Pursuant to Section 103 of the Act and 10 CFR Part 50 "Domestic Licensing of Production and Utilization Facilities," KG&E, to possess, use and operate the facility at the designated location in Coffey County, Kansas, in accordance with the procedures and limitations set forth in this license;

(2) KCPL and KEPCO to possess the facility at the designated location in Coffey County, Kansas, in accordance with the procedures and limitations set forth in this license;

(3) KG&E, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) KG&E, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) KG&E, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) KG&E, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

KG&E is authorized to operate the facility at reactor core power levels not in excess of 3411 megawatts thermal (100% power) in accordance with the conditions specified herein and in Attachment 1 to this license. The preoperational tests, startup tests and other items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license. Pending Commission approval, this license is restricted to power levels not to exceed 5 percent of full power (170 megawatts thermal);

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. KG&E shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan;

(3) Antitrust Conditions

Kansas Gas & Electric Company and Kansas City Power & Light Company shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) Environmental Qualification (Section 3.11, SSER #4, Section 3.11, SSER #5)*

All electrical equipment within the scope of 10 CFR 50.49 shall be qualified by November 30, 1985.

(5) Seismic and Dynamic Qualification (Section 3.10, SSER #5)

Prior to exceeding five percent of rated power, KG&E shall, for that equipment which is not completely qualified, complete such qualification or submit justification for safe operation at power levels greater than five percent.

(6) Fire Protection (Section 9.5.1, SER, Section 9.5.1.8, SSER #5)

(a) KG&E shall maintain in effect all provisions of the approved fire protection program as described in the SNUPPS Final Safety Analysis Report for the facility through Revision 17, the Wolf Creek site addendum through Revision 15, and as approved in the SER through Supplement 5, subject to provisions b & c below.

(b) KG&E may make no change to the approved fire protection program which would decrease the level of fire protection in the plant without prior approval of the Commission. To make such a change the licensee must submit an application for license amendment pursuant to 10 CFR 50.90.

(c) KG&E may make changes to features of the approved fire protection program which do not decrease the level of fire protection without prior Commission approval, provided:

(i) such changes do not otherwise involve a change in a license condition or technical specification or result in an unreviewed safety question (see 10 CFR 50.59).

*The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

(ii) such changes do not result in failure to complete the fire protection program approved by the Commission prior to license issuance.

KG&E shall maintain, in an auditable form, a current record of all such changes including an analysis of the effects of the change on the fire protection program and shall make such records available to NRC inspectors upon request. All changes to the approved program made without prior Commission approval shall be reported annually to the Director of the Office of Nuclear Reactor Regulation, together with supporting analyses.

(7) Qualification of Personnel (Section 13.1.2, SSER #5, Section 18, SSER #1)

KG&E shall have on each shift operators who meet the requirements described in Attachment 2.

(8) NUREG-0737 Supplement 1 Conditions (Section 22, SER)

KG&E shall complete the requirements described in Attachment 3 to the satisfaction of the NRC. These conditions reference the appropriate items in Section 22, "TMI Action Plan Requirements for Applicants for Operating Licenses," in the Safety Evaluation Report and Supplements 1, 2, 3, 4, and 5 NUREG-0881.

(9) Post-Fuel-Loading Initial Test Program (Section 14, SER Section 14, SSER #5)

Any changes in the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(10) Inservice Inspection Program (Sections 5.2.4 and 6.6, SER)

Within nine months of the date of this license, KG&E shall submit for staff review and approval, the inservice inspection program which conforms to the ASME Code in effect 12 months prior to the date of issuance of this license.

(11) Emergency Planning

(a) In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provisions of 10 CFR Section 50.54(s)(2) will apply.

(b) Prior to exceeding five percent of rated power, letters of agreement shall be signed by Coffey County with ambulance services and with funeral directors in surrounding counties providing for the transportation of non-ambulatory patients from the Coffey County Hospital and from the Golden Age Lodge Nursing Home in the event of an emergency evacuation occasioned by an accident at the Wolf Creek Plant. These executed letters of agreement shall be submitted to the NRC staff and shall be included in the Coffey County Plan.

(12) Steam Generator Tube Rupture (Section 15.4.4, SSER #5)

Prior to restart following the first refueling outage, KG&E shall submit for NRC review and approval an analysis which demonstrates that the steam generator single-tube rupture (SGTR) analysis presented in the FSAR is the most severe case with respect to the release of fission products and calculated doses. Consistent with the analytical assumptions, the licensee shall propose all necessary changes to Appendix A to this license.

(13) Low Temperature Overpressure Protection (Section 5, SSER #5)

By June 1, 1985, KG&E shall submit for NRC review and approval a description of equipment modifications to the residual heat removal system (RHRS) suction isolation valves and to closure circuitry which conform to the applicable staff requirements (SRP 5.2.2). Within one year of receiving NRC approval of the modifications, KG&E shall have the approved modifications installed. Alternately, by June 1, 1985, KG&E shall provide acceptable justification for reliance on administrative means alone to meet the staff's RHRS isolation requirements, or otherwise, propose changes to Appendix A to this license which remove reliance on the RHRS as a means of low temperature overpressure protection.

(14) LOCA Reanalysis (Section 15.3.7, SSER #5)

Prior to restart following the first refueling outage, KG&E shall submit for NRC review and approval a reanalysis for the worst large break LOCA using an approved ECCS evaluation model. At this time that model is the 1981 Westinghouse model. A modified version of the 1981 model which includes the BART computer code may be used.

(15) Generic Letter 83-28

KG&E shall submit responses to and implement the requirements of Generic Letter 83-28 on a schedule which is consistent with that given in their November 15, 1983, February 29, 1984, and February 6, 1985 letters.

(16) Surveillance of Hafnium Control Rods (Section 4.2.3.1(10), SER and SSER #2)

KG&E shall perform a visual inspection of a sample of hafnium control rods during one of the first five refueling outages. A summary of the results of these inspections shall be submitted to the NRC.

D. Exemptions from certain requirements of Appendix J to 10 CFR Part 50, and from a portion of the requirements of General Design Criterion 4 of Appendix A to 10 CFR Part 50, are described in the Safety Evaluation Report. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, these exemptions are hereby granted pursuant to 10 CFR 50.12. With the granting of these exemptions the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. KG&E shall fully implement and maintain in effect all provisions of the Commission approved Physical Security, Guard Training and Qualification, and Safeguards Contingency plans, including amendments made pursuant to the authority of 10 CFR 50.54(p). The approved plans, which contain Safeguards Information as described in 10 CFR 73.21 are collectively entitled "Wolf Creek Generating Station, Physical Security Plan Revision 0, transmitted by letter dated February 8, 1980, Revision 1, transmitted by letter dated September 8, 1981, Revision 2, transmitted by letters dated March 31, 1982 and April 30, 1982, Revision 3, transmitted by letter dated May 8, 1984, Revision 4, transmitted by letter dated August 15, 1984, Revision 5, transmitted by letter dated September 28, 1984, Revision 6, transmitted by letter dated November 30, 1984, Revision 7, transmitted by letter dated January 11, 1985, and Revision 8, transmitted by letter dated February 14, 1985; Safeguards Contingency Plan, Revision 0, transmitted by letter dated February 8,1980, Revision 1, transmitted by letter dated September 8, 1981, Revision 2, transmitted by letters dated March 31, 1982 and April 30, 1982, (Revision 3 was not submitted), Revision 4, transmitted by letter dated August 22, 1984 (Revision 5 was not submitted), Revision 6, transmitted by letter dated November 30, 1984, Revision 7, transmitted by letter dated January 11, 1985 and Revision 8, transmitted by letter dated February 14, 1985; and the Security Training and Qualification Plan Revision 0, transmitted by letter dated July 30, 1981, Revision 1, transmitted by letters dated March 31, 1982, and April 30, 1982, (Revisions 2 through 5 were not submitted), Revision 6, transmitted by letter dated November 30, 1984, Revision 7, transmitted by letter dated January 11, 1985, and Revision 8, transmitted by letter dated February 14, 1985.

F. Except as otherwise provided in the Technical Specifications or Environmental Protection Plan, the licensee shall report any violations of the requirements contained in Section 2.C of this license in the following manner: initial notification shall be made within 24 hours to the NRC Operations Center via the Emergency Notification System with written followup within thirty days in accordance with the procedures described in 10 CFR 50.73(b), (c) and (e).

G. The licensees shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

H. This license is effective as of the date of issuance and shall expire at Midnight on

FOR THE NUCLEAR REGULATORY COMMISSION

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Attachments/Appendices:
1. Attachment 1 - Tests and Other Items which must be Completed
2. Attachment 2 - Operating Staff Experience Requirements
3. Attachment 3 - NUREG-0737, Supplement 1, Requirements
4. Appendix A - Technical Specifications (NUREG-)
5. Appendix B - Environmental Protection Plan
6. Appendix C - Antitrust Conditions

Date of Issuance:

FINAL DRAFT
FEB 27 1985

FINAL DRAFT

FEB 27 1985

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	4.56	0	≤109% of RTP*	≤112.3% of RTP*
b. Low Setpoint	8.3	4.56	0	≤25% of RTP*	≤28.3% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	2.4	0.5	0	≤4% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
4. Power Range, Neutron Flux, High Negative Rate	2.4	0.5	0	≤4% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.41	0	≤25% of RTP*	≤35.3% of RTP*
6. Source Range, Neutron Flux	17.0	10.01	0	$\leq 10^5$ cps	$\leq 1.6 \times 10^5$ cps
7. Overtemperature ΔT	7.6	3.76	1.73 + 0.67	See Note 1	See Note 2
8. Overpower ΔT	5.5	1.43	0.16	See Note 3	See Note 4
9. Pressurizer Pressure-Low	3.7	0.71	2.49	≥1875 psig	≥1866 psig
10. Pressurizer Pressure-High	7.5	0.71	2.49	≤2385 psig	≤2400 psig
11. Pressurizer Water Level-High	8.0	2.18	1.96	≤92% of instrument span	≤93.9% of instrument span

*RTP = RATED THERMAL POWER
**Loop design flow = 95,700 gpm

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
12. Reactor Coolant Flow-Low	3.1	2.27	0.6	$\geq$90% of loop design flow**	$\geq$89.1% of loop design flow**
13. Steam Generator Water Level Low-Low	23.5	21.18	2.51	$\geq$23.5% of narrow range instrument span	$\geq$22.3% of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	7.5	1.3	0	$\geq$10578 Volts A.C.	$\geq$10355 Volts A.C.
15. Underfrequency - Reactor Coolant Pumps	3.3	0	0	$\geq$57.2 Hz	$\geq$57.1 Hz
16. Turbine Trip					
a. Low Fluid Oil Pressure	N.A.	N.A.	N.A.	$\geq$590.00 psig	$\geq$534.20 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	$\geq$1% open	$\geq$1% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

T'	$\leq$	588.5°F (Nominal T_{avg} at RATED THERMAL POWER);
K_3	=	0.000671;
P	=	Pressurizer pressure, psig;
P'	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform operator, s^{-1};

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant STARTUP tests such that:

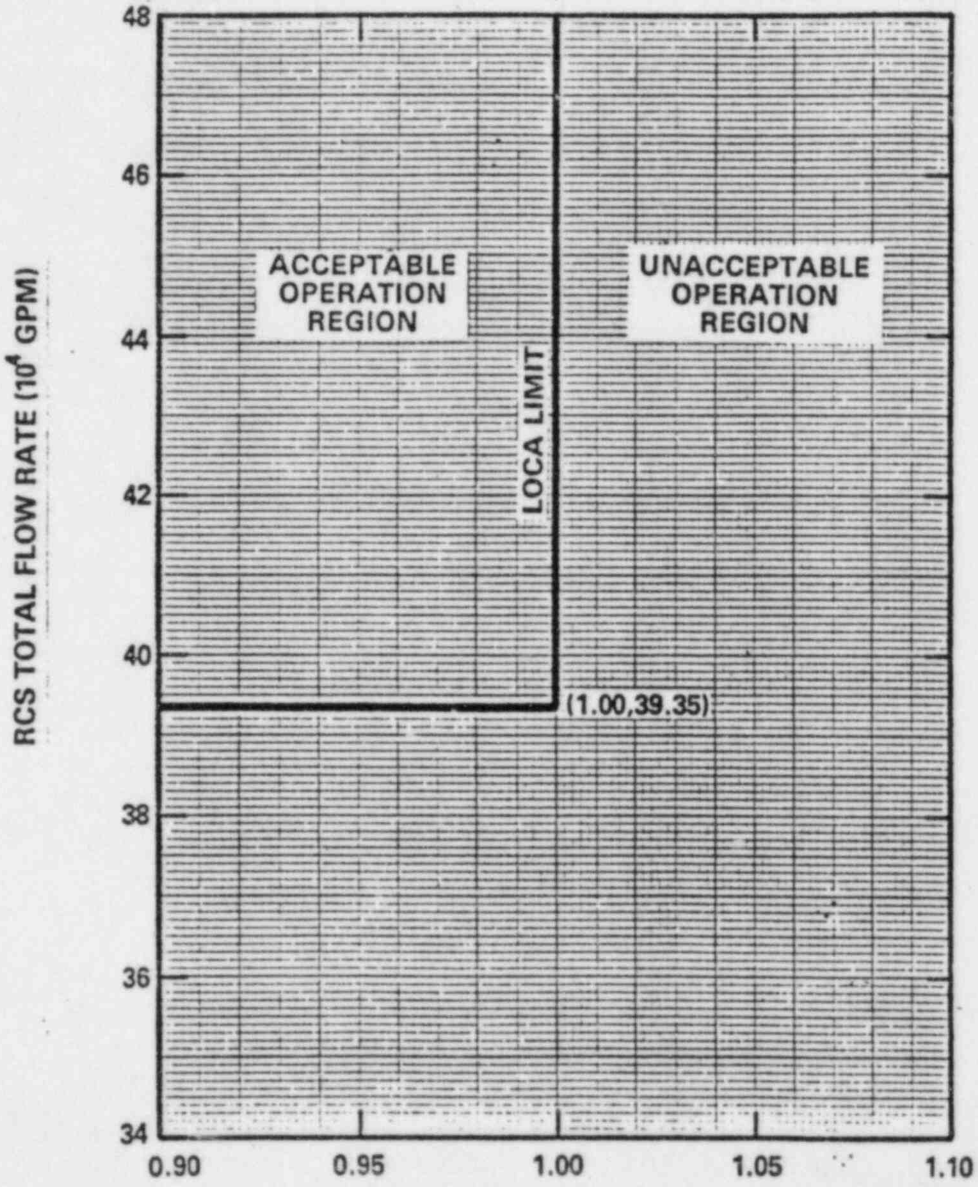
(i) for $q_t - q_b$ between -35% and + 7%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent that the magnitude of $q_t - q_b$ exceeds -35%, the ΔT Trip Setpoint shall be automatically reduced by 1.26% of its value at RATED THERMAL POWER; and

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds +7%, the ΔT Trip Setpoint shall be automatically reduced by 1.05% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.3% of ΔT span.

MEASUREMENT UNCERTAINTIES OF 2.8% FOR FLOW AND 4.0% FOR INCORE MEASUREMENT OF $F_{\Delta H}^{N}$ ARE INCLUDED IN THIS FIGURE



$R = F_{\Delta H}^{N}/1.49\left[1.0 + 0.2(1.0\text{-}P)\right]$

FIGURE 3.2-3

RCS TOTAL FLOW RATE VERSUS R
FOUR LOOPS IN OPERATION

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6 = 0.00128/°F for T > T" and K_6 = 0 for T $\leq$ T";

T = Average temperature, °F;

T" = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 588.5°F);

S = Laplace transform operator, s^{-1}; and

$f_2(\Delta I)$ = 0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 4.1% of ΔT span.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. The flow path from the Boric Acid Storage System via a boric acid transfer pump and a centrifugal charging pump to the Reactor Coolant System, and

b. Two flow paths from the refueling water storage tank via centrifugal charging pumps to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal; and

c. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2a. delivers at least 30 gpm to the Reactor Coolant System.

*The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the centrifugal charging pump declared inoperable pursuant to Specification 4.1.2.3.2 provided that the centrifugal charging pump is restored to OPERABLE status within 4 hours or prior to the temperature of one or more of the RCS cold legs exceeding 375°F, whichever comes first.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two centrifugal charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

With only one centrifugal charging pump OPERABLE, restore at least two centrifugal charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two centrifugal charging pumps shall be demonstrated OPERABLE by verifying, on recirculation flow, that the pump develops a differential pressure of greater than or equal to 2400 psid when tested pursuant to Specification 4.0.5.

*The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the centrifugal charging pump declared inoperable pursuant to Specification 4.1.2.3.2 provided that the centrifugal charging pump is restored to OPERABLE status within 4 hours or prior to the temperature of one or more of the RCS cold legs exceeding 375°F, whichever comes first.

3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R shall be maintained within the region of allowable operation shown on Figure 3.2-3 for four loop operation.

Where:

a. $R = \dfrac{F^N_{\Delta H}}{1.49\ [1.0 + 0.2\ (1.0 - P)]}$,

b. $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

c. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used to calculate R since Figure 3.2-3 includes measurement uncertainties of 2.8% for flow and 4% for incore measurement of $F^N_{\Delta H}$.

APPLICABILITY: MODE 1.

ACTION:

With the combination of RCS total flow rate and R outside the region of acceptable operation shown on Figure 3.2-3:

a. Within 2 hours either:

   1. Restore the combination of RCS total flow rate and R to within the above limits, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of R and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours; and

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	10
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2#
b. Low Setpoint	4	2	3	1###, 2	2#
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
6. Source Range, Neutron Flux					
a. Startup	2	1	2	2##**	4
b. Shutdown	2	1	2	3**, 4, 5	5
7. Overtemperature ΔT Four Loop Operation	4	2	3	1, 2	6#
8. Overpower ΔT Four Loop Operation	4	2	3	1, 2	6#
9. Pressurizer Pressure-Low	4	2	3	1	6#
10. Pressurizer Pressure-High	4	2	3	1, 2	6#

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

*Only if the Reactor Trip System breakers happen to be in the closed position and the Control Rod Drive System is capable of rod withdrawal.
**The boron dilution flux doubling signal may be blocked during reactor startup in accordance with normal operating procedures.
#The provisions of Specification 3.0.4 are not applicable.
##Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.
###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour;

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.1.1; and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint; or

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement suspend all operations involving positive reactivity changes.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 5 - a. With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip Breakers, suspend all operations involving positive reactivity changes and verify valves BG-V178 and BG-V601 are closed and secured in position within the next hour.

b. With no channels OPERABLE, open the Reactor Trip Breakers, suspend all operations involving positive reactivity changes and verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and every 12 hours thereafter, and verify valves BG-V178 and BG-V601 are closed and secured in position within 4 hours and verified to be closed and secured in position every 14 days.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour; and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.1.1.

ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 8 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 10 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor trip breakers within the next hour.

ACTION 11 - With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 1 hour.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
6. Auxiliary Feedwater (Continued)					
2) Start Turbine-Driven Pumps	23.5	21.18	2.51	≥ 23.5% of narrow range instrument span	≥ 22.3% of narrow range instrument span
e. Safety Injection - Start Motor-Driven Pumps	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
f. Loss-of-Offsite Power-Start Turbine-Driven Pump	N.A.	N.A.	N.A.	N.A.	N.A.
g. Trip of All Main Feedwater Pumps - Start Motor-Driven Pumps	N.A.	N.A.	N.A.	N.A.	N.A.
h. Auxiliary Feedwater Pump Suction Pressure-Low (Transfer to ESW)	N.A.	N.A.	N.A.	≥ 21.60 psia	≥ 20.53 psia
7. Automatic Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	N.A.
b. RWST Level-Low-Low	3.4	1.21	1.86	≥ 36% of instrument span	≥ 35.1% of instrument span
Coincident with Safety Injection	See Item 1. above for Safety Injection Trip Setpoints and Allowable Values.				

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
8. Loss of Power					
a. 4 kV Undervoltage -Loss of Voltage	N.A.	N.A.	N.A.	$\geq$ 83V (120V Bus) w/1s delay	$\geq$ 74.7V (120V Bus) w/1 + 0.2, -0.5s delay
b. 4 kV Undervoltage -Grid Degraded Voltage	N.A.	N.A.	N.A.	$\geq$ 106.9V (120V Bus) w/119s delay	$\geq$ 104.3V (120V Bus) w/119 ± 11.6s delay
9. Control Room Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	N.A
c. Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	N.A.	N.A.	N.A.	N.A.	N.A.
d. Phase "A" Isolation	See Item 3.a. above for all Phase "A" Isolation Trip Setpoints and Allowable Values.				
10. Solid-State Load Sequencer	N.A.	N.A.	N.A.	N.A.	N.A.
11. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	$\leq$ 1970 psig	$\leq$ 1979 psig
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq 29^{(1)}/12^{(4)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 7
3) Phase "A" Isolation	$\leq 2^{(5)}$
4) Auxiliary Feedwater	≤ 60
5) Essential Service Water	$\leq 60^{(1)}$
6) Containment Cooling	$\leq 60^{(1)}$
7) Component Cooling Water	N.A.
8) Emergency Diesel Generators	$\leq 14^{(6)}$
9) Turbine Trip	N.A.
4. Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq 24^{(3)}/12^{(4)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 7
3) Phase "A" Isolation	$\leq 2^{(5)}$
4) Auxiliary Feedwater	≤ 60
5) Essential Service Water	$\leq 60^{(1)}$
6) Containment Cooling	$\leq 60^{(1)}$
7) Component Cooling Water	N.A.
8) Emergency Diesel Generators	$\leq 14^{(6)}$
9) Turbine Trip	N.A.
b. Steam Line Isolation	$\leq 2^{(5)}$

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
5. Containment Pressure-High-3	
a. Containment Spray	$\leq 32^{(1)}/20^{(2)}$
b. Phase "B" Isolation	$\leq$ 31.5
6. Containment Pressure-High-2	
Steam Line Isolation	$\leq 2^{(5)}$
7. Steam Line Pressure-Negative Rate-High	
Steam Line Isolation	$\leq 2^{(5)}$
8. Steam Generator Water Level-High-High	
a. Turbine Trip	$\leq$ 2.5
b. Feedwater Isolation	$\leq$ 7
9. Steam Generator Water Level - Low-Low	
a. Start Motor-Driven Auxiliary Feedwater Pumps	$\leq$ 60
b. Start Turbine-Driven Auxiliary Feedwater Pumps	$\leq$ 60
10. Loss-of-Offsite Power	
Start Turbine-Driven Auxiliary Feedwater Pumps	N.A.
11. Trip of All Main Feedwater Pumps	
Start Motor-Driven Auxiliary Feedwater Pumps	N.A

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays (SSPS)	N.A	N.A.	N.A.	N.A.	M(1)	M(1)	Q(3)	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
b. Phase "B" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Containment Pressure-High-3	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Containment Purge Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(3)	1, 2, 3, 4
3) Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	N.A.	N.A.	N.A.	N.A.	M(1)(2)	N.A.	N.A.	1, 2, 3, 4
4) Phase "A" Isolation	See Item 3.a. above for all Phase "A" Isolation Surveillance Requirements.							

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

FUNCTIONAL UNIT	CHANNELS TO TRIP/ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. Containment					
a. Containment Atmosphere-Gaseous Radioactivity-High (GT-RE-31 & 32)	1	2	All	###	26
b. Gaseous Radioactivity-RCS Leakage Detection (GT-RE-31 & 32)	N.A.	1	1, 2, 3, 4	N.A.	29
c. Particulate Radioactivity-RCS Leakage Detection (GT-RE-31 & 32)	N.A.	1	1, 2, 3, 4	N.A.	29
2. Fuel Building					
a. Fuel Building Exhaust-Gaseous Radioactivity-High (GG-RE-27 & 28)	1	2	**	##	30
b. Criticality-High Radiation Level					
1) Spent Fuel Pool (SD-RE-37 or 38)	1	1	*	≤ 15 mR/h	28
2) New Fuel Pool (SD-RE-35 or 36)	1	1	*	≤ 15 mR/h	28
3. Control Room					
Air Intake-Gaseous Radioactivity-High (GK-RE-04 & 05)	1	2	All	#	27

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure		
a) Normal Range	2	1
b) Extended 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	1/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. PORV Position Indicator*	1/Valve	1/Valve
13. PORV Block Valve Position Indicator**	1/Valve	1/Valve
14. Safety Valve Position Indicator	1/Valve	1/Valve
15. Containment Water Level	2	1
16. Containment Radiation Level (High Range)	N.A.	1
17. Thermocouple/Core Cooling Detection System	4/core quadrant	2/core quadrant
18. Unit Vent - High Range Noble Gas Monitor	N.A.	1

*Not applicable if the associated block valve is in the closed position.

**Not applicable if the block valve is verified in the closed position and power is removed.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	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. PORV Position Indicator*	M	N.A.
13. PORV Block Valve Position Indicator**	M	N.A.
14. Safety Valve Position Indicator	M	N.A.
15. Containment Water Level	M	R
16. Containment Radiation Level (High Range)	M	R***
17. Thermocouple/Core Cooling Detection System	M	R
18. Unit Vent - High Range Noble Gas Monitor	M	R

*Not applicable if the associated block valve is in the closed position.
**Not applicable if the block valve is verified in the closed position and power is removed.
***CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/h and a one point calibration check of the detector below 10 R/h with an installed or portable gamma source.

TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	ZONE	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
1101-Aux. Bldg. 1974' Gen. Flr. #1	100			0/11
1102-Chiller & Surge Tks. Area	100			0/4
1102-Chiller & Surge Tks. Area	101			2/0
1107-Cent. Charg. Pmp. Rm. B	101			2/0
1108-Safety Inj. Pmp. Rm. B	101			2/0
1109-Res. Ht. Remov. Pmp. Rm. B	101			1/0
1110-Ctmt. Spray Pmp. Rm. B	101			1/0
1111-Res. Ht. Remov. Pmp. Rm. A	101			1/0
1112-Ctmt. Spray Pmp. Rm. A	101			1/0
1113-Safety Inj. Pmp. Rm. A	101			2/0
1114-Cent. Charg. Pmp. Rm. A	101			2/0
1115-Pos. Disp. Charg. Pmp. Rm.	101			2/0
1116, 1117-Boric Acid Tk. Rms.	101			2/0
1116, 1117-Boric Acid Tk. Rms.	101		2/0	
1120-Aux. Bldg. 1974' Gen. Flr. #2	101			4/0
1122-Aux. Bldg. 1974' Gen. Flr. #3	100			0/3
1122-Aux. Bldg. 1974' Gen. Flr. #3	101			5/0
1126-Boron Inj. Tk. & Pmp. Rm.	101			1/0
1127-Stair A-Z	109			1/0
1128-Aux. Feedwater Pump Rm. Basement	117			2/0
1130-Aux. Bldg. 1974' N. Corr.	100			0/2
1206-W. Pipe Chase Below AFWP Area	117			2/0
1203-Aux. Bldg. Elec. Chase S. 1988'	117			1/0
1301-Aux. Bldg. 2000' Corridor #1	103			0/10
1301-Aux. Bldg. 2000' Corridor #1	117			2/0
1311-Aux. Bldg. Sampling Rm.	117			2/0
1312-Boron Meter/RC Activity Mon. Rm.	103			0/1
1314-Aux. Bldg. 2000' Corridor #3	103			0/3
1314-Aux. Bldg. 2000' Corridor #3	117			2/0
1315-Ctmt. Spray Add. Tk. Area	103			0/2
1316-Vlv. Rm. by Seal Wtr. Ht. Exch.	103			0/1
1320-Aux. Bldg. 2000' Corridor #4	103			0/3
1321-Aux. Bldg. 2000' S. Exit Vest.	103			0/1
1322-Pipe Pene. Rm. B	117			5/0
1323-Pipe Pene. Rm. A	117			6/0
1325-Aux. FW Pmp. Rm. B	117			2/0
1326-Aux. FW Pmp. Rm. A	117			2/0
1331-Aux. FW Pmp. Rm. C	111		2/0	
1331-Aux. FW Pmp. Rm. C	117			1/0
1335-Aux. Bldg. Elec. Chase N. 2000'	117			1/0
1336-Aux. Bldg. Elec. Chase S. 2000'	117			1/0
1401-Comp. Cool. Pmp. & Ht. Exch. B	118			5/0
1402-Aux. Bldg. 2026' Corridor #1	104			0/6
1403-MG Set Rm.	105			0/9(1)
1403-MG Set Rm.	112			0/9(1)

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TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	ZONE	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
1405-Chemical Stg. Area	118			6/0
1406-Comp. Cool. Pmp. & Ht. Exch. A	104			0/1
1406-Comp. Cool. Pmp. & Ht. Exch. A	118			2/0
1408-Aux. Bldg. 2026' Corridor #2	104			0/9
1408-Aux. Bldg. 2026' Corridor #2	118			5/0
1409-Elec. Pene. Rm. B	106			0/4(1)
1409-Elec. Pene. Rm. B	113			0/4(1)
1410-Elec. Pene. Rm. A	107			0/8(1)
1410-Elec. Pene. Rm. A	114			0/8(1)
1413-Aux. Shutdown Pnl. Rm.	118			4/0
1501-Ctrl. Rm. A/C & Filt. Units B	110			10/0
1504-Ctmt. Purge Exh. & Mech. Equip. B	108			18/0
1506-Cmt. Purge Sup. AHU Rm. A	109			18/0
1507-Personnel Hatch Area	108			3/0
1508-Main Steam Iso. Valve Rm #1	115		1/0	
1509-Main Steam Iso. Valve Rm #2	115		1/0	
1512-Ctrl. Rm. A/C & Filt. Units A	110			10/0
1513-Ctrl. Bldg. Vent Sup. A/C Unit Rm.	109			3/0
1513-Aux. Bldg. Duct 2047'6"	119			1/0
N.A.-Containment**	201	1/0(2)		
N.A.- Containment**	202	2/0(2)		
N.A.-Containment**	203	1/0(2)		
N.A.-Containment**	204	1/0(2)		
N.A.-Containment**	206	3/0(2)		
N.A.-Containment**	215	1/0(2)		
N.A.-Containment**	216	1/0(2)		
N.A.-Containment**	217	1/0(2)		
N.A.-Containment**	218	1/0(2)		
N.A.-Containment**	219			4/0
N.A.-Containment**	220	1/0(2)		
3101-Ctrl. Bldg. 1974' Pipe Space	300			11/0
3105-Ctrl. Bldg. Elec. Chase S. 1974'	300			1/0
3106-Ctrl. Bldg. Elec. Chase N. 1974'	300			1/0
N.A.-Area Above Access Control	301			12/0
3229-Ctrl. Bldg. Elec. Chase S. 1984'	300			1/0
3230-Ctrl. Bldg. Elec. Chase N. 1984'	300			1/0
3301-ESF Swgr. Rm. #1	314			0/7(1)
3301-ESF Swgr. Rm. #1	315			0/7(1)
3302-ESF Swgr. Rm. #2	316			0/5(1)
3302-ESF Swgr. Rm. #2	317			0/5(1)
3305-Ctrl. Bldg. Elec. Chase S. 2000'	301			1/0
3306-Ctrl. Bldg. Elec. Chase N. 2000'	301			1/0
3403-Non-Vit. Swgr. & Xfmr. Rm. #1	304			0/1(1)
3403-Non-Vit. Swgr. & Xfmr. Rm. #1	305			0/1(1)
3404-Switchboard Rm. #4	321			0/2(1)

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	ZONE	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
3404-Switchboard Rm. #4	322			0/2(1)
3405-Battery Rm. #4	303			2/0
3407-Battery Rm. #1	303			2/0
3408-Switchboard Rm. #1	325			0/2(1)
3408-Switchboard Rm. #1	326			0/2(1)
3409-Non-Vit. Swgr. & Xfmr. Rm. #2	323			0/1(1)
3409-Non-Vit. Swgr. & Xfmr. Rm. #2	327			0/1(1)
3410-Switchboard Rm. #2	324			0/2(1)
3410-Switchboard Rm. #2	328			0/2(1)
3411-Battery Rm. #2	303			2/0
3413-Battery Rm. #3	303			1/0
3414-Switchboard Rm. #3	318			0/2(1)
3414-Switchboard Rm. #3	320			0/2(1)
3415-Acc. Ctrl. & Elec. Equip. A/C Units #1	303			4/0
3416-Acc. Ctrl. & Elec. Equip. A/C Units #2	303			4/0
3418-Ctrl. Bldg. Elec. Chase S. 2016'	303			1/0
3419-Ctrl. Bldg. Elec. Chase N. 2016'	303			1/0
3414-Ctrl. Bldg. Elec. Chase N. 2016'	303			1/0
3410-Ctrl. Bldg. Elec. Chase S. 2016'	303			1/0
3501-Lower Cable Spreading Rm.	306			0/13
3504-Ctrl. Bldg. Elec. Chase N. 2032'	303			1/0
3505-Ctrl. Bldg. Elec. Chase S. 2032'	303			1/0
3501-Ctrl. Bldg. Elec. Chase N. 2032'	303			1/0
3501-Ctrl. Bldg. Elec. Chase S. 2032'	303			1/0
3601-Control Room	308			4/0
3601-Control Room	309			0/7(1)
3601-Control Room	319			0/7(1)
3601-Control Room	329			20/0
3602-Pantry	308			1/0
3603-Shift Supv. Office	308			1/0
3605-Equipment Cabinet Area	308			15/0
3606-Emerg. Equip. Storage Rm.	308			1/0
3608-Janitor's Closet	308			1/0
3609-SAS Rm.	308			1/0
3617-Ctrl. Bldg. Elec. Chase S. 2047'6"	308			1/0
3618-Ctrl. Bldg. Elec. Chase N. 2047'6"	308			1/0
3605-Ctrl. Bldg. Elec. Chase S. 2047'6"	308			1/0
3801-Upper Cable Spreading Rm.	307			0/18
3804-Ctrl. Bldg. Elec. Chase S. 2073'6"	308			1/0
3801-Ctrl. Bldg. Elec. Chase S. 2073'6"	308			1/0
5201-W. Diesel Gen. Rm.	501		4/0	
5201-W. Diesel Gen. Rm.	502	0/8		
5203-E. Diesel Gen. Rm.	500		4/0	

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	ZONE	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
5203-E. Diesel Gen. Rm.	503	0/8		
6102-Fuel Bldg. Railroad Bay	600	0/8		
6104-Fuel Pool Cool. HX Rm. B	601			6/0
6105-Fuel Pool Cool. HX Rm. A	601			6/0
6202-Elec. Equipment Rm.	601			3/0
6203-Air Handling Equip. Rm.	601			3/0
6301-Fuel Bldg. 2047'6" Gen. Flr.	602		2/0	
6303-Fuel Bldg. Exh. Filt. Absorb. Rm. A	601			2/0
6304-Fuel Bldg. Exh. Filt. Absorb. Rm. B	601			2/0
N.A.-ESW Pumphouse Train B	002			3/0
N.A.-ESW Pumphouse Train A	001			3/0
N.A.-ESF Transformer XNB01	016	0/6		
N.A.-ESF Transformer XNB02	017	0/6		

TABLE NOTATIONS

*(x/y): x is number of Function A (early warning fire detection and notification only) instruments.
y is number of Function B (actuation of fire suppression systems and early warning and notification) instruments.

**The fire detection instruments located within the containment are not required to be OPERABLE during the performance of Type A containment leakage rate tests.

(1) Zone is associated with a Halon-protected space. Each space has two separate detection circuits (zones). One zone, in its entirety, needs to remain OPERABLE.

(2) Line-type heat detector.

TABLE 3.3-13 (Continued)

TABLE NOTATIONS

* At all times.

** During WASTE GAS HOLDUP SYSTEM operation.

ACTION STATEMENTS

ACTION 38 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment for up to 14 days provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 39 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours.

ACTION 40 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 12 hours and these samples are analyzed for radioactivity within 24 hours.

ACTION 41 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend PURGING of radioactive effluents via this pathway.

ACTION 42 - With the Outlet Oxygen Monitor channel inoperable, operation of the system may continue provided grab samples are taken and analyzed at least once per 24 hours. With both oxygen channels or both the inlet oxygen and inlet hydrogen channels inoperable, suspend oxygen supply to the recombiner. Addition of waste gas to the system may continue provided grab samples are taken and analyzed at least once per 4 hours during degassing operations and at least once per 24 hours during other operations.

ACTION 43 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue for up to 30 days provided samples are continuously collected with auxiliary sample equipment as required in Table 4.11-2.

ACTION 44 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirements, suspend oxygen supply to the recombiner.

ACTION 45 - Flow rate for this system shall be based on fan status and operating curves or actual measurements.

TABLE 4.3-9 (Continued)

TABLE NOTATIONS

* At all times.

** During WASTE GAS HOLDUP SYSTEM operation.

(1) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation as appropriate occur if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm/Trip Setpoint (isolation and alarm), or

b. Circuit failure (alarm only), or

c. Instrument indicates a downscale failure (alarm only) or

d. Instrument controls not set in operate mode (alarm only).

(2) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any one or combination of the following conditions exists:

a. Instrument indicates measured levels above the Alarm Setpoint

b. Circuit failure

c. Instrument indicates a downscale failure

d. Instrument controls not set in operate mode.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference (gas or liquid and solid) standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy, measurement range, and establish monitor response to a solid calibration source. For subsequent CHANNEL CALIBRATION, NBS traceable standard (gas, liquid, or solid) may be used; or a gas, liquid, or solid source that has been calibrated by relating it to equipment that was previously (within 30 days) calibrated by the same geometry and type of source traceable to NBS.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

a. One volume percent hydrogen, balance nitrogen, and

b. Four volume percent hydrogen, balance nitrogen.

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least three of the reactor coolant loops listed below shall be OPERABLE and at least two of these reactor coolant loops shall be in operation:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump, and

d. Reactor Coolant Loop D and its associated steam generator and reactor coolant pump.

APPLICABILITY: MODE 3.**

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With only one reactor coolant loop in operation, restore at least two loops to operation within 72 hours or within 1 hour open the Reactor Trip System breakers.

c. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required reactor coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side wide range water level to be greater than or equal to 10% at least once per 12 hours.

4.4.1.2.3 At least two reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

*All reactor coolant pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**See Special Test Exception Specification 3.10.4.

4.4.5.4 Acceptance Criteria

a. As used in this specification:

1) Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections;

2) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube;

3) Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;

4) % Degradation means the percentage of the tube wall thickness affected or removed by degradation;

5) Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective;

6) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness;

7) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in Specification 4.4.5.3c., above;

8) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg; and

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following Overpressure Protection Systems shall be OPERABLE:

a. Two residual heat removal (RHR) suction relief valves each with a Setpoint of 450 psig ± 3%, or

b. Two power-operated relief valves (PORVs) with Setpoints which do not exceed the limit established in Figure 3.4-4, or

c. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2 square inches.

APPLICABILITY: MODE 3 when the temperature of any RCS cold leg is less than or equal to 368°F, MODES 4 and 5, and MODE 6 with the reactor vessel head on.

ACTION:

a. With one PORV and one RHR suction relief valve inoperable, either restore two PORVs or two RHR suction relief valves to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2 square inch vent within the next 8 hours.

b. With both PORVs and both RHR suction relief valves inoperable, depressurize and vent the RCS through at least a 2 square inch vent within 8 hours.

c. In the event the PORVs, or the RHR suction relief valves, or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or the RHR suction relief valves, or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System accumulator shall be OPERABLE with:

a. The isolation valve open and power removed,

b. A contained borated water volume of between 6122 and 6594 gallons,

c. A boron concentration of between 1900 and 2100 ppm, and

d. A nitrogen cover-pressure of between 585 and 665 psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying, by the absence of alarms, the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each accumulator isolation valve is open.

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*Pressurizer pressure above 1000 psig.

3/4.5.2 ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE Safety Injection pump,

c. One OPERABLE RHR heat exchanger,

d. One OPERABLE RHR pump, and

e. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and automatically transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

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*The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the centrifugal charging pumps and the Safety Injection pumps declared inoperable pursuant to Specification 4.5.3.2 provided the centrifugal charging pumps and the Safety Injection pumps are restored to OPERABLE status within 4 hours or prior to the temperature of one or more of the RCS cold legs exceeding 375°F, whichever comes first.

SURVEILLANCE REQUIREMENTS (Continued)

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

   1) For centrifugal charging pump lines, with a single pump running:

      a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 346 gpm, and

      b) The total pump flow rate is less than or equal to 556 gpm.

   2) For Safety Injection pump lines, with a single pump running:

      a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 459 gpm, and

      b) The total pump flow rate is less than or equal to 665 gpm.

i. By performing a flow test, during shutdown, following completion of modifications to the RHR subsystems that alter the subsystem flow characteristics and verifying that the RHR pump lines, with a single pump running:

   1) The sum of the injection line flow rates is greater than or equal to 3800 gpm, and

   2) The total pump flow rate is less than or equal to 5500 gpm.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable requirements of Specification 4.5.2.

4.5.3.2 All centrifugal charging pumps and Safety Injection pumps, except the above allowed OPERABLE pumps, shall be demonstrated inoperable* by verifying that the motor circuit breakers are secured in the open position within 4 hours after entering MODE 4 from MODE 3 or prior to the temperature of one or more of the RCS cold legs decreasing below 325°F whichever comes first, and at least once per 31 days thereafter.

*An inoperable pump may be energized for testing or for filling accumulators provided the discharge of 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.

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3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained borated water volume of 394,000 gallons,

b. A boron concentration of between 2000 and 2100 ppm of boron,

c. A minimum solution temperature of 37°F, and

d. A maximum solution temperature of 100°F.

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

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status 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

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the contained borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is either less than 37°F or greater than 100°F.

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1) Less than or equal to L_a, 0.20% by weight of the containment air per 24 hours at P_a, 48 psig, or

2) Less than or equal to L_t, 0.020% by weight of the containment air per 24 hours at P_t, 24 psig.

b. 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 psig.

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

ACTION:

With either the measured overall integrated containment leakage rate exceeding 0.75 L_a or 0.75 L_t, as applicable, or the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, restore the overall integrated leakage rate to less than 0.75 L_a or less than L_t, as applicable, and the combined leakage rate for all penetrations subject to Type B and C tests to less than 0.60 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using the methods and provisions of ANSI N45.4-1972:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at a pressure not less than either P_a, 48 psig, or P_t, 24 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection;

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valves shall be OPERABLE and:

a. Each 36-inch containment shutdown purge supply and exhaust isolation valve shall be closed and blank flanged, and

b. The 18-inch containment mini-purge supply and exhaust isolation valve(s) may be open for up to 2000 hours during a calendar year.

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

ACTION:

a. With a 36-inch containment purge supply and/or exhaust isolation valve open or not blank flanged, close and/or blank flange that valve or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the 18-inch containment mini-purge supply and/or exhaust isolation valve(s) open for more than 2000 hours during a calendar year, close the open 18-inch valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

c. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate in excess of the limits of Specifications 4.6.1.7.2 and/or 4.6.1.7.4, restore the inoperable valve(s) 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.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (Seconds)
1. Phase "A" Isolation (active) - (Continued)				
P-32	LF FV-96	CTMT Normal Sumps to Floor Drain Tank Outside CTMT Iso	C	4
P-93	SJ HV-5**	PZR/RCS Liquid Sample Inner CTMT Iso	C	5
P-93	SJ HV-6**	PZR/RCS Liquid Sample Outer CTMT Iso	C	5
P-69	SJ HV-12**	PZR Vapor Sample Inner CTMT Iso	C	5
P-69	SJ HV-13**	PZR Vapor Sample Outer CTMT Iso	C	5
P-95	SJ HV-18**	Accumulator Sample Inner CTMT Iso	C	5
P-95	SJ HV-19**	Accumulator Sample Outer CTMT Iso	C	5
p-93	SJ HV-127**	PZR/RCS Liquid Sample Outer CTMT Iso	C	5
P-64	SJ HV-128**	PZR/RCS Liquid Sample Inner CTMT Iso	A,C	5
P-64	SJ HV-129**	PZR/RCS Liquid Sample Outer CTMT Iso	A,C	5
P-64	SJ HV-130**	PZR/RCS Liquid Sample Outer CTMT Iso Valve	A,C	5
P-57	SJ HV-131**	PASS Discharge to RCDT	A,C	5
P-57	SJ HV-132**	PASS Discharge to RCDT	A,C	5
2. Phase "A" Isolation (passive)*				
P-58	EM HV-8888**	Accumulator Tank Fill Line Iso Valve	C	5

*May be opened on an intermittent basis under administrative control.
**The provisions of Specification 3.0.4 are not applicable.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (Seconds)
6. Remote Manual - (Continued)				
P-29	EF HV-48	ESW Return From Containment Coolers	C	N.A.
P-73	EF HV-49	ESW Return From Containment Coolers	C	N.A.
P-29	EF HV-50	ESW Return From Containment Coolers	C	N.A.
P-74	EG HV-127*	CCW Supply to RCP	C	N.A.
P-75	EG HV-130*	CCW Return From RCP	C	N.A.
P-75	EG HV-131*	CCW Return From RCP	C	N.A.
P-76	EG HV-132*	CCW Return From RCP Thermal Barriers	C	N.A.
P-76	EG HV-133*	CCW Return from RCP Thermal Barrier	C	N.A.
P-79	EJ HV-8701A	RCS Hot Leg 1 to RHR Pump A Suction	A	N.A.
P-52	EJ HV-8701B	RCS Hot Leg 4 to RHR Pump B Suction	A	N.A.
P-82	EJ HV-8809A	RHR Pump A Cold Leg Injection Iso Valve	A	N.A.
P-27	EJ HV-8809B	RHR Pump B Cold Leg Injection Iso Valve	A	N.A.
P-15	EJ HV-8811A	CTMT Recirc Sump to RHR Pump A Suction	A	N.A.

*These valves were assumed to be closed during the accident analysis, and are normally closed but may be opened on an intermittent basis under administrative control.

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TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (Seconds)
8. Hand-Operated and Check Valves - (Continued)				
P-24	BG V-135	RCP Seal Water Return	C	N.A.
P-80	BG 8381	CVCS Charging Line	C	N.A.
P-25	BL 8046	Reactor Makeup Water Supply	C	N.A.
P-78	BM V-045	Steam Generator Drain Line Iso Valve	C	N.A.
P-78	BM V-046	Steam Generator Drain Line Iso Valve	C	N.A.
P-53	EC V-083	Refueling Pool Supply From Fuel Pool Cleanup	C	N.A.
P-53	EC V-084	Refueling Pool Supply From Fuel Pool Cleanup	C	N.A.
P-54	EC V-087	Refueling Pool Return to Fuel Pool Cooling	C	N.A.
P-54	EC V-088	Refueling Pool Return to Fuel Pool Cooling	C	N.A.
P-55	EC V-095	Refueling Pool Skimmers To Fuel Pool Cooling Loop	C	N.A.
P-55	EC V-096	Refueling Pool Skimmers To Fuel Pool Cooling Loop	C	N.A.
P-74	EG V-204	CCW Supply to RCP	C	N.A.
P-82	EP 8818A	RHR Pump to Cold Leg 1 Injection	A	N.A.
P-82	EP 8818B	RHR Pump to Cold Leg 2 Injection	A	N.A.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (Seconds)
8. Hand-Operated and Check Valves - (Continued)				
P-27	EP 8818C	RHR Pump to Cold Leg 3 Injection	A	N.A.
P-27	EP 8818D	RHR Pump to Cold Leg 4 Injection	A	N.A.
P-21	EJ 8841A	RHR Pump Disch to RCS Hot Leg 2	A	N.A.
P-21	EJ 8841B	RHR Pump Disch to RCS Hot Leg 3	A	N.A.
P-87	EM V-001	SI Pump Hot Leg 1 Injection	A	N.A.
P-87	EM V-002	SI Pump Hot Leg 2 Injection	A	N.A.
P-48	EM V-003	SI Pump Hot Leg 3 Injection	A	N.A.
P-48	EM V-004	SI Pump Hot Leg 4 Injection	A	N.A.
P-58	EM V-006	Accumulator Fill Line From SI Pumps	C	N.A.
P-49	EP V-010	SI Pump Disch to Cold Leg 1	A	N.A.
P-49	EP V-020	SI Pump Disch to Cold Leg 2	A	N.A.
P-49	EP V-030	SI Pump Disch to Cold Leg 3	A	N.A.
P-49	EP V-040	SI Pump Disch to Cold Leg 4	A	N.A.
P-88	EM V-8815	BIT to RCS Cold Leg Injection	A	N.A.
P-89	EN V-013	CTMT Spray Pump A to CTMT Spray Nozzles	A	N.A.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (Seconds)
8. Hand-Operated and Check Valves - (Continued)				
P-66	EN V-017	CTMT Spray Pump B to CTMT Spray Nozzles	A	N.A.
P-45	EP V-046	Accumulator Nitrogen Supply Line	C	N.A.
P-43	HD V-016	Auxiliary Steam to Decon System	C	N.A.
P-43	HD V-017	Auxiliary Steam to Decon System	C	N.A.
P-63	KA V-039	Rx Bldg Service Air Supply	C	N.A.
P-63	KA V-118	Rx Bldg Service Air Supply	C	N.A.
P-98	KB V-001	Breathing Air Supply to RX Bldg	C	N.A.
P-98	KB V-002	Breathing Air Supply to RX Bldg	C	N.A.
P-30	KA V-204	Rx Bldg Instrument Air Supply	C	N.A.
P-67	KC V-478	Fire Protection Supply to RX Bldg	C	N.A.
P-57	SJ V-111	Liquid Sample from PASS to RCDT	A,C	N.A.

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TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (Seconds)
9. Other Automatic Valves				
P-1	AB-HV-11***	Mn. Stm. Isol.	A	N.A.
P-2	AB-HV-14***	Mn. Stm. Isol.	A	N.A.
P-3	AB-HV-17***	Mn. Stm. Isol.	A	N.A.
P-4	AB-HV-20***	Mn. Stm. Isol.	A	N.A.
P-5	AE-FV-42***	Mn. FW Isol.	A	N.A.
P-6	AE-FV-39***	Mn. FW Isol.	A	N.A.
P-7	AE-FV-40***	Mn. FW Isol.	A	N.A.
P-8	AE-FV-41***	Mn. FW Isol.	A	N.A.
P-9	BM-HV-4**	SG Blowdn. Isol.	A	10
P-10	BM-HV-1**	SG Blowdn. Isol.	A	10
P-11	BM-HV-2**	SG Blowdn. Isol.	A	10
P-12	BM-HV-3**	SG Blowdn. Isol.	A	10

**The provisions of Specification 3.0.4 are not applicable.

***These valves are included for table completeness. The requirements of Section 3.6.3 do not apply; instead, the requirements of Specification 3.7.1.5 and Specification 3.3.2 apply to the Main Steam Isolation Valves and Main Feedwater Isolation Valves, respectively.

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HYDROGEN MIXING SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.4.3 Two independent hydrogen mixing systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one independent hydrogen mixing system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.3 Each independent hydrogen mixing system shall be demonstrated OPERABLE:

a. At least once per 92 days on a STAGGERED TEST BASIS by starting each non-operating system from the control room and verifying that the system operates for at least 15 minutes.

b. AT least once per 18 months by verifying that a Safety Injection test signal, the systems start in slow speed or, if operating, shift to slow speed.

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

   1) Verifying that the Control Room Emergency Ventilation System satisfies the in-place penetration and bypass leakage testing acceptance criteria; of less than 1% for HEPA filters and 0.05% for charcoal adsorbers and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 2000 cfm ±10% at greater than or equal to 6.6 inches Water Gauge (W.G.) (dirty filter) for the Filtration System and 2200 cfm ±10% at greater than or equal to 3.8 inches W.G. (dirty filter) for the Pressurization System with 500 cfm ±10% going through the Pressurization System filter adsorber unit;

   2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%; and

   3) Verifying system flow rate of 2000 cfm ~~+3, -0%~~ ±10% at greater than or equal to 6.6 inches W.G. (dirty filter) for the Filtration System and 2200 cfm ±10% at greater than or equal to 3.8 inches W.G. (dirty filter) for the Pressurization System with 500 cfm ±10% going through the Pressurization System filter adsorber unit during system operation when tested in accordance with ANSI N510-1980.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%;

e. At least once per 18 months by:

   1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6.6 inches Water Gauge while operating the system at a flow rate of 2000 cfm ±10% for the Filtration System and 500 cfm ±10% for the Pressurization System filter adsorber unit,

   2) Verifying that on a Control Room Ventilation Isolation or High Gaseous Radioactivity test signal, the system automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks,

SURVEILLANCE REQUIREMENTS (Continued)

3) Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/4 inch Water Gauge relative to the outside atmosphere during system operation,

4) Verifying that the Pressurization System filter adsorber unit heaters dissipate 15 ± 2 kW in the Pressurization System when tested in accordance with ANSI N510-1975, and

5) Verifying that on a High Chlorine test signal, the system automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks within 15 seconds.

f. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing criteria of less than 1% for HEPA filters and 0.05% for charcoal adsorbers in accordance with ANSI N510-1975 (however Prerequisite Testing, Sections 8 and 9 shall be in accordance with ANSI N510-1980) for a DOP test aerosol while operating the system at a flow rate of 2000 cfm ±10% for the Filtration System and 500 cfm ±10% for the Pressurization System filter adsorber unit; and

g. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing criteria of less than 1% for HEPA filters and 0.05% for charcoal adsorbers in accordance with ANSI N510-1975 (however Prerequisite Testing, Sections 8 and 9 shall be in accordance with ANSI N510-1980) for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 2000 cfm ±10% for the Filtration System and 500 cfm ±10% for the Pressurization System filter adsorber unit. .

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3/4.7.7 EMERGENCY EXHAUST SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 Two independent Emergency Exhaust Systems shall be OPERABLE.

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

ACTION:

With one Emergency Exhaust System inoperable, restore the inoperable Emergency Exhaust System to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.7 Each Emergency Exhaust System shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;

b. At least once per 18 months, or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system, by:

   1) Verifying that the Emergency Exhaust System satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% for HEPA filters and 0.05% for charcoal adsorbers and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 9000 cfm ±10% at $\geq$ 7.2 inches W.G. (dirty filter);

   2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%;

SURVEILLANCE REQUIREMENTS (Continued)

3) Verifying a system flow rate of 9000 cfm ±10% at $\geq$ 7.2 inches W.G. (dirty filter) during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%;

d. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks of less than or equal to 7.2 inches Water Gauge while operating the system at a flow rate of 9000 cfm ±10%,

2) Verifying that the system maintains the Fuel Building at a negative pressure of greater than or equal to ¼ inch Water Gauge relative to the outside atmosphere during system operation,

3) Verifying that the system starts on a Safety Injection test signal, and

4) Verifying that the heaters dissipate 37 ± 3 kW when tested in accordance with ANSI N510-1975.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing criteria of less than 1% for HEPA filters and 0.05% for charcoal adsorbers in accordance with ANSI N510-1975 (however Prerequisite Testing, Sections 8 and 9 shall be in accordance with ANSI N510-1980) for a DOP test aerosol while operating the system at a flow rate of 9000 cfm ±10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing criteria of less than 1% for HEPA filters and 0.05% for charcoal adsorbers in accordance with ANSI N510-1975 (however Prerequisite Testing, Sections 8 and 9 shall be in accordance with ANSI N510-1980) for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 9000 cfm ±10%.

3/4.7.8 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.8 All snubbers shall be OPERABLE. The only snubbers excluded from the requirement are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.8g on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.8 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these groups (inaccessible and accessible) may be inspected independently according to the schedule below. The first inservice visual inspection of each type of snubber shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all hydraulic and mechanical snubbers. If all snubbers of each type are found OPERABLE during the first inservice visual inspection, the second inservice visual inspection of that type shall be performed at the first refueling outage. Otherwise, subsequent visual inspections of a given type shall be performed in accordance with the following schedule:

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 6 months by performance of a yard loop and fire hydrant flush,

e. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel,

f. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

   1) Verifying that each pump develops at least 3300 gpm at a system pressure of 80 psig,

   2) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

   3) Verifying that the electric driven fire pump starts on a start signal initiated on decreasing header pressure of 75 psig and the diesel driven fire pump starts on a start signal on decreasing header pressure of 70 psig after 10 second time delay to avoid simultaneous start of both pumps.

g. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

4.7.10.1.2 Each fire pump diesel engine shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

   1) The fuel storage tank contains at least 200 gallons of fuel, and

   2) The diesel starts from ambient conditions and operates for at least 30 minutes on recirculation flow.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-1975, is within the acceptable limits specified in Table 1 of ASTM D975-1977 when checked for viscosity, water, and sediment; and

c. At least once per 18 months, during shutdown, by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

TABLE 3.7-3

FIRE HOSE STATIONS

BUILDING	ELEVATION	AREA	HOSE RACK
Auxiliary	1974	1122	KC-HR-051
Auxiliary	1974	1122	KC-HR-047
Auxiliary	1974	1120	KC-HR-031
Auxiliary	1974	1120	KC-HR-025#
Auxiliary	1974	1101	KC-HR-023#
Auxiliary	1974	1101	KC-HR-040
Auxiliary	1974	1101	KC-HR-042
Auxiliary	1988	1201	KC-HR-024
Auxiliary	2000	1329	KC-HR-111
Auxiliary	2000	1320	KC-HR-048
Auxiliary	2000	1320	KC-HR-046#
Auxiliary	2000	1314	KC-HR-030
Auxiliary	2000	1321	KC-HR-029#
Auxiliary	2000	1301	KC-HR-035#
Auxiliary	2000	1301	KC-HR-039
Auxiliary	2000	1301	KC-HR-041#
Auxiliary	2026	1408	KC-HR-049
Auxiliary	2026	1408	KC-HR-044
Auxiliary	2026	1408	KC-HR-032#
Auxiliary	2026	1408	KC-HR-026#
Auxiliary	2026	1401	KC-HR-034
Auxiliary	2026	1403	KC-HR-037#
Auxiliary	2047	1506	KC-HR-050
Auxiliary	2047	1513	KC-HR-043
Auxiliary	2047	1506	KC-HR-045
Auxiliary	2047	1501	KC-HR-038
Auxiliary	2047	1504	KC-HR-033
Auxiliary	2047	1502	KC-HR-027
Auxiliary	2064	1119	KC-HR-028#
Control	1974	3101	KC-HR-002#
Control	1974	3101	KC-HR-014#
Control	1984	3204	KC-HR-015#
Control	1984	3221	KC-HR-001#
Control	2000	3301	KC-HR-004#
Control	2000	3301	KC-HR-017#
Control	2000	3302	KC-HR-016#
Control	2016	3401	KC-HR-005
Control	2016	3401	KC-HR-019
Control	2016	3401	KC-HR-018

TABLE 3.7-3 (Continued)

FIRE HOSE STATIONS

BUILDING	ELEVATION	AREA	HOSE RACK
Control	2032	3501	KC-HR-006#
Control	2032	3501	KC-HR-020#
Control	2047	3604	KC-HR-007
Control	2047	3616	KC-HR-021
Control	2073	3801	KC-HR-008#
Control	2073	3801	KC-HR-022#
Reactor	2000	2201	KC-HR-120*
Reactor	2000	2201	KC-HR-131*
Reactor	2000	2201	KC-HR-124*
Reactor	2000	2201	KC-HR-129*
Reactor	2026	N.A.	KC-HR-121*
Reactor	2026	N.A.	KC-HR-132*#
Reactor	2026	N.A.	KC-HR-125*
Reactor	2026	N.A.	KC-HR-130*
Reactor	2047	N.A.	KC-HR-128*
Reactor	2047	N.A.	KC-HR-122*
Reactor	2047	N.A.	KC-HR-126*
Reactor	2068	N.A.	KC-HR-123*
Reactor	2068	N.A.	KC-HR-127*
Fuel	2000	6102	KC-HR-142#
Fuel	2000	6102	KC-HR-054#
Fuel	2000	6102	KC-HR-143
Fuel	2000	6104	KC-HR-057
Fuel	2026	6201	KC-HR-133
Fuel	2026	6203	KC-HR-052
Fuel	2047	6301	KC-HR-055#
Fuel	2047	6302	KC-HR-056#
Fuel	2047	6301	KC-HR-053#
ESW	2000	N.A.	KC-HR-140
ESW	2000	N.A.	KC-HR-141

TABLE NOTATIONS

#Secondary means of fire suppression to Water Sprays/Deluge or Halon Systems.
*Fire hose for station to be stored external to Reactor Building.

3) Verifying the fuel transfer pump starts and transfers fuel from the storage system to the day tank,

4) Verifying the diesel starts from ambient condition and accelerates to at least 514 rpm in less than or equal to 12 seconds.* The generator voltage and frequency shall be 4160 + 160 - 420 volts and 60 $\pm$ 1.2 Hz within 12 seconds* after the start signal. The diesel generator shall be started for this test by using one of the following signals:

a) Manual, or

b) Simulated loss-of-offsite power by itself, or

c) Safety Injection test signal.

5) Verifying the generator is synchronized, loaded to greater than or equal to 6201 kW in less than or equal to 60 seconds,* operates with a load greater than or equal to 6201 kW for at least 60 minutes, and

6) Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day tanks;

c. At least once per 31 days by checking for and removing accumulated water from the fuel oil storage tanks;

d. By sampling new fuel oil in accordance with ASTM D4057 prior to addition to storage tanks and:

(1) By verifying in accordance with the tests specified in ASTM D975-81 prior to addition to the storage tanks that the sample has:

(a) An API Gravity of within 0.3 degrees at 60°F or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89 or an API gravity of greater than or equal to 27 degrees but less than or equal to 39 degrees;

*These diesel generator starts from ambient conditions shall be performed only once per 184 days in these surveillance tests and all other engine starts for the purpose of this surveillance testing shall be preceded by an engine prelube period and/or other warmup procedures recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

(b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes, if gravity was not determined by comparison with the supplier's certification;

(c) A flash point equal to or greater than 125°F; and

(d) A clear and bright appearance with proper color when tested in accordance with ASTM D4176-82.

(2) By verifying within 30 days of obtaining the sample that the other properties specified in Table 1 of ASTM D975-81 are met when tested in accordance with ASTM D975-81 except that the analysis for sulfur may be performed in accordance with ASTM D1552-79 or ASTM D2622-82.

e. At least once every 31 days by obtaining a sample of fuel oil in accordance with ASTM D2276-78, and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM D2276-78, Method A.

f. At least once per 18 months, during shutdown, by:

1) Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service,

2) Verifying the diesel generator capability to reject a load of greater than or equal to 1352 kW (ESW pump) while maintaining voltage at 4160 + 160 - 420 volts and frequency at 60 ± 5.4 Hz,

3) Verifying the diesel generator capability to reject a load of 6201 kW without tripping. The generator voltage shall not exceed 4784 volts during and following the load rejection,

4) Simulating a loss-of-offsite power by itself, and:

a) Verifying deenergization of the emergency busses and load shedding from the emergency busses, and

b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 12 seconds, energizes the auto-connected shutdown loads through the shutdown sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 + 160 - 420 volts and 60 ± 1.2 Hz during this test.

5) Verifying that on a Safety Injection test signal without loss-of-offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes; and the offsite power source energizes the auto-connected emergency (accident) load through the LOCA sequencer. The generator voltage and frequency shall be 4160 + 160 - 420 volts and 60 ± 1.2 Hz within 12 seconds after the auto-start signal; the generator steady-state generator voltage and frequency shall be maintained within these limits during this test;

6) Simulating a loss-of-offsite power in conjunction with a Safety Injection test signal, and

   a) Verifying deenergization of the emergency busses and load shedding from the emergency busses;

   b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 12 seconds, energizes the auto-connected emergency (accident) loads through the LOCA sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 + 160 - 420 volts and 60 ± 1.2 Hz during this test; and

   c) Verifying that all automatic diesel generator trips, except high jacket coolant temperature, engine overspeed, low lube oil pressure, high crankcase pressure, start failure relay, and generator differential, are automatically bypassed upon loss of voltage on the emergency bus concurrent with a Safety Injection Actuation signal.

7) Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 6821 kW and during the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal to 6201 kW. The generator voltage and frequency shall be 4160 + 160 - 420 volts and 60 + 1.2 Hz, - 3 Hz within 12 seconds after the start signal; the steady-state generator voltage and frequency shall be maintained within 4160 ± 160 - 420 volts and 60 ± 1.2 Hz during this test. Within 5 minutes after completing this 24-hour test, perform Specification 4.8.1.1.2f.6)b)*;

*If Specification 4.8.1.1.2f.6)b) is not satisfactorily completed, it is not necessary to repeat the preceding 24-hour test. Instead, the diesel generator may be operated at 6201 kW for 1 hour or until operating temperature has stabilized.

A.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the Onsite Class 1E Distribution System, and

b. One diesel generator with:

1) A day tank containing a minimum volume of 390 gallons of fuel,

2) A fuel storage system containing a minimum volume of 85,300 gallons of fuel, and

3) A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel, or crane operation with loads over the spent fuel pool. In addition, when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the requirements of Specifications 4.8.1.1.1, 4.8.1.1.2 (except for Specification 4.8.1.1.2a.5)), and 4.8.1.1.3.

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following D.C. electrical sources shall be OPERABLE:

a. 125-Volt Battery Bank NK11 and NK13, and its associated Full Capacity Chargers NK21 and NK23, and

b. 125-Volt Battery Bank NK12 and NK14, and its associated Full Capacity Chargers NK22 and NK24.

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

ACTION:

With one of the required battery banks and/or full capacity chargers inoperable, restore the inoperable battery bank and/or full capacity charger to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1) The parameters in Table 4.8-2 meet the Category A limits, and

2) The total battery terminal voltage is greater than or equal to 130.2 volts on float charge.

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TABLE 4.8-2

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A[1]	CATEGORY B[2]	
PARAMETER	LIMITS FOR EACH DESIGNATED PILOT CELL	LIMITS FOR EACH CONNECTED CELL	ALLOWABLE[3] VALUE FOR EACH CONNECTED CELL
Electrolyte Level	>Minimum level indication mark, and < ¼" above maximum level indication mark	>Minimum level indication mark, and < ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts[6]	> 2.07 volts
Specific Gravity[4]	≥ 1.200[5]	≥ 1.195	Not more than 0.020 below the average of all connected cells
		Average of all connected cells > 1.205	Average of all connected cells ≥ 1.195[5]

TABLE NOTATIONS

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.

(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

(4) Corrected for electrolyte temperature and level.

(5) Or battery charging current is less than 2 amps when on charge.

(6) Corrected for average electrolyte temperature.

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
480-V Motor Control Center (Continued)	
P-52NG01BEF3 B-15A Fuse	Ctmt Recirc Sump Iso Vlv ENHV1
P-52NG01BGF3 B-60A Fuse	Accumulator Iso Vlv EPHV8808A
P-52NG01BGF2 B-60A Fuse	Accumulator Iso Vlv EPHV8808C
P-52NG01BFF2 B-15A Fuse	Ctmt Air to Aux Bldg ESF Filter Iso Vlv GSHV20
P-52NG01BBR2 B-15A Fuse	React Bldg Discharge Iso Vlv LFFV95
P-52NG02BBF3 B-40A Fuse	RHR Loop Inlet Iso Vlv BBPV8702B
P-52NG02BCF2 B-40A Fuse	RHR Loop Inlet Iso Vlv BBPV8702A
P-52NG02BHF3 B-15A Fuse	ESW to Ctmt Air Coolers Iso Vlv EFHV34
P-52NG01BCF2 B-15A Fuse	ESW to Ctmt Air Coolers Iso Vlv EFHV33
P-52NG01BDF2 B-15A Fuse	ESW from Ctmt Air Coolers Iso Vlv EFHV45
P-52NG01BEF2 B-40A Fuse	RHR Loop Inlet Iso Vlv EJHV8701A
P-52NG03CDF4 B-15A Fuse	RCP Thermal Barrier CCW Iso Valve BBHV13
P-52NG03CHF1 B-15A Fuse	RCP Thermal Barrier CCW Iso Vlv BBHV14
P-52PG19NAF4 B-150A Fuse	Reactor Cavity Cooling Fan DCGN02A
P-52PG19NCF3 B-60A Fuse	Ctmt Atmospheric Control System Fan DCGR01A

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
480-V Motor Control Center (Continued)	
P-52PG19NGF2 B-40 Fuse	RCP A Space Heater
P-52PG19NGF3 B-40 Fuse	RCP B Space Heater
P-52PG19NEF1 B-40A Fuse	RCP A Oil Lift Pump
P-52PG19NGR3 B-40A Fuse	RCP B Oil Lift Pump
P-52PG19NFF1 B-15A Fuse	Ctmt Normal Sump Pump DPLF05A
P-52PG19NFF2 B-15A Fuse	Ctmt Normal Sump Pump DPLF05C
P-52PG19NAF2 B-25A Fuse	Instrument Tunnel Sump Pump DPLF07A
P-52NG03CBF4 B-15A Fuse	RCP Thermal Barrier CCW Iso Vlv BBHV15
P-52NG03CLF2 B-15A Fuse	RCP Thermal Barrier CCW ISo Vlv BBHV16
P-52PG20NCR3 B-150A Fuse	Reactor Cavity Cooling Fan DCGN02B
P-52PG20NFF4 B-60A Fuse	Ctmt Atmospheric Control System Fan DCGR01B
P-52PG20NBF1 B-40A Fuse	RCP C Space Heater
P-52PG20NCF1 B-40A Fuse	RCP D Space Heater
P-52PG20NFF3 B-40A Fuse	RCP C Oil Lift Pump
1EPR08C P-3A Fuse RP139 B-3A Fuse	Accumulator Tank A Isol Vlv EPHV8808A

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
Low Voltage Power and Control (Continued)	
1EJG05B P-2A Fuse NG01BDF3 B-1A Fuse	RHR Shutdown Suction Line Isolation Valve EJHV8701B
1EJG06A P-2A Fuse NG01BFF3 B-1A Fuse	Cont Recirc Sump Isolation Valve EJHV8811A
4EJG06B P-3A Fuse NG02BEF2 B-2A Fuse	Cont Recirc Sump Isolation Valve EJHV8811B
1EJK07A P-3A Fuse RL017 B-3A Fuse	Test Line Isol Vlv Hot Leg Inj Line Solenoid EJHCV8825
1EJK07C P-3A Fuse RL017 B-3A Fuse	RHR Test Line Vlv EJHCV8890A
4EJK07B P-3A Fuse RL017 B-3A Fuse	RHR Test Line Vlv EJHCV8890B
P-1EJY13A 3A Fuse RL011 B-1RLY01G 15A Breaker NG01ACR119	Ctmt Sump Sample Isolation Vlv EJHV21
P-4EJY13B 3A Fuse RL011 B-4RLY01G 15A Breaker NG02ACR140	Ctmt Sump Sample Isolation Vlv EJHV22
P-4BMY01D 3A Fuse RL024 B-4RLY01H 15A Breaker NG02ACR127	S.G.C Cnt to Nuc Sample Sys Vlv BMHV22
P-4BMY02A 3A Fuse RL024 B-4RLY01H 15A Breaker NG02ACR127	S.G.A Tube Sheet Sample Vlv BMHV35
P-4BMY02B 3A Fuse RL024 B-4RLY01H 15A Breaker NG02ACR127	S.G.B Tube Sheet Sample Vlv BMHV36

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

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PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
Low Voltage Power and Control (Continued)	
1BBK40A P-15A Fuse NK5108 B-15A Fuse	PZR PORV BBPCV455A
4BBK40B P-15A Fuse NK4421 B-15A Fuse	PZR PORV BBPCV456A
5BGK04B P-3A Fuse RL001 B-3A Fuse	Alternate Charging Path Isol Valv BGHV8147
6BGK04A P-3A Fuse RL001 B-3A Fuse	Normal Charging Path Isol Valv BGHV8146
P-5LFY10A 3A Fuse RL023 B-5RLY01H 15A Breaker PG19GCR217	Containment Cooler Drain Valve LFLV97
P-5LFY10C 3A Fuse RL023 B-5RLY01H 15A Breaker PG19GCR217	Containment Cooler Drain Valve LFLV99
P-6LFY10B 3A Fuse RL023 B-6RLY01G 15A Breaker PG20GBR217	Containment Cooler Drain Valve LFLV98
P-6LFY10D 3A Fuse RL023 B-6RLY01G 15A Breaker PG20GBR217	Containment Cooler Drain Valve LFLV100
P-6LFY17A 3A Fuse RL023 B-6RLY01G 15A Breaker PG20GBR217	Refueling Pool Stand Pipe Discharge Valve LFLV122
P-5LFY20A 15A Breaker PG19NHF224 B-5LFY20A 30A Fuse PG19NHF1	Instrument Tunnel Sump Moisture Sensor TLVF01

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
Low Voltage Power and Control (Continued)	
6EPK05E P-3A Fuse RL018 B-3A Fuse	Accumulator Water Fill Vlv EPHV8878B
6EPK05F P-3A Fuse RL018 B-3A Fuse	Accumulator Water Fill Vlv EPHV8878D
P-4SJY01B 3A Fuse RL011 B-4RLY01G 15A Breaker NG02ACR140	Press. Vapor. Cont. Iso. Space Vlv. SJHV12
P-4SJY01C 3A Fuse RL011 B-4RLY01G 15A Breaker NG02ACR140	Accums Sample Cont Isol Vlv SJHV18
P-5SJY03B 3A Fuse RP211 B-5RPY09D 15A Breaker PG19NHF236	Accumulator Sample Line Vlv SJHV16
P-5SJY03C 3A Fuse RP211 B-5RPY09D 15A Breaker PG19NHF236	Accumulator Sample Line Vlv SJHV17
P-5SJY04B 3A Fuse RP211 B-5RPY09D 15A Breaker PG19NHF236	Accumulator Sample Line Vlv SJHV14
P-5SJY04C 3A Fuse RP211 B-5RPY09D 15A Breaker PG19NHF236	Accumulator Sample Line Vlv SJHV15
P-1SJY06B 3A Fuse RP332 B-1RPY09F 15A Breaker NG01BAR140	HL Sample 3 Vlv SJHV4
P-4SJY06A 3A Fuse RP333 B-4RPY09F 15A Breaker NG02BAR140	HL Sample 1 Vlv SJHV3

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
Low Voltage Power and Control (Continued)	
P-5SJY06C 3A Fuse RP211 B-5RPY09D 15A Breaker PG19NHF236	Press Liquid Space Samp Isol Vlv SJHV20
P-4BMY01A 3A Fuse RL024 B-4RLY01H 15A Breaker NG02ACR127	S.G. A Out to Nuc Sample Sys Vlv BMHV19
P-4BMY01B 3A Fuse RL024 B-4RLY01H 15A Breaker NG02ACR127	S.G. B Out to Nuc Sample Sys Vlv BMHV20
P-4BMY01C 3A Fuse RL024 B-4RLY01H 15A Breaker NG02ACR127	S.G. C Out to Nuc Sample Sys Vlv BMHV21
P-5GNY08A 3A Fuse RL020 B-5RLY01L 15A Breaker PG19GCR230	CRDM Cooling Discharge Damper GNHZ71
P-5GNY08C 3A Fuse RL020 B-5RLY01L 15A Breaker PG19GCR230	CRDM Cooling Discharge Damper GNHZ73
P-6GNY08C 3A Fuse RL020 B-6RLY01J 15A Breaker PG20GBR222	CRDM Cooling Discharge Damper GNHZ72
P-6GNY08A 3A Fuse RL020 B-6RLY01J 15A Breaker PG20GBR222	CRDM Cooling Discharge Damper GNHZ74
5BGK10A P-3A Fuse RL001 B-3A Fuse	Normal Letdown Isolation Vlv BGLCV459

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
Low Voltage Power and Control (Continued)	
5BBK14C P-3A Fuse RK021 B-3A Fuse	RCP Standpipe Makeup Vlv BBLCV180
5BBK14D P-3A Fuse RK021 B-3A Fuse	RCP Standpipe Makeup Vlv BBLCV181
6BBK14A P-3A Fuse RL021 B-3A Fuse	RCP Standpipe Makeup Vlv BBLCV178
6BBK14B P-3A Fuse RL021 B-3A Fuse	RCP Standpipe Makeup Vlv BBLCV179
5BBK15B P-3A Fuse RL021 B-3A Fuse	Reactor Coolant Loops Instrumentation
5BBK15C P-3A Fuse RL021 B-3A Fuse	Reactor Coolant Loops Instrumentation
6BBK15D P-3A Fuse RL021 B-3A Fuse	Reactor Coolant Loops Instrumentation
6BBK15E P-3A Fuse RL021 B-3A Fuse	Reactor Coolant Loops Instrumentation
5BBK19A P-3A Fuse RL002 B-3A Fuse	Pressurizer Spray Valve BBPCV455B
5BBK19B P-3A Fuse RL002 B-3A Fuse	Pressurizer Spray Valve BBPCV455C
1BBK30A P-3A Fuse RL021 B-3A Fuse	Reactor Vessel Head Vent Vlv BBHV8001A
6GNG03B P-5A Fuse NG02BJF5 B-3A Fuse	CRDM Cooling Fan B Discharge Isolation Damper GNHZ42
6GNG03D P-5A Fuse PG20GAR2 B-3A Fuse	CRDM Cooling Fan A Discharge Isolation Damper GNHZ41

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
Low Voltage Power and Control (Continued)	
1HBK03A P-3A Fuse RL021 B-3A Fuse	RCDT Vapor Space CTMT Isol Vlv HBHV7126
6HBK04A P-3A Fuse HB115 B-3A Fuse	RCDT Vapor Space CTMT Isol Vlv HBHV7127
5EPY07B P-3A Fuse RP043 B-15A CB-1	Accumulator Tank Discharge Valve Position Switch EPHV8808DA EPHV8808BA
6EPY07A P-3A Fuse RP044 B-15A CB-1	Accumulator Tank Discharge Valve Position Switch EPHV8808AA EPHV8808CA
6GTY12A P-15A Breaker PG20GBR134 B-20A Fuse	CTMT Minipurge Exhaust Isolation Damper GTHZ41
6GTY12A P-15A Breaker PG20GBR134 B-20A Fuse	CTMT Minipurge Exhaust Isolation Damper GTHZ42
P-5SRY09A 5A Fuse SR057 B-5SRY09A 20A Breaker PG19GEF6	In-Core Neutron Monitoring Drive Unit Heater SR01A, B
P-5SRY09A 5A Fuse SR057 B-5SRY09A 20A Breaker PG19GEF6	In-Core Neutron Monitoring Drive Unit Heater SR01C, D

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3/4.9.13 EMERGENCY EXHAUST SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.13 Two independent Emergency Exhaust Systems shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the spent fuel pool.

ACTION:

a. With one Emergency Exhaust System inoperable, fuel movement within the fuel storage areas or crane operation with loads over the fuel storage areas may proceed provided the OPERABLE Emergency Exhaust System is in operation and discharging through at least one train of HEPA filters and charcoal adsorbers.

b. With no Emergency Exhaust System OPERABLE, suspend all operations involving movement of fuel within the fuel storage areas or crane operation with loads over the fuel storage areas until at least one Emergency Exhaust System is restored to OPERABLE status.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.13 The above required Emergency Exhaust Systems shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;

b. At least once per 18 months, or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system, by:

   1) Verifying that the Emergency Exhaust System satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% for HEPA filters and 0.05% for charcoal adsorbers and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 9000 cfm ±10% at $\geq$ 7.2 inches W.G. (dirty filter);

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revisison 2, March 1978, for a methyl iodide penetration of less than 1%; and

3) Verifying a system flow rate of 9000 cfm ±10% at $\geq$ 7.2 inches W.G. (dirty filter) during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%;

d. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than or equal to 7.2 inches Water Gauge while operating the system at a flow rate of 9000 cfm ±10%.

2) Verifying that on a Fuel Building Exhaust Gaseous Radioactivity-High test signal, the system automatically starts (unless already operating) and directs its exhaust flow through the HEPA filters and charcoal adsorber banks and isolates the normal fuel building exhaust flow to the auxiliary/fuel building exhaust fan;

3) Verifying that the system maintains the Fuel Building at a negative pressure of greater than or equal to 1/4 inches Water Gauge relative to the outside atmosphere during system operation; and

4) Verifying that the heaters dissipate 37 ± 3 kW when tested in accordance with ANSI N510-1975.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% for HEPA filters and 0.05% for charcoal adsorbers in accordance with ANSI N510-1975 (however Prerequisite Testing, Sections 8 and 9 shall be in accordance with ANSI N510-1980) for a DOP test aerosol while operating the system at a flow rate of 9000 cfm ±10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% for HEPA filters and 0.05% for charcoal adsorbers in accordance with ANSI N510-1975 (however Prerequisite Testing, Sections 8 and 9 shall be in accordance with ANSI N510-1980) for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 9000 cfm ±10%.

3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of the following requirements may be suspended:

a. Specification 3.4.1.1 - During the performance of startup and PHYSICS TESTS in MODE 1 or 2 provided:

1) The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

2) The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set less than or equal to 25% of RATED THERMAL POWER.

b. Specification 3.4.1.2 - During the performance of hot rod drop time measurements in MODE 3 provided at least three reactor coolant loops as listed in Specification 3.4.1.2 are OPERABLE.

APPLICABILITY: During operation below the P-7 Interlock Setpoint or performance of hot rod drop time measurements.

ACTION:

a. With the THERMAL POWER greater than the P-7 Interlock Setpoint during the performance of startup and PHYSICS TESTS, immediately open the Reactor trip breakers.

b. With less than the above required reactor coolant loops OPERABLE during performance of hot rod drop time measurements, immediately place two reactor coolant loops in operation.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint at least once per hour during startup and PHYSICS TESTS.

4.10.4.2 Each Intermediate and Power Range channel, and P-7 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating startup and PHYSICS TESTS.

4.10.4.3 At least the above required reactor coolant loops shall be determined OPERABLE within 4 hours prior to initiation of the hot rod drop time measurements and at least once per 4 hours during the hot rod drop time measurements by verifying correct breaker alignments and indicated power availability.

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TABLE 4.11-1 (Continued)

TABLE NOTATIONS

(1)The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection (microCuries per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22×10^6 = the number of disintegrations per minute per microCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (s^{-1}), and

Δt = the elapsed time between the midpoint of sample collection and the time of counting (s).

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

(2)A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed by a method described in plant procedures to assure representative sampling.

TABLE 4.11-2 (Continued)

TABLE NOTATIONS (Continued)

(2)The principal gamma emitters for which the LLD specification applies include the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 in noble gas releases and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, I-131, Cs-134, Cs-137, Ce-141, and Ce-144 in iodine and particulate releases. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.7, in the format outlined in Regulatory Guide 1.21, Appendix B, Revision 1, June 1974.

(3)Sampling and analysis shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15% of RATED THERMAL POWER within 1 hour period.

(4)Tritium grab samples shall be taken and analyzed at least once per 24 hours when the refueling canal is flooded.

(5)Tritium grab samples shall be taken and analyzed at least once per 7 days from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool. Grab samples need to be taken only when spent fuel is in the spent fuel pool.

(6)The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2, and 3.11.2.3.

(7)Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing, or after removal from sampler. For unit vent, sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15% of RATED THERMAL POWER within a 1-hour period, and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement does not apply if: (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the reactor coolant has not increased more than a factor of 3, and (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3.

(8)Continous sampling of the spent fuel building exhaust needs to be performed only when spent fuel is in the spent fuel pool.

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LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 2.5×10^5 Curies of noble gases (considered as Xe-133 equivalent).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and, within 48 hours, reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.7.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 7 days when radioactive materials are being added and within 7 days following any addition of radioactive material to the tank.

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
2. Airborne			
Radioiodine and Particulates	Samples from five locations Three samples from close to the three SITE BOUNDARY locations, in different sectors, of the highest calculated annual average ground level D/Q. One sample from the vicinity of a community having the highest calculated annual average ground-level D/Q. One sample from a control location, as for example 15 to 30 km (10 to 20 mile) distant and in the least prevalent wind direction.[3]	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.	Radioiodine Cannister: I-131 analysis weekly. Particulate Sampler: Gross beta radioactivity analysis following filter change;[4] and gamma isotopic analysis[5] of composite (by location) quarterly.
3. Waterborne			
a. Surface	One sample upstream.[6] and sample downstream.	Monthly grab sample	Gamma isotopic analysis[5] monthly. Composite for tritium analysis quarterly.
b. Ground	Samples from one or two sources only if likely to be affected.[8]	Quarterly.	Gamma isotopic[5] and tritium analysis quarterly.
c. Drinking	One sample of each of one to three of the nearest water supplies that could be affected by its discharge. One sample from a control location.	Composite sample over 2-week period[7] when I-131 analysis is performed, monthly composite otherwise.	I-131 analysis on each composite when the dose calculated for the consumption of the water is greater than 1 mrem per year.[9] Composite for gross beta and gamma isotopic analyses[5] monthly. Composite for tritium analysis quarterly.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission, that correspond to samples required by Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.3 The Interlaboratory Comparison Program shall be described in the ODCM. A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6.

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

c. The control rod insertion limits of Specification 3.1.3.6 are maintained, and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. As noted on Figure 3.2-3, RCS flow rate and $F^N_{\Delta H}$ may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the measured $F^N_{\Delta H}$ is also low) to ensure that the calcu lated DNBR will not be below the design DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

R as calculated in Specification 3.2.3 and used in Figure 3.2-3, accounts for $F^N_{\Delta H}$ less than or equal to 1.49. This value is used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad tem perature, and thus is the maximum "as measured" value allowed.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic margins, totaling 9.1% DNBR, completely offset any rod bow penalties. This margin includes the following:

1) Design limit DNBR of 1.30 vs. 1.28,
2) Grid spacing (K_s) of 0.046 vs. 0.059,
3) Thermal Diffusion Coefficent of 0.038 vs. 0.059,
4) DNBR Multiplier of 0.86 vs. 0.88, and
5) Pitch Reduction.

The applicable values of rod bow penalties are referenced in the FSAR.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System, and a 3% allowance is appropriate for manufacturing tolerance.

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

The Radial Peaking Factor, $F_{xy}(Z)$, is measured periodically to provide assurance that the Hot Channel Factor, $F_Q(z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) as provided in the Radial Peaking Factor Limit Report per Specification 6.9.1.9 was determined from expected power control manuevers over the full range of burnup conditions in the core.

When RCS flow rate and $F_{\Delta H}^{N}$ are measured, no additional allowances are necessary prior to comparison with the limits of Figure 3.2-3. Measurement errors of 2.1% for RCS total flow rate and 4% for $F_{\Delta H}^{N}$ have been allowed for in determination of the design DNBR value.

The measurement error for RCS total flow rate is based upon performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venture which might not be detected could bias the result from the precision heat balance in a non-conservative manner. Therefore, an inspection is performed of the feedwater venture each refueling outage.

The 12-hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation shown on Figure 3.2-3. This surveillance also provides adequate monitoring to detect any core crud buildup.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective ACTION is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such ACTION does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

3/4.3.3.8 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that both adequate warning capability is available for the prompt detection of fires and that Fire Suppression Systems, that are actuated by fire detectors, will discharge extinguishing agents in a timely manner. Prompt detection and suppression of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

Fire detectors that are used to actuate fire suppression systems represent a more critically important component of a plant's fire protection program than detectors that are installed solely for early fire warning and notification. Consequently, the minimum number of operable fire detectors must be greater.

The loss of detection capability for Fire Suppression Systems, actuated by fire detectors represents a significant degradation of fire protection for any area. As a result, the establishment of a fire watch patrol must be initiated at an earlier stage than would be warranted for the loss of detectors that provide only early fire warning. The establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.3.9 LOOSE-PART DETECTION INSTRUMENTATION

The OPERABILITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the Reactor Coolant System and avoid or mitigate damage to Reactor Coolant System components. The allowable out-of-service times and Surveillance Requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.3.10 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.11 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The Alarm/Trip Setpoints for these instruments shall be adjusted to values calculated in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the WASTE GAS HOLDUP SYSTEM. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50. The sensitivity of any noble gas activity monitor used to show compliance with the gaseous effluent release requirements of Specification 3.11.2.2 shall be such that concentrations as low as 1 x 10-6 μCi/cc are measurable.

3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Although the orientation of the turbine is such that the number of potentially damaging missiles which could impact and damage safety-related components, equipment, or structures is minimal, protection from excessive turbine overspeed is required.

STEAM GENERATOR (Continued)

Unscheduled inservice inspections are performed on each steam generator following; 1) reactor to secondary tube leaks; 2) seismic occurrence greater than the Operating Basis Earthquake; 3) a loss-of-coolant accident requiring actuation of the Engineered Safety Features, which for this specification is defined to be a break greater than that equivalent to the severance of a 1" inside diameter pipe, or, for a main steamline or feedline, a break greater than that equivalent to a steam generator safety valve failing open; to ensure that steam generator tubes retain sufficient integrity for continued operation. Transients less severe than these do not require inspections because the resulting stresses are well within the stress criteria established by Regulatory Guide 1.121 which unplugged steam generator tubes must be capable of withstanding.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission pursuant to Specification 6.9.2 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

PRESSURE/TEMPERATURE LIMITS (Continued)

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\ K_{IM} + K_{It} \leq K_{IR} \qquad (2)$$

Where: K_{IM} = the stress intensity factor caused by membrane (pressure) stress,

K_{It} = the stress intensity factor caused by the thermal gradients,

K_{IR} = a function of temperature relative to the RT_{NDT} of the material as provided by the Code,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{It}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the ΔT developed

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TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

COMPONENT	COMP CODE	ASME MATERIAL TYPE	Cu (%)	P (%)	T_{NDT} (°F)	50 FT-LB 35 Mil Temp (°F)	RT_{NDT} (°F)	AVG. UPPER SHELF NMWD (FT-LB)	AVG. UPPER SHELF MWD (FT-LB)
Closure Head Dome	R2516-1	A533B, CL.1	0.12	0.010	-40	60	0	112	----
Closure Head Torus	R2515-1	A533B, CL.1	0.11	0.009	-20	<40	-20	119	----
Closure Head Flange	R2504-1	A508, CL.2	----	0.013	20	<80	20	139	----
Vessel Flange	R2501-1	A508, CL.2	----	0.012	20	<80	20	102	----
Inlet Nozzle	R2502-1	A508, CL.2	----	0.010	-20	<40	-20	147	----
Inlet Nozzle	R2502-2	A508, CL.2	----	0.009	-20	<40	-20	137	----
Inlet Nozzle	R2502-3	A508, CL.2	0.11	0.010	-20	<40	-20	156	----
Inlet Nozzle	R2502-4	A508, CL.2	0.11	0.010	-30	<30	-30	156	----
Outlet Nozzle	R2503-1	A508, CL.2	----	0.006	-10	<50	-10	126	----
Outlet Nozzle	R2503-2	A508, CL.2	----	0.009	0	<60	0	129	----
Outlet Nozzle	R2503-3	A508, CL.2	----	0.007	0	<60	0	136	----
Outlet Nozzle	R2503-4	A508, CL.2	----	0.007	0	<60	0	114	----
Nozzle Shell	R2004-1	A533B, CL.1	0.05	0.010	-40	70	10	118	----
Nozzle Shell	R2004-2	A533B, CL.1	0.04	0.011	-40	80	20	121	----
Nozzle Shell	R2004-3	A533B, CL.1	0.04	0.008	-50	60	0	133	----
Inter. Shell	R2005-1	A533B, CL.1	0.04	0.008	-20	<40	-20	127	156
Inter. Shell	R2005-2	A533B, CL.1	0.04	0.007	-30	40	-20	127	143
Inter. Shell	R2005-3	A533B, CL.1	0.05	0.007	-30	40	-20	135	164
Lower Shell	R2508-1	A533B, CL.1	0.09	0.009	-40	60	0	87	118
Lower Shell	R2508-2	A533B, CL.1	0.06	0.008	-10	70	10	100	127
Lower Shell	R2508-3	A533B, CL.1	0.07	0.008	-20	100	40	86	127
Bottom Head Torus	R2517-1	A533B, CL.1	0.11	0.010	-80	30	-30	92	----
Bottom Head Dome	R2518-1	A533B, CL.1	0.12	0.009	-60	0	-60	154	----
Inter. & Lower Shell Long. Weld Seams	G2.06	SAW	0.04	0.006	-50	<10	-50	150	----
Inter. to Lower Shell Girth Weld Seam	E3.16	SAW	0.05	0.007	-50	10	-50	98	----

NMWD - Normal to Major Working Directions

MWD - Major Working Directions

3/4.5 EMERGENCY CORE COOLING SYSTEMS

BASES

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2, 3/4.5.3, and 3/4.5.4 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

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.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or 0.75 L_t, as applicable, during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

For reduced pressure tests, the leakage characteristics yielded by measurements L_{tm} and L_{am} shall establish the maximum allowable test leakage rate L_t of not more than $L_a (L_{tm}/L_{am})$. In the event L_{tm}/L_{am} is greater than 0.7, L_t shall be specified as equal to $L_a (P_t/P_a)^{1/2}$

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR Part 50.

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.

3/4.6.1.7 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 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 pressure control or the reduction of airborne radioactivity to facilitate personnel access for surveillance and maintenance activities, should be used to support the 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 supply 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 of Specification 3.6.1.2.b. 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 consistent with the assumptions used in the safety analyses.

The Containment Spray System and the Containment Cooling System are redundant to each other in providing post-accident cooling of the containment atmosphere. However, the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable Spray System to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the Spray Additive System ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between 8.5 and 11.0 for the

SPRAY ADDITIVE SYSTEM (Continued)

solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained solution volume limit includes an allowance for solution not usable because of tank discharge line location or other physical characteristics. The educator flow test of 52 gpm with RWST water is equivalent to 40 gpm NaOH solution. These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses.

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 GDC54 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 detection and 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 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. 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.

The Hydrogen Mixing Systems are provided to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

ULTIMATE HEAT SINK (Continued)

The limitations on minimum water level and maximum temperature are based on providing a 30-day cooling water supply from the Essential Service Water pumps to safety-related equipment without exceeding its design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," March 1974.

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

The OPERABILITY of the Control Room Emergency Ventilation System ensures that: (1) the ambient air temperature does not exceed the allowable temperature for continuous-duty rating for the equipment and instrumentation cooled by this system, and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. Operation of the system with the heaters operating to maintain low humidity using automatic control for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the charcoal adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rems or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50. ANSI N510-1975 and N510-1980 will be used as procedural guides for surveillance testing.

3/4.7.7 EMERGENCY EXHAUST SYSTEM

The OPERABILITY of the Emergency Exhaust System ensures that radioactive materials leaking from the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the environment. Operation of the system with the heaters operating to maintain low humidity using automatic control for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the charcoal adsorbers and HEPA filters. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses. ANSI N510-1975 and N510-1980 will be used as procedural guides for surveillance testing.

A.C. SOURCES, D.C. SOURCES, AND ONSITE POWER DISTRIBUTION (Continued)

The Surveillance Requirement for demonstrating the OPERABILITY of the Station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and 0.015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to 7 days. During this 7-day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than 0.040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.12 SPENT FUEL ASSEMBLY STORAGE

The restrictions placed on spent fuel assemblies stored in Region 2 of the spent fuel pool ensure inadvertent criticality will not occur.

3/4.9.13 EMERGENCY EXHAUST SYSTEM

The limitations on the Emergency Exhaust System ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. Operation of the system with the heaters operating to maintain low humidity with automatic control 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. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. ANSI N510-1975 and N510-1980 will be used as procedural guides for surveillance testing.

DOSE RATE (Continued)

assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR 20.106(b)). For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrems/year to the whole body or to less than or equal to 3000 mrems/year to the skin. These release rate limits also restrict, at all times the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

The required detection capabilities for radioactive materials in gaseous waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.11.2.2 DOSE - NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the ODCM for calculating the doses due to the actual release rates of radioactive materials in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled

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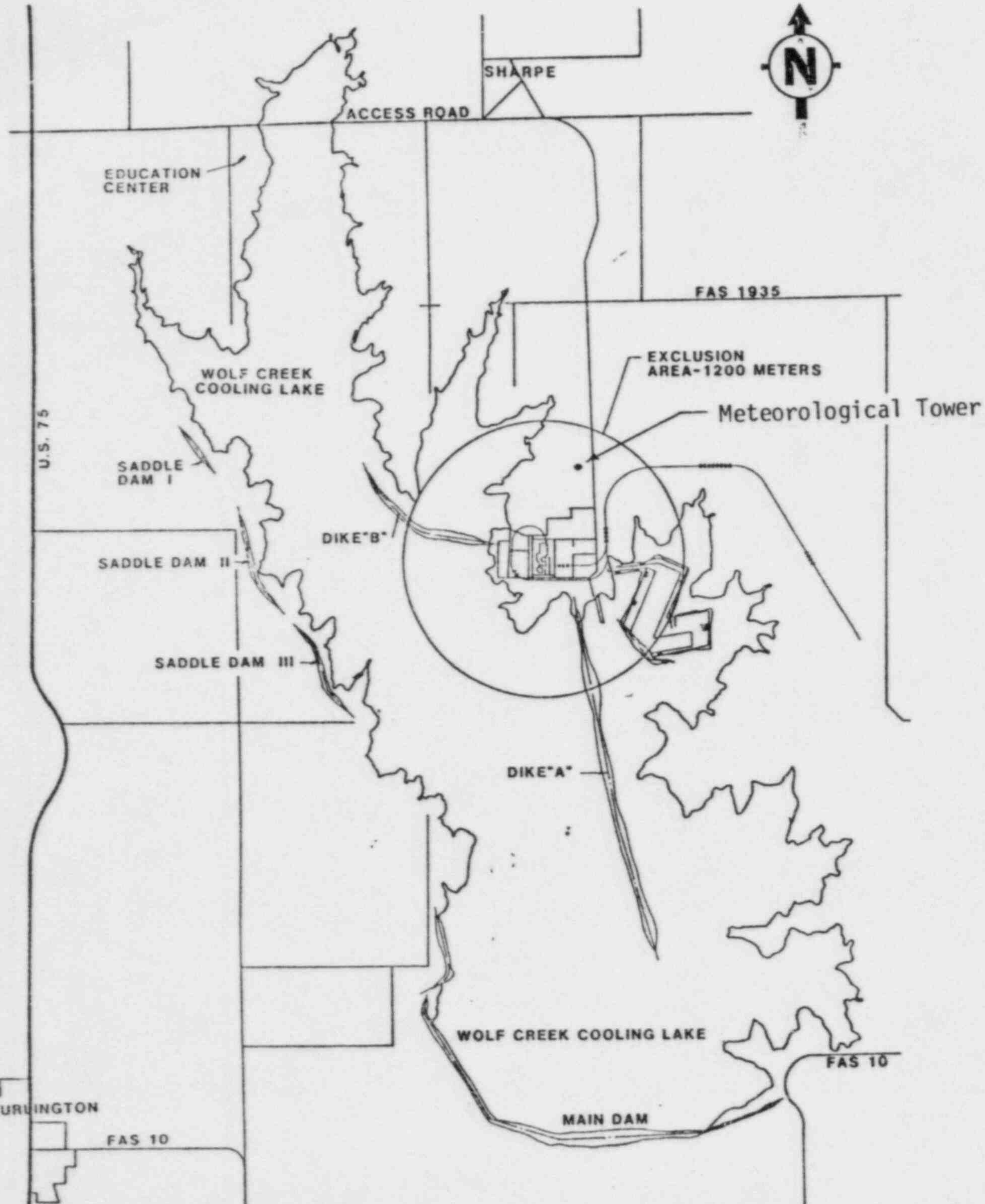

Note:

1. The exclusion-restricted area is a 1200 meter radius circle centered around Unit 1 containment.

FIGURE 5.1-1

EXCLUSION AREA

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at ≤ 100°F/h and 200 cooldown cycles at ≤ 100°F/h.	Heatup cycle - T_{avg} from ≤ 200°F to ≥ 550°F. Cooldown cycle - T_{avg} from ≥ 550°F to ≤ 200°F.
	200 pressurizer cooldown cycles at ≤ 200°F/h.	Pressurizer cooldown cycle temperatures from ≥ 650°F to ≤ 200°F.
	80 loss of load cycles, without immediate Turbine or Reactor trip.	≥ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss-of-offsite A.C. electrical power.	Loss-of-offsite A.C. electrical ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 Reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	10 auxiliary spray acutation cycles.	Spray water temperature differential > 320°F.
	50 leak tests.	Pressurized to ≥ 2485 psig.
	5 hydrostatic pressure tests.	Pressurized to ≥ 3106 psig.
Secondary Coolant System	1 large steam line break.	Break in a > 6-inch steam line.
	5 hydrostatic pressure tests.	Pressurized to ≥ 1350 psig.

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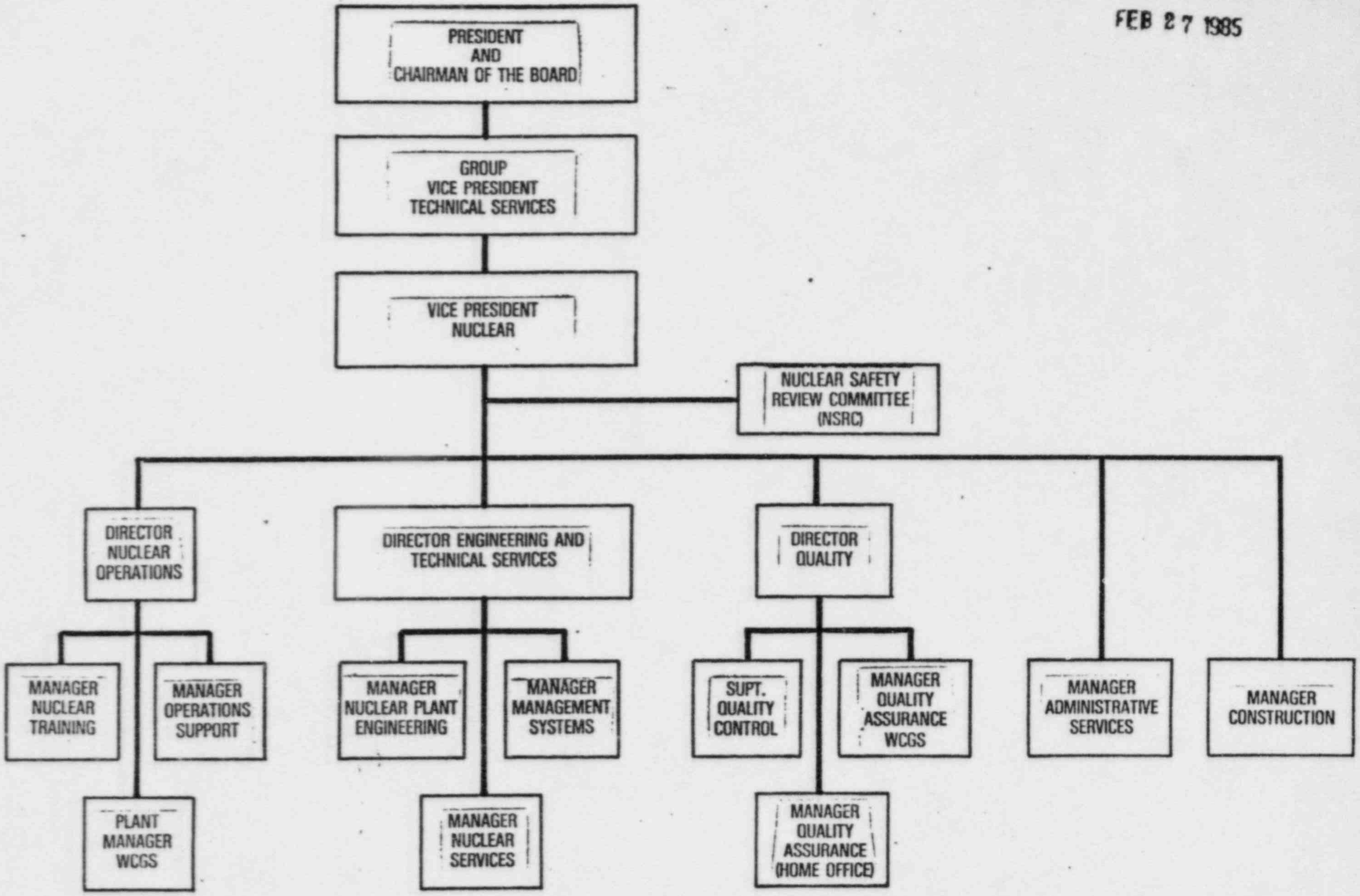

FIGURE 6.2-1

OFFSITE ORGANIZATION

FINAL DRAFT

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- Plant Manager
  - Super of Maint
    - Maint Svcs Supvr
      - Maint Engr
      - Mat Cont Supvr
        - Mat Coord Supvr
        - Storeroom Supvr
          - Warehouse Att
      - Utility Supvr
        - Utility Mech
        - Bldg Svcman
    - Maint Supp Supvr
      - Mech Supvr
        - Welder
        - Machinist
        - Mechanic
        - Utility Hlpr
      - Electrical Supv
        - Electrician
        - Utility Hlpr
  - Supt of Operations
    - Ops Coord - Ops#
      - Shift Supervisor
        - Supervising Opr
        - Reactor Operator
        - Station Operator
        - Utility Helper
    - Ops Coord - Planning and Projects
      - Engineer/Spec
      - Surv Coord
        - System Analyst
  - Supt of Tech Supp
    - Reactor Eng Supvr
      - Reactor Engineer
    - I&C Supervisor
      - Engineer/Spec
      - I&C Coordinator
        - I&C Specialist
        - I&C Technician
        - Utility Helper
      - Lead Computer Eng
        - Comp Eng/Spec
        - App Programmer
    - Site Chemist
      - Chemistry Supvr
        - HP/Chem Tech
        - Utility Helper
    - Site Health Phys*
      - HP Supervisor
        - HP/Chem Tech
        - Utility Helper
  - Supt of Plant Supp
    - Fire Prot Spec*
      - Training Spec
    - Chief of Security*
      - Security Ops Supv
      - Lieutenant
        - Sergeant
        - Officer
    - Data Mgmt Supvr
      - Data Base Analyst
      - Asst Supv Comp Ops
        - Technician
        - Computer Opr
      - Asst Supv Sys Sftwr
      - Data Comm Engr Spec
        - Comp Sys Tech
        - Clerk
    - Results Eng Supvr
      - Results Eng (M)
      - Results Eng (E)
  - Supt of Regulatory, Quality and Admin Services
    - Engineering Spec
    - Safety Specialist
    - Nuclear Medical Spe
    - Site Emerg Planning Administrator
    - Document Cont Sup
      - Document Cont Cl
      - Analysis & Film Op
    - Administrative Supvr
      - Clerk
  - Startup Manager
    - Startup Organization

*For technical matters of an immediate nature the respective individual reports directly to the Plant Manager.

#This position requires an SRO License.

FIGURE 6.2-2 UNIT ORGANIZATION

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODE 1, 2, 3, or 4	MODE 5 or 6
SS	1	1*
SRO	1	None
RO	2	1
SO	4	1
STA	1**	None
CHM	1	None

SS - Shift Supervisor with a Senior Operator license on Unit 1
SRO - Individual with a Senior Operator license on Unit 1
RO - Individual with an Operator license on Unit 1
SO - Station Operator
STA - Shift Technical Advisor
CHM - Chemistry Personnel

The Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the control room while the unit is in MODE 1, 2, 3, or 4, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Shift Supervisor from the control room while the Unit is in MODE 5 or 6, an individual with a valid Operator license (other than the Shift Technical Advisor) shall be designated to assume the control room command function.

*One SRO, either Shift Supervisor or Supervising Operator.

**The STA position shall be manned in MODES 1, 2, 3, and 4 unless the Shift Supervisor or the individual with a Senior Operator license meets the qualifications for the STA as required by the NRC.

FUNCTION (Continued)

g. Mechanical and electrical engineering, and

h. Quality assurance practices.

The NSRC shall report to and advise the Vice President-Nuclear on those areas of responsibility specified in Specifications 6.5.2.7 and 6.5.2.8.

COMPOSITION

6.5.2.2 The NSRC shall be composed of at least the following:

Chairman:	Manager Nuclear Services
Member:	Manager Nuclear Plant Engineering
Member:	Manager Quality Assurance (Home Office)
Member	Director Nuclear Operations
Member:	Manager Licensing and Radiological Services
Member:	Vice President-Engineering
Member:	Manager Nuclear Safety

Additional members and Vice Chairman may be appointed by the Chairman.

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the NSRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in NSRC activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the NSRC Chairman to provide expert advice to the NSRC.

MEETING FREQUENCY

6.5.2.5 The NSRC shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per 6 months thereafter.

QUORUM

6.5.2.6 The quorum of the NSRC necessary for the performance of the NSRC review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least two-thirds of the NSRC members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the Unit.

HIGH RADIATION AREA (Continued)

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

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them, or

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Site Health Physicist in the RWP.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels greater than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Supervisor/Supervising Operator on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, direct or remote (such as closed-circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mR/h that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.